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EXPERIMENTAL RESEARCH ON TOROIDAL DEVICES AT PPPL: 1960-1985

Papers Submitted
for the
Twenty-Fifth Anniversary Issue of Nuclear Fusion

JUNE 1985

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EXPERIMENTAL RESEARCH ON TOROIDAL DEVICES AT PPPL: 1960-1985

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EXPERIMENTS ON THE MODEL C STELLARATOR

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ABSTRACT

The Model C Stellarator was the last and most powerful of a series of stellarator devices operated at Princeton during the period from 1953 to 1969. Experiments on Model C addressed the problem of anomalous plasma loss and tested major innovative concepts including the divertor and radio-frequency (rf) heating. This review focuses on portions of Model C work that have strongly influenced present-day research and understanding of toroidal confinement in stellarators and tokamaks.

1. INTRODUCTION

Planning for the Model C Stellarator started in 1954 under the direction of Lyman Spitzer, originator of the stellarator concept [1], and machine operation began in 1961. Machine parameters are given in Table I. The configuration of the device was that of a racetrack, namely, two semicircular U-bends connected by two straight sections. One of the straight sections later contained the toroidal divertor and the other the coil and shield box for the ion-cyclotron resonance heating (ICRH) section, Fig. 1. The $l = 2$ and $l = 3$ helical windings installed on the U-bends provided a modest total rotational transform, up to about 180° . The total circumference along the magnetic axis was about 1200 cm, while the radius of the circularized plasma cross section (inside a circular limiter) varied from 5 to 7.5 cm, producing an equivalent aspect ratio, $L/2a$, in the range 30-40. The equilibrium and stability β limits were both below 1%. Typical operating plasma pressures were sufficiently low so that these limits did not set significant constraints.

A detailed review of Model C experiments has been given by Young [2]; studies of the plasma fluctuations were summarized in Ref. [3].

2. TOKAMAK-LIKE CONDITIONS

In initial operation, the Model C Stellarator used neither a divertor nor helical windings. A rotational transform was generated, as in a tokamak, solely by the ohmic-heating current in the plasma flowing along the lines of

magnetic force. Under typical operating conditions, the toroidal field was 35 kG, the plasma radius 5 cm, and the ohmic current 2 kA, leading to a value of the safety factor, q , of about 10. The working gas was admitted at a constant rate to the vacuum chamber. The plasma density, after its initial rise at the beginning of an ohmic discharge, decayed monotonically with time. The rate of density decay gave an upper limit to the particle containment time, while the decrease in loop voltage tracked the decrease in plasma resistance due to the ohmic heating of the plasma electrons. An increase in loop voltage at the end of the pulse typically marked the onset of a plasma disruption.

Initially, the conditions for the onset of disruption depended on whether the directions of the plasma current and the toroidal magnetic field were parallel or anti-parallel. Following a suggestion from Spitzer, error-field correction coils were placed around the stellarator [4]. Subsequent experiments not only confirmed the utility of these correction units, but also demonstrated the need for "vertical" fields to optimize the plasma performance [5,6] and led to an understanding of toroidal equilibria of current-carrying plasmas, complementing that achieved by groups in the USSR and the United Kingdom [7-9].

3. PLASMA PARTICLE CONFINEMENT

In an informal description of the experimental results from an early set of experiments on magnetically confined electric arcs, Bohm [10] had proposed a plasma diffusion coefficient

$$D = \alpha \frac{kT_e c}{eB} \quad (1)$$

with $\alpha = 1/16$, using cgs-Gaussian units. Applying this formula to a plasma of cylindrical cross section in which the density varies with radius as a Bessel function, $n(r) = n_0 J_0(\Lambda r/a)$, where $\Lambda \sim 2.4$, and setting α at $1/23$, one finds a global value for the particle confinement time

$$\tau = \left(\frac{a}{5}\right)^2 \frac{B}{T_e} \quad (2)$$

This simple empirical formula, with τ in ms, a in cm, B in kG and T_e in eV, was found to be in remarkable agreement with Model C data taken over many years in experiments with a broad range of plasmas, magnetic fields, and electron temperatures, Fig. 2. Confinement scaling in accord with Eq. (1), with $\alpha = 1/5$, had been found earlier in a detailed analysis by Stodiek, et al. [12] of the loss rates in Model C's predecessor machine, the B-3 Stellarator (2.5 cm radius, 648-cm-long stainless-steel vacuum chamber, $B_0 \leq 40$ kG). The observation of Bohm-like diffusion in both the B-3 and C Stellarators was discouraging for extrapolation to fusion conditions in that the confinement time appears to scale only linearly with magnetic field rather than quadratically, and to scale inversely with electron temperature rather than directly with $T_e^{1/2}$, as classical theory would predict. In retrospect, however, one notes that the confinement time in the experiments reported by

Pease, et al. [13] may have exceeded the formula in Eq. (2) by a factor at least as large as 5. Since the electron temperature was inferred from the plasma conductivity, a serious underestimate may have been made: the small amount of Thomson scattering data taken late in the life of Model C showed that the true electron temperatures were sometimes considerably higher [14]. Confinement times up to 5 times longer than Bohm were also found by Grove, et al. [15] in stellarator operation with long-mean-free-path gun-injected or microwave-formed plasmas.

4. STELLARATOR OPERATION

Finite-length helical windings were placed on both U-bends of Model C to provide the externally imposed rotational transform characteristic of a true stellarator. Both $l = 2$ and $l = 3$ windings were used, but the winding laws were modified in "circularizer" sections located at the ends of the U-bends, reshaping the cross sections of the magnetic surfaces from ellipses and trefoils, respectively, to approach concentric circles.

While the total external rotational transform was only about 180° , Sinclair, et al. [16] found that even this modest value led to clearly observable changes in the MHD stability of the Model C plasma. Also, it was now possible to confine plasmas with no ohmic current at all, as reported in Ref. 15. Moreover, Harries, et al. [17] demonstrated that the positional stability of tokamak-type plasmas carrying finite ohmic current was improved by the stellarator transform.

5. EXPERIMENTS ON RADIO-FREQUENCY HEATING

Apparatus for heating the plasma by electromagnetic waves in the ion-cyclotron frequency range was installed in one of the two straight sections of Model C. A periodic induction coil ($m = 0$, two wavelengths, $\lambda = 38.1$ cm) was placed just outside an 8-inch-i.d. ceramic cylinder. Up to 4 MW of 25-MHz radio-frequency power was available to feed the coil during a 5-ms pulse. The induction coil generated a slow Alfvén wave, also called the "ion cyclotron wave," that appears first underneath the coil and then propagates out in both directions along the magnetic field lines. Beyond the coil, the confining magnetic field strength was slightly reduced, forming "magnetic beach" regions [18] where the local value of the ion-cyclotron frequency approached the rf frequency, and the wave energy was absorbed by the collision-free ion-cyclotron damping mechanism or by electron Landau damping.

First attempts at ion-cyclotron resonance heating on Model C were frustrated by poor coupling to the ion-cyclotron wave and by an anomalous disappearance of rf power into a mode then called "Mode X." (In retrospect, it is probable that Mode X was the cold-plasma electrostatic ion-cyclotron surface wave.) Coupling to Mode X was greatly reduced by the installation of a Faraday shield [19] which eliminated the component of the rf electric field parallel to the magnetic field lines, while leaving the other two rf components unaffected.

Ion-cyclotron-wave absorption was observed in Model C experiments, accompanied by both ion and electron heating [20,21], the latter being

observed for the first time in these experiments. After persevering effort by the experimental team, a truly impressive degree of ion heating was achieved [22], namely, a global average plasma ion temperature of 400 eV.

Even higher ion temperatures were attained locally, adjusting the magnetic beach configuration so that the heated ions would be mirror-confined in a short axial section of the plasma. Reference [20] cites local ion temperatures in the range 2 to 3 keV, while a deuterium ion temperature of 9 keV is reported for a deuterium-minority ICRH experiment [23]. It is interesting to note that in the case of mirror-confined plasmas the decay of the local pressure was in reasonable agreement with classical mirror theory [24]. Evidently, the filling of the loss cone by the lower-temperature main toroidal plasma served to stabilize the mirror regions against interchange and velocity-space modes.

Radio-frequency current drive was also observed in these experiments [21]. As shown in Fig. 3, the application of an rf pulse to a preionized plasma led to the generation of a unidirectional (ohmic) plasma current. Under conditions where the wave propagation was expected to be symmetric, this observed current could be attributed to bootstrap effects, but by energizing the magnetic beaches asymmetrically, current could be driven in either toroidal direction. The observed generation of current could be explained only in terms of the direct acceleration of electrons by the rf waves.

6. MAGNETIC ISLANDS

Only in the later stages of operation was it realized that plasma confinement in Model C might be seriously affected by magnetic islands. Detailed studies were then made by numerical calculation of the magnetic fields due to the actual coil configurations, and directly by means of electron beams. The theoretical and experimental determinations of the surfaces were in excellent agreement [25], and it was found that island formation could be strongly exacerbated using vertical fields in an attempt to shift the magnetic surfaces formed by the helical windings, Fig. 4. On the other hand, the islands that occurred for normal field settings were not of sufficient size to account directly for the anomalous plasma losses in Model C. Probe measurements [26] seemed to show that, at least with finite ohmic heating current, plasma loss in Model C could be accounted for by fluctuations.

7. OTHER EXPERIMENTS

A toroidal divertor, placed in the straight section of the racetrack across from the ICRH section, was successful in improving plasma purity and reducing recycling from the walls [2]. A special series of experiments was also carried out with alkali-metal plasmas formed by contact ionization of a beam of neutral alkali-metal atoms on a hot tungsten sphere suspended in the vicinity of the magnetic axis. The experimental evidence did not, however, support a clear conclusion as to whether confinement was significantly improved relative to the Bohm law, Eq. (2).

8. CONCLUSION

The experiments on the Model C Stellarator established a strong foundation in the U.S. for the investigation of the physics of toroidal magnetic confinement. In addition, innovative concepts including impurity reduction by a divertor and plasma heating by ion-cyclotron-wave absorption and electron Landau damping were significantly advanced through experiments on this machine. Contributions from Model C research to today's experimental plasma physics also include meticulous attention to the mechanical and vacuum engineering of the apparatus, utilization of extensive and purposely redundant diagnostics, systematic application of computers to the acquisition and analysis of data, and -- perhaps most important -- emphasis on the close interplay of theory and experiment.

ACKNOWLEDGMENTS

The Model C Stellarator represented an enormous advance in size and complexity for a plasma research facility, and its design, construction, and operation were the work of teams of dedicated physicists, engineers, and machine technicians. We would like to acknowledge their essential contributions and we wish also to recognize the underlying leadership of Lyman Spitzer, who founded Project Matterhorn, now called the Princeton Plasma Physics Laboratory; of Melvin B. Gottlieb, Laboratory Director from 1961-1980, and of Donald J. Grove, Physicist in Charge of Model C, 1959-69.

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Table I. Model C Stellarator Parameters.

Total Length (Racetrack)	1200 cm
Minor Radius	5 - 7.5 cm
B_t (Kilogauss)	10 - 50 (typically to 35) kG
Ohmic Heating Current	0 to 8 kA
RF Heating Power (at 25 MHz)	to 4 MW and to 5 ms
Density Range	to 10^{14} cm^{-3}
Temperature	to 400 eV

Fig. 1. Top view of the Model C Stellarator. Two helical windings, a divertor, and an rf coil are in view. Two transformers were used to induce a modest ohmic-heating current. From Ref. [20]. (653092)

Fig. 2. Product of characteristic plasma confinement time, τ_c (in milliseconds) and electron temperature, kT_e (in eV) versus toroidal field, B (in kilogauss). Data from hydrogen and helium discharges at 1.5 and 3.0 kA, $a = 5$ cm. Equation (2) corresponds to slope = 1.0. From Bishop and Hinnav [11]. (653382)

Production of DC current by Ion Cyclotron Resonance Wave

Fig. 3. Production of (dc) plasma current by ion-cyclotron waves. Hydrogen ion density was $2 \times 10^{12} \text{ cm}^{-3}$. The gas was broken down by an rf-oscillator (100 kHz) and two ICRH pulses were applied to the plasma. From Ref. [21]. (653521)

Fig. 4. Intentionally distorted magnetic surfaces in Model C, measured (upper) and computed (lower). To achieve an accurate count of the number of toroidal orbits, bunching of the multi-pass electron beams was maintained by phase stabilization. A strong vertical magnetic field, applied to shift the magnetic surfaces towards increasing major radius, produced much larger deviations of the magnetic surfaces for this illustration than was the case for normal Model C operation. From Ref. [24]. (693471)

EXPERIMENTS ON THE ST TOKAMAK

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ABSTRACT

The confinement time on the ST Tokamak was proportional to the plasma current and density, and reached values of 10-20 ms near the disruption limit. The confinement showed the effect of neoclassical, single-particle behavior for nonaxisymmetry. The Spitzer conductivity was found considering Z_{eff} from measured impurities. Runaway electron production was classical, the cross-field transport anomalous. The MHD modes, internal (sawtooth) and major disruptions, and island structures for the $m=2$ modes were analyzed with soft X-ray imaging techniques. Ion-cyclotron heating at second harmonic operation increased the plasma temperature by 100 eV.

1. INTRODUCTION

In 1970, the C Stellarator facility was modified into an (axi)symmetric copper-shell tokamak -- thereby becoming the first major tokamak device outside the USSR. At plasma currents above 10 kA, electron temperatures exceeded 100 eV [1], implying that the energy confinement time clearly exceeded values previously observed with the C Stellarator (see Fig. 1). In subsequent work on ST, it was shown that magnetic asymmetries, up to $\pm 30\%$, produced by a local toroidal-field inhomogeneity resulted in a decrease of only 30% in the energy confinement time; however, depletion of the energetic tail of the ion distribution confirmed the neoclassically-predicted losses of trapped ions in nonsymmetric magnetic geometries [2].

2. STABILITY

Two types of MHD-like instability were observed: (1) slow-growing small-amplitude oscillatory modes; and (2) fast-growing, sometimes destructive, instabilities that limited the attainable q-value. Both phases of instabilities pointed to the importance of the plasma current profile. The mode structure and amplitude of the slow-growing oscillations were determined with multiple magnetic probes, microwave interferometry, beam probes, and X-ray emission [3,4].

Detailed studies of these instabilities were carried out by examining the fluctuations in the continuum X-ray emission [5]. This technique detected -- for the first time in tokamak research -- the existence of $n=0$, $m=0$ internal disruptions, often preceded by small $n=1$, $m=1$ oscillations (see Fig. 2). Large $m=2$ internal oscillations were also observed, preceding a disruptive instability. In the experimental arrangement, an image was formed by a slot aperture, and the X-ray emission from different regions of the plasma was measured by silicon-surface-barrier detectors placed in the image plane. Near the center of the plasma, the internal disruption manifested itself as a "sawtooth-like" oscillation, which was inverted a small distance away from the center; the inversion radius occurred near the (calculated) location of the $q = 1$ surface. The growing $n=1$, $m=1$ sinusoidal oscillation, the amplitude of which was very small in the center, had a maximum at the $q = 1$ surface, and vanished just outside. The energy transported through the $q = 1$ surface during such an internal disruption amounted to about half of the ohmic-heating power, and limited the thermal collapse of the current channel: a marginally-stable thermal equilibrium was maintained in which the $q = 1$ surface lay just outside the plasma center. The observations were later found to be in general agreement with the predicted behavior of the resistive $m=1$ tearing mode [6].

Large $m=2$ oscillations were also observed, and generally preceded a large-scale, external disruption. External disruptives were accompanied by sharp changes in average X-ray intensity, negative voltage spikes, and even termination of the plasma current. The X-ray fluctuations implied a structure of rotating $m=2$ magnetic islands (see Fig. 3); again, the observations were in general agreement with predictions of tearing-mode theory [7].

3. CONFINEMENT

Radial temperature profiles were produced by Thomson scattering of a ruby-laser pulse [1]. For typical plasma conditions, T_e was 800-2200 eV at the center. Hydrogenic ion temperatures were measured by charge-exchange neutrals, and reached peak values of 400-500 eV. Impurity ion temperatures, from Doppler broadening of OVII and CIV lines, were in good agreement with the hydrogenic ion temperatures. The plasma resistance was not significantly different from the Spitzer value, using a Z_{eff} -value from the measured content of impurity ions (primarily oxygen) and incorporating neoclassical corrections.

Over a range of two orders of magnitude, the plasma energy content was found to vary approximately as I_p^2 . The machine voltage did not vary much, implying that the energy confinement time was approximately proportional to I_p . At constant I_p , the confinement time increased about proportionally with n_e [8].

Early in the discharge, the electron temperature showed evidence of a skin-like profile that developed on a timescale of several milliseconds into a centrally-peaked profile.

The average particle confinement time scaled with $T_e^{3/2}$, and reached a maximum of 13-14 ms [9]. The power balance was dominated by electron thermal losses and particle recycling; energy losses by electrons constituted about three-quarters of the total loss and were divided about equally between line radiation and thermal conduction. The recycling processes at the limiter

involved sufficiently energetic neutral atoms so that a fairly flat radial particle source function was produced [10]. Improved T_e and energy confinement with aluminum limiters was the result of more favorable radial-current and power-input distributions, rather than a reduction in direct energy losses by radiation [11].

4. IMPURITIES

Quantitative vacuum ultraviolet (UV) measurements yielded an effective ionic charge, Z , which was consistent with the Z_{eff} -value inferred from plasma resistivity. The radial distribution of high- Z materials was obtained from X-ray measurements [12].

The slopes of the soft X-ray emission spectra were in good agreement with the electron temperature measured by Thomson scattering. The absolute intensity of X-ray emission, which was a factor of 5 to 100 above hydrogenic bremsstrahlung, was quantitatively accounted for by recombination radiation from oxygen and heavy-metal impurities. These results, as well as those from experiments in which high- Z gases (Xe, Kr, A) were pulsed into the discharge, indicated that within the measuring accuracy the radial profile of Z_{eff} was very flat [13].

5. RUNAWAY ELECTRONS

The runaway electron flux, as determined from plasma bremsstrahlung, was found to be less intense than predicted from theoretical work (including the effect of impurities). Production of runaway electrons occurred mainly at the plasma center; a bell-shaped runaway-density profile implied radial transport, with confinement times of 10-20 ms. Runaway transport occurred in the MHD stable regime, but was especially large at the low- q limit, suggesting the possibility that degeneration of the magnetic surfaces was responsible for the loss [14,15].

6. AUXILIARY HEATING

Wave generation by two antennae at the fundamental and the second harmonic of the ion-cyclotron frequency provided initial tests of rf heating of a tokamak at power levels up to 1 MW [16,17]. The predicted wave-generation efficiencies of about 90% were observed. The ion-cyclotron (slow) wave was strongly damped, and the fast hydromagnetic wave propagated around the machine and produced standing toroidal eigenmodes. With relatively short rf pulses at the second-harmonic of the cyclotron frequency, the ion temperature increased from about 100 eV to about 200 eV, corresponding to a heating efficiency of about 20%. At the fundamental cyclotron frequency, the ion temperature increased by only 25% under similar conditions. Longer rf pulses led to pronounced neutral-particle influx and MHD instabilities [18].

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Fig. 1. Particle and energy confinement time and temperature in the ST-Tokamak and the C-Stellarator, for various values of aperture-limiter setting and plasma current. The $T_e^{3/2}$ line is drawn approximately through the upper points. The T_e^{-1} line is the Bohm-loss line for $B_t = 43 \text{ kG}$ and $a = 5 \text{ cm}$ (C-stellarator parameters).

Fig. 2. Radial profiles of electron temperature and density for Thomson scattering data (limiter at $r = 12 \text{ cm}$); the safety factor $q(r)$ from laser data; amplitude $\Delta A/A$ of the internal disruption (sawtooth); amplitude $\Delta \tilde{A}/A$ of the sinusoidal $m=1$ oscillations preceding the sawtooth. Inset oscillograms illustrate the phase relations between various radial positions. X-ray data are intensities integrated over the chord of observation; laser data are for local radii. (743154)

THE ATC TOKAMAK

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ABSTRACT

The Adiabatic Toroidal Compressor (ATC) successfully demonstrated major-radius compression of ohmic and auxiliary-heated tokamak plasmas. Plasma densities well above 10^{14} cm^{-3} were reached, along with central ion and electron temperatures above 1 keV.

1. INTRODUCTION

The demonstration in the T-3 tokamak of relatively long-lifetime plasmas with temperatures in the keV range encouraged attempts at the further extension of tokamak parameters. The ATC (Adiabatic Toroidal Compressor) was proposed in 1970 for the investigation of (1) supplemental plasma heating by major-radius compression and (2) control of the plasma equilibrium and stability by means of externally generated poloidal fields, rather than by a close-fitting conducting shell. The original objectives of ATC were achieved within the first six months of operation. The ATC capabilities were then utilized to achieve a number of additional advances that have turned out to be important for tokamak physics and technology.

Yoshikawa and Furth [1] showed that significant enhancements in tokamak parameters could be achieved by compressing the plasma in major radius. For this purpose, the axial (vertical) magnetic field is increased, while the toroidal field can remain constant. The conservation of q (safety factor) then implies a reduction of the minor radius a , so that the plasma volume decreases as shown in Table I. If the compression time is small compared with the energy-containment time of the system, the process can be treated as adiabatic. If, in addition, the compression time is longer than the ion-ion and electron-electron collision times, the velocity distributions will be isotropic and near-thermal, and the adiabatic exponent will be $\gamma = 5/3$. The scalings shown in Table I appeared to offer a relatively straightforward approach to increasing the density and temperature of a tokamak plasma on a transient basis, and the ATC experiment was proposed. A theoretical analysis by Greene, Johnson, and Weimer showed that the evolution of the tokamak equilibrium would probably be benign [2]. The ATC had machine parameters somewhat smaller than those of T-3, as shown in Table II.

A major-radius compression experiment requires a substantial change in major radius and a minimum of highly conducting material near the vacuum vessel. To satisfy these requirements, ATC was designed without a close-fitting conducting shell. The vacuum vessel was a resistive stainless-steel bellows. A special air-core ohmic-heating transformer was designed to concentrate the initial poloidal flux in the part of the vacuum chamber where the plasma would be located at the end of compression. The coils were placed in such a way that, when the ohmic-heating transformer was charged to a field of about 3 T, the field was only a few gauss in the volume occupied by the uncompressed plasma. Compression was timed to occur when the ohmic transformer current was near zero, thus limiting the allowable magnitude of the flux swing. No attempt was made to maintain the compressed plasma beyond the few milliseconds (5-10 ms) required to measure its properties. Figure 1 shows the field configuration of ATC and some experimental data [3]. A summary of the results for compression experiments is given in Table III.

The parameters listed in Table I scaled according to the adiabatic laws, except for the electron temperature, which tended to scale linearly with $C = R_0/R_f$ rather than with $C^{4/3}$, where R_0 and R_f are the initial and final values of the major radius. A breakdown in the criterion for adiabatic behavior, however, was not clearly demonstrated, because of impurity effects and insufficient separation between the compression time and the energy confinement time. (Subsequent work on Tuman-3 and TFTR suggests that transient phenomena caused by the compression process itself can shorten the confinement time.)

Auxiliary heating of ions in ATC plasmas began with some of the first neutral-beam-injection experiments in a tokamak. The sources, which were provided by the Lawrence Berkeley Laboratory, could deliver up to 10-ampere-equivalent neutral-beam current of deuterium or hydrogen atoms at 20 kV, but were operated most often at 15 kV. In most experiments, two sources were mounted to provide co- and counter-tangential injection into the uncompressed plasma. Some experiments on perpendicular injection were also carried out. The results obtained were entirely consistent with the classical theory of energy transfer from fast ions to the plasma electrons and ions, and direct loss by charge exchange. A summary of the 200-kW neutral-beam-heating results is given in Table III [4].

Ion-cyclotron heating, near the second harmonic of deuterium, was carried out at 25 MHz with an rf-generator developed for the ST tokamak. The antenna configuration was a single 130° poloidal arc, located on the outer (low-field) side of the plasma torus. No Faraday shield was used. The antenna was encased in a ceramic material (MACOR) and placed between two stainless-steel bumper limiters to reduce interaction with the plasma. The maximum power that could be applied without inducing a disruption was 200 kW for 10 ms. The ion-heating results are shown in Table III. These results supported the conjecture that the loss of ions with high perpendicular energy had been a serious limitation in the first ST ICRH heating experiments [5], which were limited to low plasma currents.

Lower-hybrid heating experiments were performed at a frequency of 800 MHz, with powers up to 150 kW. Variably-phased double and quadruple waveguides, as well as a single waveguide were used to test the linear theory of accessibility. Qualitative agreement with the theory was established. There was evidence for an ion temperature increase of about 100 eV when 100 kW

of power was applied, as well as evidence for electron heating by Landau damping [6].

In addition to the major auxiliary-heating experiments, an important series of pioneering investigations was carried out on ATC. A technique for investigation of transport processes by injection of various impurities by laser desorption was developed. The diffusion of the high-Z ions was evaluated by comparing the variation in time and space of the vacuum ultraviolet radiation from the highly ionized states of the injected species. Comparison of the data with results obtained from numerical codes led to the conclusion that the transport of impurities was in reasonable agreement with neoclassical theory [7].

Some of the first data on the internal density-fluctuation spectra in a tokamak plasma were obtained on ATC by scattering of 70-GHz microwaves [8] and by forward-scattering of a CO₂ laser beam [9]. The experimental results yielded spatial resolution in frequency and wavelength in the ranges 10⁴-10⁶ Hz and 0.4 to 2 cm, respectively, for the 70-GHz source, and in the ranges 10⁴-10⁹ Hz and 0.1 to 1 cm for the CO₂ laser source. The observed fluctuation magnitudes $\Delta n/n$ were in the range 10⁻⁴ to 10⁻². These fluctuations could be interpreted as drift waves: they propagated in the electron diamagnetic-drift direction and had maximal amplitudes for $k\rho_i \approx 1$ (where ρ_i is the ion gyroradius). Their poloidal structure was similar to their radial structure, and there was strong localization in the outer half of the plasma torus. The amplitude of these fluctuations was found to be sufficient, if properly phased with potential fluctuations, to account for most of the observed electron energy loss in ATC.

Several other investigations on ATC can only be mentioned briefly. The use of titanium gettering for control of low-Z impurities in tokamaks was given its first trial [10]. The application of fast-neutral measurements for the detailed determination of the orbits and spectra of beam-injected ions permitted estimates of $q(r)$ and Z_{eff} [11]. Measurements of electron-cyclotron emission at the fundamental and higher harmonics were carried out to investigate runaway electrons [12]. Hsuan and Bol [13] demonstrated that the plasma would tolerate the cyclical translation between two regions of stable vertical-field index at small and large major radius, separated by an unstable region. This technique was proposed as a practical means for implementing the original suggestion of L.A. Artsimovich that a forced oscillation of the tokamak major radius could be used to heat tokamak plasmas by magnetic pumping [14].

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Table I. Adiabatic Compression Scaling.
(Major radius compression with B_T constant in time.)

<u>Parameters</u>	<u>Scaling^a</u>
Toroidal Field	C
Temperature	$C^{4/3}$
Density	C^2
Major Radius	C^{-1}
Minor Radius	$C^{-1/2}$
Volume	C^{-2}

$$^a C = R_{\text{initial}}/R_{\text{final}}$$

Table II. ATC Design Parameters.

	Initial <u>Plasma</u>	Compressed <u>Plasma</u>
Major radius (m)	0.88	0.38
Minor radius (m)	0.17	0.11
Plasma current (kA)	50	118
Toroidal field (T)	2	4.74
Current duration (ms)	~40	~5-10
Compression ratio C		1.34
Compression time (ms)		~2

Table III. ATC Results.

<u>Parameters</u>	<u>Ohmic Heating</u>		<u>Neutral-Beam Heating</u>		<u>ICRF Heating</u>	
			(230 kW)		(200 kW)	
	<u>Initial</u>	<u>Compressed</u>	<u>Initial</u>	<u>Compressed</u>	<u>Initial</u>	<u>Compressed</u>
	<u>Plasma</u>	<u>Plasma</u>	<u>Plasma</u>	<u>Plasma</u>	<u>Plasma</u>	<u>Plasma</u>
T_{eo} (keV)	0.85	1.95	0.85	1.95	0.85	1.95
T_{io} (keV)	0.2	0.6	0.4	1.2	0.35	1.0
n_{eo} (10^{13} cm^{-3})	2.2	12	3.7	20	2	10
τ_{comp} (ms)	~2	--	~2	--	~2	--
τ_E (ms)	~2	--	~2	--	~2	--
I_p (kA)	60	145	60	145	60	145

Fig. 1. Magnetic field patterns and plasma profiles. Upper drawing is a vertical section showing the rail limiters, the design locations of the initial and compressed plasma, and magnetic field plots for the transformer field B_{OH} (top half), and the "vertical" field B_V (bottom half). Lower plots show results of Thomson scattering measurements (T.S.) before and after compression (32 ms). Actual compressed plasma location is 40 ± 1 cm. Peak ion (deuteron) temperature deduced from fast neutrals is indicated by T_i (C.X.). (723486)

THE PRINCETON LARGE TORUS (PLT)

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ABSTRACT

In the PLT experimental program, ohmic, neutral-injection, and ion-cyclotron heating techniques have been studied and improved, resulting in record ion temperatures. In addition, noninductive current-drive with lower-hybrid waves has been demonstrated. These achievements were made possible by the continual development and deployment of advanced plasma diagnostics, high-power rf systems, and specially-tailored materials and structures. The intermediate size of PLT has permitted flexible experimental operation and the separation of plasma core from plasma surface effects.

1. OHMIC AND NEUTRAL-BEAM HEATING

The Princeton Large Torus (PLT) became operational in 1975. With a major radius of 1.32 m and a minor radius of 0.40 m, it constituted a significant new step in tokamak size scaling. The first results were reported at the IAEA (Berchtesgaden) Conference in 1975: electron energy confinement times in the range of 40 ms had been obtained, at a line-average density of $3.5 \times 10^{13} \text{ cm}^{-3}$ [1].

During the following year, experiments on PLT were aimed at maximizing the plasma density and controlling impurity influx. The original PLT limiters were made of tungsten, chosen for its refractory properties and low sputtering yield. While the total amount of tungsten present in PLT plasmas was low, it proved to be a very efficient radiator: X-ray and spectroscopic techniques were used to identify the tungsten spectrum, a strong quasi-continuum in the 50 Å range. Bolometric measurements indicated nearly 100% radiated power in the worst cases, with the radiation profile strongly peaked at the plasma center. In such cases, the electron temperature profiles were frequently hollow; these discharges exhibited poor global energy confinement, although they proved interesting from the point of view of MHD stability, microscopic fluctuations, and impurity transport. Metallic impurity radiation was reduced in PLT by using controlled gas feed to contract the plasma column during start-up [2]. High-density helium and deuterium plasmas were obtained in this manner, and confinement times of close to 100 ms were observed at $n_e(0) = 1 \times 10^{14} \text{ cm}^{-3}$. "Taylor" discharge cleaning [3] and titanium gettering of the torus walls were employed extensively, yielding Z_{eff} values close to unity in this high-density range (see Fig. 1) [4].

Much emphasis was placed on the development of new diagnostic techniques. A multi-point image-intensified Thomson scattering system [5] was used to provide single-shot radial profiles of T_e and n_e . Soft and ultra-soft X-ray imaging arrays were used to study MHD activity and disruptions [6]. In preparation for auxiliary heating experiments, extensive ion diagnostics were developed; active charge-exchange techniques and a neutron detector system with wide dynamic range were used to improve the quality of central ion-temperature measurements. Doppler broadening of "forbidden" impurity lines [7] also proved to be a valuable diagnostic for ion temperature, requiring only near-ultraviolet spectroscopy. X-ray crystal spectroscopy was developed, both for determining charge-state distributions of metallic impurities and for measuring the ion temperature from Doppler-broadened K_α lines [8].

The first neutral-beam heating experiments on PLT were performed using two of the final complement of four tangential beamlines. The ion sources were supplied by the Oak Ridge National Laboratory, and would eventually provide a heating capability of 3 MW of 40 keV D^0 beams, with all four beamlines operating; two beamlines were oriented in the direction of the plasma current (co-injection) and two were opposed (counter-injection). With metallic limiters, especially during counter-injection, strong centrally peaked radiation was observed [9], and the electron temperature could fall during injection. A strong density rise was observed, considerably greater than the beam-particle influx. These two effects together served to clamp the ion temperature in the vicinity of 2 keV.

In the winter of 1977-8, the remaining two neutral beams were added, the tungsten limiters were replaced with carbon limiters, and a capability for inter-shot titanium gettering was installed. Central impurity radiation was dramatically reduced, although it was still significant at the lowest plasma densities. With titanium-coated walls, the rate of density rise during injection was comparable to the beam-ion influx rate. Under these circumstances the central ion temperature rose linearly with P_b/n_e at a rate of about 4.5 keV/(MW/ 10^{13} cm^{-3}) to values of up to 6.5 keV (see Fig. 2) [10]. The electron temperature rose somewhat, despite the considerable density rise, and the global energy confinement was unchanged when compared to similar low-density ohmically-heated plasmas. The ion collisionality entered decisively into the banana regime for the first time in tokamak research, with ν_{*i} reaching a minimum of 2×10^{-2} .

Damping of toroidal momentum was investigated by using unbalanced tangential beam injection [10,11]. It was found that the spinup, spindown, and equilibrium rotation rates were consistent with an anomalous momentum loss rate comparable in magnitude to the rate of loss of electron thermal energy.

Low-density operation in 1980 resulted in a record central ion temperature of 7 keV [12]. These discharges were interesting in that a substantial fraction of the central ion density was calculated to be composed of fast ions still in the process of slowing down. At higher densities, $\bar{n}_e = 3 \times 10^{13} \text{ cm}^{-3}$, the central electron thermal diffusivity was seen to decrease with injection, but the global energy confinement fell modestly. At these densities, uncertainties in the ion power balance were minimized, and the central ion thermal diffusivity was estimated to be at most three times the neoclassical value. At the highest attainable densities, $\bar{n}_e \approx 5.7 \times 10^{13} \text{ cm}^{-3}$, the central electron energy confinement was unchanged with injection heating, but the global electron energy confinement fell by a factor

of 1.8. This result prefigured later observations of the absence of energy confinement scaling with density in neutral-beam-heated plasmas.

Equally interesting was a series of experiments aimed at assessing the role of beam penetration in neutral-beam heating. In these experiments, 40 keV D^0 neutral beams were injected tangentially into high-density plasmas; the calculated beam-heating profile was hollow. While the electron heating efficiency was lower than with better penetrating H^0 beams, the shape of the electron temperature profile was essentially unchanged (see Fig. 3).

During this period, extensive hydrogenic and impurity particle transport studies were also performed. Intense hydrogenic gas-puffing experiments [13] indicated the presence of an inward (hydrogenic) particle pinch, but suggested strongly that an inward transport of heat at the pinch velocity did not occur. Spectroscopic studies during argon puffing experiments [14] showed strong deviations from local coronal equilibrium, allowing determination of impurity transport coefficients. Impurity transport studies with co- and counter-injection were performed [15]: no difference in impurity transport was found between the two beam directions. However, impurities injected by means of the laser-blow-off (LBO) technique were observed to penetrate the scrape-off layer much more readily with counter- than with co-injection.

2. RF HEATING AND CURRENT DRIVE

The primary emphases of radio-frequency (rf) experiments on PLT are on heating in the ion-cyclotron range of frequencies (ICRF) and on current drive with lower-hybrid waves (LHCD). The experiments were designed not only to maximize plasma parameters, but also to guide the designs of the next generation of devices and prototype reactors.

2.1 ICRF Program

The ICRF heating program has primarily used the fast magnetosonic wave, due to its favorable coupling and penetration characteristics. Wave generation has been accomplished generally with half-turn loops located on the low-field side of the plasma. An important early discovery was that a small concentration of an ion species resonant at its fundamental cyclotron frequency exhibited a strong heating rate when imbedded in a nonresonant background plasma. At 30 kW, the resonant ion population was heated into a highly non-Maxwellian state [16]; the process could be described remarkably well by the use of the quasi-linear Fokker-Planck model. Although the resonant ion distribution was highly energetic, the capability of the PLT for confining fast ions was sufficient to permit the efficient transfer of power from the minority ion species (H) to the background plasma ions (D), with a heating efficiency comparable to neutral injection [17].

A nearly two-fold improvement in heating efficiency for powers up to 1 MW was realized in 1979 with the introduction of a ^3He minority species in a background deuterium plasma [17,18]. Due to its higher charge and mass, the ^3He was more strongly coupled to the bulk deuterons than was the hydrogen minority. Moreover, higher toroidal fields and plasma currents could be utilized, which led to substantial improvement in the fast-ion confinement. Poor ion heating efficiency, high recycling, and enhanced fast-ion loss for currents below 300 kA were clearly shown by use of novel diagnostics developed

to monitor the behavior of the ^3He [19], of the recycling hydrogen [20] and of the fast ions [21] on loss orbits. The continued refinement of spectroscopy and μ -wave scattering diagnostics was also accelerated at this time.

Experiments were initiated in 1979 on rf heating at the second harmonic of the cyclotron frequency ($2\omega_{\text{CH}}$). An energetic ion tail was produced in the majority resonant ion species (see Fig. 4) [18,22-24]; the heating was again found to be consistent with Fokker-Planck theory. The damping rates were sufficiently strong to affect single-pass absorption of the incident energy. The first complete simulations of the rf heating were performed by combining a one-dimensional transport simulation code with Fokker-Planck theory [22]; the results indicated a centrally-peaked rf deposition profile, consistent with geometric focusing of the incident wave power.

Further increases in the rf power, by using multiple antenna systems, led, in 1982, to the establishment of record ion temperatures for rf heating in both the minority and second-harmonic heating regimes [23]. Electron as well as bulk ion heating was observed, and was attributed to collisional coupling with the resonant ions. During this period ICRF heating was combined with neutral injection to produce the highest beta values achieved in PLT (~2%).

Two technical changes proved essential for achieving the maximum power and the best high-power heating efficiency: (1) a new rf-vacuum feedthrough [25], which increased the power handling capability to 1.5 MW per antenna; and (2) a change of source frequency to 30 MHz, which permitted ^3He minority heating at the highest available toroidal field. By these means, the rf power was increased in steps to 5 MW (net power) during 1984 [26,27], resulting in central ion and electron temperatures of 5 keV and 3.5 keV, respectively (see Fig. 5). The rf deposition profiles remained centrally peaked and changed little with applied power. The increase in stored energy was roughly linear with applied power to a level of 110 kJ.

Considerable effort has been applied to understanding the mechanisms and role of impurity generation in ICRF-heated plasmas [28]. Both low- and high-Z impurities are observed to increase with applied rf power [29]: approximately a two-fold increase in the plasma Z_{eff} is observed under the highest power conditions. However, most of the radiated power is due to low-Z radiation from the plasma periphery, and the fraction of the total applied power that is radiated is roughly the same as in the ohmic and neutral-beam-heated cases. The central power balance remains unaffected by the impurity increase; moreover, impurity generation can be controlled by choice of antenna and limiter materials.

Other experiments have confirmed basic aspects of the wave-damping physics for minority heating. The presence of short-wavelength waves (ion-Bernstein waves) localized near the damping layers has been directly confirmed by μ -wave scattering measurements [26]. In recent experiments, direct heating of majority deuterium has been observed, in agreement with the expected transition from minority to majority heating at very low minority concentrations.

2.2 LHCD Program

Noninductive current drive using lower-hybrid waves made an important advance when a low-density target plasma was used, permitting the production of suprathermal electrons. Under these circumstances, the injection of lower-hybrid power via waveguide arrays resulted in the maintenance of a substantial plasma current with no inductive assist [30]. Current-drive efficiency was highest when the fastest waves were excited, allowing wave interaction with electrons of lowest collisionality [31]. A substantial current (up to 450 kA) was driven with the quite modest available power (0.5 MW at 800 MHz). Long-pulse lower-hybrid current drive was demonstrated (see Fig. 6), establishing that steady-state conditions could be maintained. Although the observed current drive efficiency followed the theoretical scaling, $I_p \propto P_{rf}/\bar{n}_e$, the efficiency fell somewhat below the theoretical magnitude. In addition, the current-drive efficiency was observed to decline rapidly above a density of $7 \times 10^{12} \text{ cm}^{-3}$.

In light of these observations, much effort has been spent in understanding the current-drive efficiency and density limit in terms of the underlying wave and particle physics. A detailed study of the rf-induced, nonthermal electron distribution was made using soft X-ray diagnostics, which led to both spatial and velocity information on the current-drive plasma [27,31]. In addition, extensive studies of current ramp-up and current initiation (see Fig. 7) were performed [29,32-34], and have led to the establishment of a current-drive model in the presence of a reverse electric field [35].

Presently, a higher frequency system (2.45 GHz, 1 MW) is being used to investigate the scaling of the density limit. Thus far, current drive has been established in moderately high densities ($\sim 3 \times 10^{13} \text{ cm}^{-3}$).

The basic coupling physics for lower-hybrid waves has been well established for the waveguide-array launchers employed. The dependences of the reflection coefficient on waveguide phase and waveguide size have been measured, and found to be consistent with the theory [30]. Under strong current-drive conditions, an intense, localized hot spot due to the loss of fast electrons was observed on the limiter. A study of this loss process was undertaken, culminating in the installation of a rotating graphite limiter to diffuse the heat flux over a larger limiter surface [36]. The removal of surface heat flux is expected to become more critical as longer pulse lengths and higher powers are approached.

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Fig. 1. Plasma properties of gettered discharges. O: carbon limiter; ●: steel limiter. Z_{eff} = mean ion charge with neoclassical correction to Spitzer resistivity. (From 186080.)

Fig. 2. Ion temperature (H^+) versus time as measured by analysis of charge-exchange neutrals and Doppler broadening of FeXXIV for 2.4-MW D^0 injection into a H^+ plasma. Calculations indicate that the maximum temperature of FeXXIV should exceed that of H^+ by 1700 eV. (796117)

Fig. 3. $T_e(r)$ before injection, during injection with 2 and 4 D^0 beams, and after injection. (806347)

Fig. 4. (a) Hydrogen-ion energy distribution as measured by perpendicular charge-exchange during second harmonic H^+ heating experiments at $P_{rf} \sim 1$ MW, $\bar{n}_e = 3.8 \times 10^{13} \text{ cm}^{-3}$. (b) Time evolution of the average energy, $T_{eff} = 2/3 \langle E \rangle$.

Fig. 5. Time evolution of the central electron (solid line) and deuteron (Δ charge-exchange, $\blacksquare$ neutron emission) temperatures, in a $D-^3\text{He}$ plasma with 4.3 MW of rf power coupled at the ^3He resonant frequency. $\bar{n}_e = 3.7 \times 10^{13} \text{ cm}^{-3}$, $B_T = 3.3 \text{ T}$, and $I_p = 620 \text{ kA}$.

Fig. 6. Plasma current and electron density versus time for 75 kW of coupled lower hybrid power at 800 MHz with a six-element waveguide array.

Fig. 7. Lower-hybrid driven-current start-up with no OH assist. $p_{rf} \sim 200 \text{ kW}$, $\bar{n}_e \sim 5 \times 10^{11} \text{ cm}^{-3}$.

Fig. 3. Profile of X-ray intensity: dots (•) represent peak-intensity and crosses (x) valley-intensity just before disruption; between the points of phase change ($r = 4$ cm, 8.5 cm) the dots become minima, and the crosses become maxima. The top inset gives profiles of T_e , n_e , and q . The bottom inset illustrates the truncated shape expected from a rotating island structure. The lower trace on each oscilloscope picture is a monitor kept at fixed position (looking through the center). (743287)

THE POLOIDAL DIVERTOR EXPERIMENT (PDX)
AND THE PRINCETON BETA EXPERIMENT (PBX)

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ABSTRACT

A review of high-power neutral-beam heating, impurity studies, and high-beta research on the PDX/PBX tokamak, is presented.

1. INTRODUCTION

The Poloidal Divertor Experiment (PDX) [1] operated from late 1978, when it was commissioned, until mid-1983, when the conversion to the Princeton Beta Experiment (PBX) [2,3] began; in early 1984, PBX commenced operations. The outstanding features of the PDX were its 4-null divertor and the 8 MW Princeton Plasma Physics Laboratory-Oak Ridge National Laboratory (PPPL-ORNL) neutral-beam injection (NBI) system, then the most advanced in the world. All four neutral beams were initially oriented to be co-perpendicular ($R_{\text{tangency}} = 38$ cm); as part of the PBX modification, two of the beams were reoriented to co-parallel, with $R_{\text{tangency}} = 130$ cm. Meanwhile, in late 1982 the 4-null divertor was changed to 2-null, and at the same time the channels connecting the main plasma chamber to the divertor chambers were closed off as much as possible. The PBX magnetic topology, which has as its main feature a plasma cross section strongly indented on the inboard side, allows the separatrix to fall inside the limiters, but there are no special baffles to create a distinguishable divertor chamber.

The PDX research program addressed heating and confinement of neutral-beam (NB) heated plasmas, beta limits, fueling and impurity control, and -- just before the 1983 termination -- electron cyclotron heating (ECH) [4]; in contrast, the focus of PBX has been entirely on the beta issue. In this paper we review the heating and confinement results (Section 2) and impurity studies (Section 3) from PDX, and the beta results from both PDX and PBX (Section 4).

2. NB HEATING AND CONFINEMENT STUDIES ON PDX

Initial results from NB heating carried out at low power in 1980 were encouraging [1], but at full power the ion heating efficiency was quickly seen to be significantly worse than in the Princeton Large Torus (PLT). Concern

that there might be an unforeseen problem with the near-perpendicular injection angle of the PDX beams (as indeed was to be the case at high beta -- Cf. Section 4) led to a detailed study of confinement dependence on various plasma parameters, with emphasis on overlapping the parameter range explored by PLT. These discharges were therefore run with graphite limiters rather than with the divertor. The conclusions were as follows [5]:

- (1) Ion temperatures improved with current. At 500 kA, typical of PLT discharges in 1978 and 1980, ion heating in PDX was equivalent to that seen with tangential injection in PLT, although overall T_e was 10-20% lower. Temperatures of ~6 keV were achieved at densities of $3-4 \times 10^{13} \text{ cm}^{-3}$.
- (2) The dependence of heating on current was broadly analogous to the "soft β_p " limit reported for the Impurity Studies Experiment (ISX) [6], which could also be interpreted as a dependence of confinement on plasma current. However, in PDX the magnetically determined β_p increased linearly with power, though at a rate generally below the ohmic case.
- (3) In agreement with ISX data, the dependence of the confinement on the toroidal field strength was found to be very weak; the density dependence -- reported as less than linear in ISX -- was even weaker in PDX.
- (4) From the different confinement scalings for ohmic- and beam-heated plasmas, it followed that at high current and low density there was little or no confinement degradation with neutral-beam heating, whereas at low current and high density a large drop in T_e could be expected to occur when the beams were turned on.

The results of this work were twofold: it firmly established the "low-mode" confinement properties of beam-heated plasmas, and it showed that -- at low beta -- beam orientation had at most a minor effect on confinement. "High (H)-mode" studies carried out in 1983 with the closed divertor configuration did not modify the basic scaling but confirmed the Axially Symmetric Divertor Experiment (ASDEX) finding that improvements in confinement were nevertheless possible. The PDX contribution was to demonstrate the strong adverse dependence of T_e on the pressure of neutral gas in the main plasma chamber, and the favorable effect of localized recycling in the divertor throat [7]. The work also quantified the existence of very steep gradients in n_e and T_e just inside the separatrix, and showed that the improvement in global confinement resulted not only from better thermal insulation at the edge but also from a drop in electron thermal diffusivity over the whole cross section.

3. IMPURITY STUDIES IN PDX

The PDX was operated extensively with both open and closed divertor configurations, rail and scoop limiters [8], and a toroidal bumper limiter (see Fig. 1). Early experiments with ohmically heated plasmas [9] showed the divertor to be effective in producing very clean plasmas by reducing the amount of material from the limiting surface that reached the main plasma. With the advent of high-power injection, impurity characteristics of a wide variety of plasmas with NBI were studied [10].

Neutral-beam power levels of 6 MW and pulse lengths exceeding 200 ms were satisfactorily accommodated by either divertor or limiter configurations, with Z_{eff} typically rising when the neutral beams came on and then dropping as the impurities were diluted by increasing density. At $P_{\text{inj}} \geq 5$ MW, most data showed $Z_{\text{eff}} < 2$ with occasional values up to about 3. However, limiter plasmas showed a monotonic increase of impurity density with P_{inj} , while the divertor discharges showed little change and had values at least two times lower than the limiter cases at high P_{inj} . Metallic impurity concentrations were usually negligible, $< 10^{-3}$, with the exception of the closed-divertor cases which showed substantial iron contamination.

Radiative losses, as determined from bolometric measurements, typically accounted for less than 10% of the power flow from the plasma core (some closed-divertor and counter-injection cases excepted), and independently of power input contributed about 40% to the total energy loss rate. Power flow to the divertor neutralizer plates was about 50% of input power for the open-divertor case, reducing to less than 10% for the closed divertor. Thus, the power accountability was reasonably good in the first case, whereas, in the second case about 50% was unaccounted for; it is unknown how much of this power may have been deposited in the uninstrumented divertor throat region.

4. BETA STUDIES ON PDX AND PBX

4.1 PDX

Systematic beta studies carried out in 1982 demonstrated a saturation of beta with beam power that could be expressed as: $\beta q \leq 0.06$ [11]. However, the beta-dependent energy loss was found to be incurred primarily by precessing beam ions in resonant interaction with an internal $m/n = 1/1$ mode, rather than by transport processes within the bulk plasma (see Fig. 2). Thus, the observed saturation was due to a progressive loss of effective heating power. Detailed experiments [12] and subsequent theoretical modeling [13] provided a coherent picture of the relevant instability — termed a "fishbone" because of the characteristic shape of the observed magnetic fluctuations.

The observed beta limit can be expressed in Troyon's "canonical" form [14], $\beta (\%) \leq c \mu_0 I / aB$; thus defined, $c \approx 2.5-3.0$, the same value found in other tokamaks [15] that do not have the particular form of fast-ion loss observed in PDX.

4.2 PBX

The PBX was designed to explore, in a preliminary way, the theoretical prediction that a strongly indented plasma (see Fig. 3) would allow stable access to the second stable region for ballooning modes [16], while also stabilizing internal kinks [17]. The design included plates connected saddle-coil fashion to stabilize the $n=0$ axisymmetric instability, but left the outboard region of the plasma unencumbered to allow maximum flexibility in cross sectional shape. The main results of the experiments to date are the following [3]:

- (1) Stable, low-beta plasmas have been formed with indentation exceeding 20%. To achieve the theoretically desirable 30% will require either more sophisticated control of the shaping field or a broader current profile.

- (2) The axisymmetric vertical instability has generally been controllable, but may be entering in the late stages of disruptive plasma decay.
- (3) Indentation significantly raises the mean current-density that the plasma can carry: in many cases, a cylindrical plasma with the same current and equal cross-sectional area would have $q < 1$.
- (4) The "Troyon" limit continues to be a good predictor for the maximum obtainable beta in PBX, particularly if the mean radius $\langle a \rangle$ (the radius of a circle having the same area as the indented plasma) is employed (Fig. 4). Because q_{cy} is low and $I/\langle a \rangle$ correspondingly high, it has been possible to reach $\beta \geq 5\%$.
- (5) The beta limit is evinced by a hard disruption, rather than "fishbone"-like fast-ion losses as in PDX. Nevertheless, a strong $m/n = 1/1$ mode is often, but by no means always, evident before the disruption. Fishbones do occur, but with such low frequency and intensity as to have negligible effect on fast ion confinement.
- (6) A significant effect of indentation on low-beta confinement has not been observed. However, when operated with a "free" separatrix (X-points inside the limiters), typical "L-mode/H-mode" transitions are seen in the H_α light and in confinement. During the quiescent "H-mode" phase, nickel (with which the passive stabilizers are plated) shows strong peaking in the plasma core and degrades the stored energy content through radiation losses.

The current plan is to improve the stability of the plasma to $n=1$ modes at high beta by changing the passive stabilizer configuration to include an outboard conducting shell. Higher moments added to the magnetic shaping field will ensure that the plasma shape conforms to the shell, and that a 30% indentation can be maintained for a wide variety of plasma current distributions. These modifications will be completed in early 1987.

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Fig. 1. PDX configurations: (a) top and bottom graphite rails; (b) inside toroidal limiter; (c) closed divertor with asymmetric null, for H-mode operation; (d) standard open divertor, with plasma attached to inner divertors.

Fig. 2. Magnetohydrodynamic (MHD) effects on fast ions in PDX. The precession of ions injected by the neutral beam drives a synchronous MHD kink mode, with coupled internal (soft X-ray signals) and external (magnetic loops) components. In this process the energetic ions escape (as shown by the charge-exchange signal) and the neutron emission drops. The time scale is shown expanded on the right side of the figure. (A2X2203)

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Fig. 3. Cross section of PBX showing the midplane pusher coil with graphite armor, passive stabilizer plates with graphite protection, and other coils providing the shaping field.

Fig. 4. Volume-averaged beta in percent for (a) PDX and (b) PSX vs $\beta_c = \mu_0 I_p / \langle a \rangle B_T$, where $\langle a \rangle$ is the volume-averaged minor radius.

S-1 SPHEROMAK

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ABSTRACT

The S-1 device inductively generates spheromak plasmas with major radii of 40-65 cm and toroidal plasma currents up to 500 kA. The major objective is the investigation of MHD stability and transport characteristics of spheromaks that are stabilized in externally applied magnetic fields by loose-fitting conductors and coils.

1. INTRODUCTION

The S-1 device began experimental operation at PPPL in early 1983. The spheromak is a magnetic confinement configuration of the compact-toroid type, with an internal toroidal field and a nonlinked plasma-coil topology [1,2]. These features are expected to ease reactor engineering requirements and add potential flexibility by permitting translation and compression of the plasma [3]. The S-1 experiments are designed to investigate spheromak plasmas produced by an electrodeless inductive technique that will lend itself to reactor-sized scale-up. The formation technique, was pretested in the Proto S-1 devices [4], and has now been verified in S-1. A figure-8-coil stabilization system has been found to make a dramatic improvement in the gross stability of the spheromak plasma [5]. The study of transport in S-1 plasmas has just begun, with initial measurements of T_e scaling as a function of total plasma current.

2. S-1 DEVICE

As is shown in Fig. 1, the major components of the S-1 device [6] are the flux core, vacuum vessel, and equilibrium field coils. The flux core has a major radius of 1 m and a minor radius of 0.19 m. It contains poloidal- and toroidal-field windings as well as an equilibrium-field winding. The inconel cover of the flux core, which prevents outgassing, must be thin enough to allow the flux to penetrate in a short time, creating a challenging fabrication problem. The first flux-core liner for S-1 had an average thickness of 0.25 mm. Operating vacuum pressures of 5×10^{-8} to 2×10^{-7} Torr are regularly obtained. The quasi-steady equilibrium field (EF) is produced by a set of coils mounted on the vacuum vessel domes and by two turns located inside the flux core. These coils can be connected in different combinations

to yield various equilibrium-field shapes. The EF coils are powered by two large dc generators.

3. FORMATION OF SPHEROMAK WITH FLUX CONVERSION

The plasma formation is based on inductive transfer of toroidal and poloidal magnetic flux from the flux core into a surrounding plasma [7,4]. If the two coil currents are timed so that the formed plasma approaches the Taylor minimum-energy state, a spheromak configuration is created (major radius = 40-65 cm).

Relaxation to the Taylor state has been observed to be an important process for spheromak formation in S-1. This relaxation consists of an anomalous conversion of poloidal-to-toroidal flux during and after the formation of the plasma [8]. The spheromak plasma seeks a Taylor state in such a way that the "pinch parameter" (the ratio of poloidal flux to toroidal flux) is a constant, independent of initial conditions. Magnetohydrodynamic fluctuations with toroidal mode numbers $n = 4, 3, 2$ were observed sequentially during the flux-conversion period. The observations are directly correlated with the time evolution of the $q(r)$ profile [9], with $q(0)$ rising through the integral fractions $1/4, 1/3, 1/2$. Shown in Fig. 2 are the time evolutions of the central q value, of the poloidal flux function ψ_{MAX} at the magnetic axis, and the toroidal flux Φ within the spheromak plasma. All values were deduced from internal probe measurements. Coincident with a precipitous drop of ψ_{MAX} and a sharp rise of Φ , one observes that the central q value approaches the value of the Taylor state for an oblate spheromak.

4. ATTAINMENT OF STABLE SPHEROMAK WITH FIGURE-8 COILS

In the absence of a passive stabilizer, the spheromak plasma is unstable to at least one of three rigid-body modes in any type of EF field: the tilt mode and the horizontal and vertical shift modes. In order to stabilize these gross MHD modes, figure-8 coils (named according to their eddy current pattern) [10] were installed on each side of the midplane of the flux core, with the cross points of the coil bars 60 cm ($\approx 1.5 a_{plasma}$) away from the midplane. The figure-8 system was used to maintain the spheromak plasma in a stable equilibrium position for more than a millisecond -- an order-of-magnitude improvement in the duration of the stable quiescent period [5]. With the stabilizing coils, the spheromak discharges became so reproducible that the time evolution of the global magnetic field structure could be mapped out, based on internal magnetic probe data of 40 consecutive discharges. A cross section of the plasma in a chosen toroidal plane was traced by scanning a radial arm probe in the z direction; grid sizes of sampling were 7.5 to 15 cm. The time evolution of poloidal and toroidal flux plots taken 0.2, 0.3, 0.4, 0.5 ms after the start of the discharge (Fig. 3) shows a good match between the poloidal flux surfaces and the constant- RB_t surfaces.

With the figure-8 coil system, the plasma was kept grossly MHD-stable only when a fraction (typically more than 10%) of the poloidal flux remained linked to the flux core. Recently, a pair of small funnel-shaped stabilizers placed on the major axis just off the midplane have been used to keep spheromaks MHD-stable with their separatrices completely detached from the flux core.

5. INVESTIGATION OF SPHEROMAK PROPERTIES

As a result of recent improvements in vacuum conditions and plasma stability, the S-1 plasma parameters have advanced notably [11]. Typical experiments have been carried out with toroidal plasma currents up to 350 kA, creating internal magnetic fields up to 3 kG. The configuration lifetime has exceeded 1 ms, with a magnetic field decay time $\tau_B = B (dB/dt)^{-1}$ ranging from 0.5 ms to 2 ms. The major and minor radii of the plasma have been 40-65 cm and 25-45 cm, respectively. The electron temperatures ranged from 30 eV to 110 eV. The volume-average beta was 5-10%. The electron density ramped up to peak values of 0.6 to $2 \times 10^{14} \text{ cm}^{-3}$ during the formation phase and decayed smoothly, with a time constant of over 1 ms. Figure 4 shows measured values of T_e versus the capacitor-bank voltage of the poloidal- and toroidal-field power supplies. The plasma current is a linear function of the bank voltage. Within the large shot-to-shot scatter, the spheromak data seem to agree roughly with the Reverse-Field Pinch (RFP) scaling of $\beta = \text{const}$. The data also indicate that the electron temperature is not limited by impurity radiation, although radiation cooling still plays a significant role in most discharges. Future research plans include the placement of a current transformer along part of the major axis of the S-1 device, designed to increase the toroidal plasma current to ~1 MA and correspondingly enhance the internal toroidal flux of the spheromak plasma by means of the Taylor mechanism.

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Fig. 1. Schematic of S-1 device with figure-8 coils.

Fig. 2. Time evolution of (a) central q value, (b) $\psi_{\max}$ value at the magnetic axis, and (c) toroidal flux Φ within the spheromak plasma.

Fig. 3. Time evolution of poloidal magnetic flux plots obtained by internal magnetic probe data from 40 consecutive reproducible shots. The probe is scanned through $-52.5 \text{ cm} < z < 52.5 \text{ cm}$, and $-77.5 \text{ cm} < R < 77.5 \text{ cm}$. The separatrices of the spheromak configuration are shown by the outermost poloidal flux lines around the plasmas.

Fig. 4. Electron temperature measured in the S-1 device versus bank voltage ($I_p \propto V_{\text{Bank}}$). T_e increases in proportion to $I^{1+\alpha}$ with $\alpha \approx 0.5 - 1$. $\bar{n}_e \approx \text{const.}$

INITIAL STUDIES OF CONFINEMENT, ADIABATIC COMPRESSION,
AND NEUTRAL-BEAM HEATING IN TFTR

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ABSTRACT

The Tokamak Fusion Test Reactor (TFTR) is intended to achieve approximate energy breakeven in D-T plasmas. Construction approval was received in March 1976, and the first plasma was produced in December 1982. Three major experimental run periods, the last ending April 1985, have yielded experimental results on confinement and heating that extend the scaling laws of smaller machines. The plasma parameters in TFTR are now approaching those required for breakeven in unthermalized, two-component regimes. They span the range from high-density operation with $n_e(0)\tau_E \sim 4 \times 10^{19} \text{ m}^{-3} \text{ s}$ to a low-density regime with $T_i(0) \sim (9 \pm 2) \text{ keV}$. The maximum product $n_e(0)\tau_E T_i(0)$ of about $9 \times 10^{19} \text{ m}^{-3} \text{ s keV}$ was obtained at a central density, $1.1 \times 10^{20} \text{ m}^{-3}$, by means of pellet injection.

1.0 INTRODUCTION

The formal TFTR Project Objectives, as defined by the United States Department of Energy in the TFTR Project Management Plan (1976), are that TFTR plasmas should enter the temperature range 50-100 million °C with an $n\tau_E$ -value of at least $10^{19} \text{ m}^{-3} \text{ s}$ and make 1-10 megajoules of D-T fusion reactions per pulse.

The machine parameters are given in Table I. At the present time, TFTR has operated the toroidal field at the design value of 5.2 T at 2.48 m, while running design-rated plasma currents of 2.5 MA. The neutral-beam power has reached 20% of the design level. After the addition of the two remaining beamlines in early summer 1985, full-power operation is expected in mid-1986. The TFTR diagnostic system described by Johnson and Young [1] is in nearly full operation.

2.0 CONDITIONING AND IMPURITIES

The initial vacuum vessel conditioning process described by Dylla *et al.*, [2] involved: glow-discharge cleaning, pulse-discharge cleaning, high-power pulse operation, and chromium gettering. After a typical vacuum opening,

about 20 hours of glow-discharge cleaning in hydrogen was followed by some 50,000 pulse-discharge cleaning shots. During operation, high-power pulses of ~1-2 MA contributed to a long-term cleanup process, as evidenced by the gradual decrease of the effective ionic charge (Z_{eff}). Also, the ratio of radiated power to input power decreased from 0.9 to 0.4 after about 2000 pulses. Chromium was evaporated at two toroidal locations, covering about 70% of the surface area with a 0.1 μm -thick film, and initially decreased the oxygen content by a factor of about 2. The overall impurity level in TFTR was reduced to $Z_{\text{eff}} \sim 1.4$ as the density, $\bar{n}_e$, was increased to about $4.6 \times 10^{19} \text{ m}^{-3}$. At low densities, metals contributed significantly to Z_{eff} , while at high densities the impurities were predominately carbon and oxygen.

3.0 OPERATING RANGE

A typical TFTR discharge has a plasma current pulse length of 6 s with a flat-top of ~1.5 s. Feedback circuits control the major radius vertical position, plasma current, and plasma density, thus providing specified reproducible discharges for physics studies. Plasmas were easily started with ohmic-heating system voltages as low as 9 V per turn. The minor radius was programmed to increase with plasma current, so as to minimize MHD disturbances and radiated power during start-up. Current penetration during ramp-up appeared to be essentially classical after the initial start-up phase of ~0.1 s.

The TFTR was operated at currents up to 2.5 MA, with $q_0 \sim 2.4$. The plasma regimes summarized in the Hugill diagram (Fig. 1) indicate maximum Murakami parameters of about $3.3 \times 10^{19} \text{ m}^{-2} \text{ T}^{-1}$ with gas puffing of ohmic heated and neutral-beam heated deuterium plasmas. Helium plasmas with $\bar{n}_e \sim 3 \times 10^{19} \text{ m}^{-3}$ and Murakami parameters of $5 \times 10^{19} \text{ m}^{-2} \text{ T}^{-1}$ have also been produced with gas puffing. As the high-density limit was approached, oxygen radiation was found to increase and poloidally localized radiation (MARFES) appeared on the small-major-radius side of the plasma. The ORNL/TFTR pellet injector was used to increase the density in deuterium to $\bar{n}_e \sim 9 \times 10^{19} \text{ m}^{-3}$, with a central density of about $1.7 \times 10^{20} \text{ m}^{-3}$. Low-density plasmas could also be formed at currents of ~800 kA, with $\bar{n}_e \sim 7 \times 10^{18} \text{ m}^{-3}$ and with few runaways. Disruptions at the high-density and low- q boundaries of the operating range were characterized by a minimum current-decay time of ~8 ms, which corresponds to a maximum decay rate of ~0.4 MA ms⁻¹. The design value is 1 MA ms⁻¹.

4.0 OHMIC-HEATING CONFINEMENT STUDIES

Systematic scans of plasmas with $a = 0.83 \text{ m}$ were carried out as a function of $\bar{n}_e$, I_p , and B_t as reported by Hawryluk *et al.*, [3] and Efthimion *et al.*, [4]. During ohmic heating, the central electron temperature ranged up to 5 keV, and its variation can be approximated by $T_e(0) \sim B_t^{0.8} Z_{\text{eff}}^{0.5} a^{0.7} I_p^{-0.2} R^{-0.2}$. The energy confinement time in deuterium plasmas, defined as $\tau_E = (W_i + W_e) / [P_{\text{in}} - d(W_i + W_e)/dt]$, shows a linear rise with density up to ~0.4 s. When the toroidal field, B_t , and the plasma current, I_p , were varied independently, the confinement time varied as $\tau_E \sim B_t^x / I_p^y$ where x and y are in the range 0.8 to 1.1. Consequently, the ohmic-heating confinement of large TFTR plasmas can be approximated by $\tau_E \sim \bar{n}_e q$, for confinement times up to ~0.4 s. Size-scaling was investigated using a movable limiter that could be used to vary the minor radius from $a = 0.4 \text{ m}$ to 0.83 m . Typically, the

scaling of τ_E with minor radius was $\tau_E \sim a^y$ with $0.5 < y < 1.5$. Major-radius scaling experiments for $a = 0.55$ m plasmas compared operation at large R using the graphite movable limiter with operation at small R using an incoel limiter. The impurity content and radiated power varied significantly during the major radius scans, thereby, preventing the determination of a scaling law for $\tau_E(R)$ from only TFTR data. Comparison of ohmic-heating confinement times in TFTR and PLT suggests an overall size-cubed scaling of τ_E [5]. The observed scaling is consistent, within errors, with the general constraints of Connor-Taylor [6] and with the empirical neo-Alcator result [7] $\tau \sim \bar{n}_e a R^2 q$, as shown in Fig. 2. A scaling of this type was first proposed by Pfeifer-Waltz [8] and is also similar to the size scaling observed on T-11 [9]. At low densities, the confinement is dominated by electron conduction for all plasma sizes. Small minor radius, high-density plasmas, on the other hand, exhibit a saturation of τ_E with density, and they are found to be dominated by ion transport losses with values close to the Chang-Hinton neoclassical prediction.

5.0 INITIAL EXPERIMENTS ON PELLETT INJECTION INTO OHMICALLY-HEATED PLASMAS

The ORNL/TFTR pellet injector, as described by Combs et al., [10], was used to inject single 4-mm pellets into TFTR. Central densities up to $\sim 1.6 \times 10^{19} \text{ m}^{-3}$ (Fig. 3) were produced, with subsequent central-density decay times of ~ 2 s. The pellet injector was then used to inject up to five 2.7-mm pellets, at a rate up to 5 s^{-1} , into a single discharge, so as to produce better-controlled increases in plasma density. Pellets extended the line-average density range by about 20% and doubled the central density over the highest values achieved by gas puffing. However, there was no significant increase in global energy confinement, possibly because the TFTR gas puffing confinement results are already near the projections of the best results from smaller machines. Pellet injection into cmic plasmas produced a maximum $n_e(0)\tau_E \sim 4 \times 10^{19} \text{ m}^{-3} \text{ s}$ with $T_i(0) \sim 2.1 \text{ eV}$.

6.0 CONFINEMENT STUDIES WITH NEUTRAL-BEAM HEATING

Two co-tangential neutral-beam lines with power levels up to ~ 6 MW were used to investigate confinement scaling with P_b/P_{OH} up to about 5. Initial results have been reported by Eubank et al., [11] and Murakami et al., [12]. In the standard density regime ($\bar{n}_e \sim 3\text{--}5 \times 10^{19} \text{ m}^{-3}$), Z_{eff} during the initial ohmic phase was typically about 2-3 and increased by about 0.5 during beam injection. The total heating power in TFTR can be accounted for within $\pm 10\%$ in this regime by measuring the total radiated power and the power to the limiter. The central ion temperature typically increased from ~ 2 keV to ~ 4 keV, which corresponds to $\Delta T_i (\text{keV}) = 4 P_b (\text{MW}) / R (\text{m}) \bar{n}_e (10^{19} \text{ m}^{-3})$. This formula is also in agreement with the PLT ion heating results, which suggests that the gross ion energy confinement is proportional to a^2 . As in smaller tokamaks, the electron-heating efficiency is somewhat lower.

The confinement time during neutral-beam heating is defined as $\tau_E = (W_i + W_e) / [(P_{OH} + P_{bi} + P_{be} - d(W_i + W_e)/dt)]$. For neutral-beam powers of ~ 3 MW, the confinement time is independent of density in the range $2\text{--}5 \times 10^{19} \text{ m}^{-3}$. At a constant heating power of ~ 4 MW, the confinement time increased linearly with plasma current, reaching a maximum of ~ 0.73 s with 2.2 MA (Fig. 4). On the other hand, the confinement time decreased as a function of neutral-beam

power, as indicated in Fig. 5. These parametric variations are roughly consistent with the L-mode scaling projections derived by Goldston [13] from smaller tokamaks.

By operating at low currents, $I_p \sim 0.8-1.0$ MA, the neutral-beam experiments were extended to a very low-density regime where $\bar{n}_e \sim 10^{19} \text{ m}^{-3}$. As shown in Fig. 6, the ion-temperature measurements from the Doppler-broadening of the X-ray lines of titanium that are emitted from the plasma center and from the perpendicular line-averaged passive hydrogen charge-exchange indicate "raw" ion temperatures of about 12 and 6 keV, respectively. The corresponding central electron temperature is ~ 4 keV.

The Doppler shift of the titanium line suggests toroidal rotation speeds up to $7 \times 10^5 \text{ m s}^{-1}$ and linear scaling with $\bar{v}_b/\bar{n}_e$ which is supported by measurements on the $m=1/n=1$ MHD mode. However, the MHD measurements have only been made for velocities up to $3 \times 10^5 \text{ m s}^{-1}$. These very high rotation speeds introduce new phenomena that may affect the interpretation of plasma temperatures and transport. Preliminary analyses of this very low-density regime suggest that energetic injected ions constitute about 50% of the central ion density, and that the average central ion energy is ~ 24 keV. The thermalized deuterium and proton temperatures in the central region are estimated to be in the range (9 ± 2) keV, for consistency with the raw temperature measurements. However, there are substantial uncertainties in the interpretation of the raw temperatures in these very low-density, rapidly-rotating plasmas.

Some important features of this new regime are that the central impurity content is relatively low, with $Z_{\text{eff}}(0) \sim 2$, and that the $Z_{\text{eff}}(r)$ profile inferred from the visible bremsstrahlung is hollow. Future experiments will determine if this is due to the strong central beam fueling or to the associated momentum input. In this regime, beam-driven current effects are clearly seen. As the plasma current is varied from 1.0 MA to about 0.4 MA, the residual loop voltage during beam injection decreases from 0.4 V to 0.05 V, suggesting a beam-driven current of ~ 0.4 MA, which is close to the theoretical predictions. The observed energetic-ion regime resembles the two-component mode of operation for which TFTR was designed and promises to be useful in enhancing D-T performance for breakeven experiments.

7.0 COMPRESSION EXPERIMENTS

Initial major-radius compression experiments have been reported by Tait et al., [14]. More recently, neutral-beam heated plasmas with $I_p \sim 450$ kA, $a_i \sim 0.57$ m, $R_i \sim 3.0$ m were compressed in about 20 ms to $I_p \sim 620$ kA, $a_f \sim 0.49$ m, and $R_f = 2.17$ m. In these experiments, the ion temperature increase is adiabatic within about 10% (ideally, $T_i \propto C^{4/3}$, where C is the compression ratio, R_i/R_f). The toroidal mass-rotation velocity increases roughly as C , in accordance with theoretical prediction. Figure 7 shows the charge-exchange spectrum for beam injection at 80 keV, followed by compression with $C = 1.38$. The beam ions are accelerated to the predicted energy of ~ 160 keV. The central electron temperature and density, however, are about 20% below their projected increases of $C^{4/3}$ and C^2 , respectively. During compression, the normal recycling at the limiter is suppressed and the plasma energy diffuses out rapidly. In free-expansion (partial compression) experiments this leads to a reversal of the usual relationship between the relative widths

of the electron-density and temperature profiles. Initial measurements of the magnetic fluctuations with Mirnov loops indicate the occurrence of MHD bursts as the plasma is compressed beyond the center of the vacuum vessel. These signals resemble the periodic MHD bursts that are usually seen in tokamaks as the limiter q value goes through integral values during the initial plasma-current rise and may be caused by a similar skin-relaxation phenomenon which occurs in the "vacuum region" surrounding the compressed plasma.

8.0 SUMMARY

During the initial operation of TFTR, plasma-confinement studies have been extended into regimes of reactor-plasma interest, and several new phenomena of basic plasma physics interest have been discovered.

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Table I. TFTR Parameters.

Major Radius	2.5 m
Minor Radius	0.82 m
Toroidal Field	5.2 T
Plasma Current	2.5 MA
Equilibrium Field	0.5 T
Ohmic Heating Flux	15 Vs
Compression Capability, $C = 1.4$, $\tau_{\text{comp}} = 30$ ms	

<u>Neutral-Beam Capability:</u>	<u>April 85</u>	<u>December 86</u>
Pulse Length	0.5 s	2.0 s
Voltage	80 kV	120 kV
Number of beamlines	2	4
Beamline Configuration	2-Co	3-Co, 1-Ctr
Total Power (D)	6.3 MW	27 MW
Power at Full Energy	~3 MW	20 MW

Initial Pellet Injection:

Pellet Size	(2.7 or 4)-mm diameter, H or D
Pellet Speed	1.9×10^3 m per s (H)
Multiple Rate	5 s^{-1}
Average Density Increase for 2.7-mm Pellet	$2 \times 10^{19} \text{ m}^{-3}$

Fig. 1. Hugill diagram for large plasmas ($R = 2.6$ m, $a = 0.8$ m) run on the movable limiter in TFTR. (A) Region accessible during the plasma current flat-top of ohmically-heated, deuterium plasmas. The left-hand boundary represents the recycling limit, the right-hand boundary the density limit reached in scans at fixed q_a . Neutral-beam injection has not yet extended the density limit beyond this region for the large plasmas. (B) Trajectory of an ohmically-heated discharge with injection of a single, large deuterium pellet (~ 4 -mm diameter) during the current flat-top. (C) Boundary of the region which has been accessed with injection of multiple (up to 3) small deuterium pellets (~ 2.7 -mm diameter) into ohmically and neutral-beam heated plasmas.

Fig. 2. The global energy confinement time for chemically-heated deuterium plasmas.

TFTR PELLET INJECTION (d=4mm)

Fig. 3. Radial profiles of electron density and temperature just before and 50 ms after the injection of a 4-mm diameter deuterium pellet into an ohmically-heated TFTR discharge.

Fig. 4. The global energy confinement time of neutral-beam heated plasmas in TFTR as a function of the plasma current.

Fig. 5. The global energy confinement time of neutral-beam heated plasmas as a function of the neutral-beam power absorbed by the plasma.

Fig. 6. Raw ion temperatures from Doppler broadening of impurity lines (T_{imp}) and perpendicular charge-exchange [$T_i(cx)$] in neutral-beam heated low-density plasmas in TFTR. Analysis of these data indicates a central ion temperature of (9.2) keV.

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Fig. 7. Comparison of charge-exchange spectra (solid lines) with Fokker-Planck code simulation (dotted lines). The sight-line of the analyzer has a tangential radius of 2.17 m.

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