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PDX EXPERIMENTAL RESULTS IN FY82

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ABSTRACT

This report presents a detailed summary of the major experimental results of PDX in FY82 and represents the efforts of the entire PDX group. Topics covered include β -scaling and fishbone studies, fluctuations, disruptions, impurities and impurity transport, power handling, limiter conditioning, edge studies, plasma fueling, counter-injection, and diagnostic development. A less detailed version will appear as the FY82 PDX contribution to the PPPL Annual Report.

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I. INTRODUCTION

The PDX experimental program in FY82 was focused mainly on achieving high- β plasmas, and on gaining a coherent understanding of the energy transport and MHD activity of these plasmas. Most of this work was done with the plasma boundary defined by poloidal rail limiters, but some was carried out using the inner wall as a toroidal limiter, and some in the Inside D divertor configuration (Fig. 1). Experiments were also carried out to analyze the effects of pellet fueling, along with studies of energy and particle flow, and impurity content and transport. Diagnostic techniques were developed in connection with impurity studies. Results of these experiments were presented at the Ninth International Conference on Plasma Physics and Controlled Nuclear Fusion Research, the Third Joint Varenna-Grenoble International Symposium on Heating in Toroidal Plasmas, the Fifth International Conference on Plasma Surface Interactions in Controlled Fusion Devices, and the Ninth Annual Conference on Plasma Physics (Oxford).¹⁻⁵

II. MAJOR ACTIVITIES

During the month of October 1981, experiments were performed in the areas of low- q operation, scattering of CO_2 laser radiation, plasma rotation measurements, and plasma fueling. The machine was then opened from November to mid-December 1981 for routine divertor maintenance, replacing the graphite blades on the limiters, installing new thermocouples, and repairing and upgrading machine diagnostics.

In mid-January, after two weeks of conditioning, the machine had to be opened again to repair a break in an OR bus lead. Concern about the mechanical support of this lead, even after the break had been repaired, led to a decision to operate the machine at reduced field and current levels until

a more complete repair could be implemented. Fortunately, since low- q and high- β operations were planned for FY82, this restriction in operating range did not compromise the experimental objectives for the year.

Low- q operations began in earnest in March, initially on the inner wall. A few good discharges with $q \sim 2.5$ were obtained, but the success rate was low, and in late March operations switched to the poloidal graphite rail limiters. Because of high MHD activity leading to fast disruptions at low- q values, an operating method to reduce the number of disruptions was developed. The technique was an automatic feedback and abort system in which both gas flow and current were increased if the MHD activity, as measured by the Mirnov coils, exceeded a critical value. This usually controlled the level of MHD. If, however, the activity became too large, the discharge was terminated by shutting off the current in the OH primary. Throughout FY82, PDX operation was aided by this feedback system as well as fully automated feedback systems controlling plasma current, density and vertical,⁶ and radial position.

By the end of April 1982, PDX was operated at toroidal fields as low as 0.9 T and $q_a \sim 1.7$, and achieved values of $\langle \beta_{\perp} \rangle$ above 3%. Studies of β -scaling continued through the months of May and July 1982 with toroidal fields down to 0.7 T. To aid in the breakdown of these low toroidal field plasmas, the discharges were run with a small, negative Divertor Field, greatly improving the radial stability of the plasma. Some of the β -scaling experiments were repeated during the several weeks of divertor operation that were carried out in June-July 1982.

In August 1982 plasma operation was stopped in order to upgrade the divertor hardware, including the replacement of the titanium neutralizer plates and liners with stainless steel components, blocking the outer divertor

channels, and taking the outer divertor coils out of the DF circuit for operation in a double null, divertor geometry. For the planned electron cyclotron heating (ECH) system, two inside launchers and a window for the outside launcher with an associated polarizing reflector on the inside wall were installed. In addition, a scoop limiter and vertical soft X-ray array were installed. The interlead void of the damaged OH bus lead was packed with an epoxy/glass mixture. Other repairs included replacing the Thomson scattering system shutters, work on the magnetic pickups and plasma position loops, and scraping off titanium flakes. Plasma operation resumed in November 1982.

During most of FY82, PDX ran on a nine-hour operating shift for six days/week on alternate weeks. The total period of operation in FY82 was eighteen weeks. During this time, approximately 5500 discharges were made, 40% of them with neutral beam injection (NBI). About half of both the total and NBI discharges were high power; $I_p > 200$ kA for > 0.5 sec. The shot efficiency (# shots/time of operation) in FY82 was 50% greater than that in FY81. The milestones for FY82 are given in Table I.

The four duopigatron neutral beam injection systems of the joint Princeton Plasma Physics Laboratory - Oak Ridge National Laboratory Heating Project provided up to 7 MW of D⁺ power and 5 MW of H⁺ power for pulse lengths of up to 300 msec. The four beam lines inject at a tangency radius of 35 cm, which gives an angle of 14° from perpendicular at the center of the vessel. The beam injection systems are routinely operated under automatic computer control providing both reliability and reproducibility of beam line operations (Figs. 2a,b). Beamline calorimetry with direct computer processing of temperature sensor waveforms permits an accurate energy calibration within 17 msec after conditioning shots, which occur at a 60 sec repetition rate between the 5 min PDX shot repetition rate.

To provide quick, on-line analysis of the PDX experiments, a between-shots confinement analysis code (CAAN) was developed. CAAN is an O-D code that relies primarily on a magnetic equilibrium analysis providing a time history of such quantities as plasma current, surface voltage, line averaged density, $I_1/2 + \beta_0$, β_0 and β_T (using an assumed q-dependent plasma current profile), injected and absorbed beam power, q, and a global confinement time which was generally in good agreement ($\pm 15\%$) with results from a more sophisticated transport analysis code. An example of a CAAN output is shown in Fig. 3. CAAN has proved to be immensely valuable in guiding plasma operation.

A. β -Scaling experiments

Several different techniques were used to either measure or infer $\langle \beta \rangle$ values from the experiments on PDX. The first technique, the diamagnetic β measurement, uses the measured toroidal flux in the toroidal field coils with correction for residual coupling to sources of flux to calculate the volume averaged, perpendicular plasma β , $\langle \beta_\perp \rangle$. The highest measured diamagnetic $\langle \beta_\perp \rangle$ was achieved during a 4-beam $P_{inj} = 5.5$ MW discharge (Fig. 4). For this shot, deuterium beams were injected into a hydrogen plasma to maximize the beam power absorbed. The $\langle \beta_\perp \rangle$ attained a value of $\geq 3\%$; this value was achieved in the difficult operating regime of limiter q value < 2 ($q_a \propto a^2 B_T / R I_p$).

An alternate determination of $\langle \beta \rangle$ was obtained from the vertical magnetic field necessary to maintain radial equilibrium of the discharge. To obtain the "equilibrium" value, $\langle \beta_{eq} \rangle$, an approximation to the internal inductance was made. Because $\langle \beta_{eq} \rangle$ is the average of the perpendicular and parallel pressure from the plasma and beam particles, the $\langle \beta_{eq} \rangle$ was on the

average less than the $\langle \beta_{dia} \rangle$ by $\sim 12\%$ in FDX conditions, the difference between the two resulting from the anisotropy of beam particles due to the near perpendicular injection. A full MHD equilibrium code was also employed to determine $\langle \beta_{eq} \rangle$.

The last method of inferring β was to make use of the kinetic transport code, TRANSP.^{7,8} The transport analysis code is a data interpretation code analysis that models the ion and electron energy balance using the electron temperature and density profiles measured by the Thomson scattering system, line-averaged density from either the 2 mm microwave or far infrared interferometer, central ion temperature measured by a mass-resolving passive thermal charge exchange system, radiated power from a bolometer array, and plasma and beam characteristics. Beam deposition is modeled using a full Monte Carlo beam orbit calculation, and the thermal neutral density is modeled with a neutral transport code⁹ an assumed gross particle recycling time. An appropriate multiplier of the ion neoclassical conductivity is determined time-dependently within the code in order to match the evolution of the measured passive charge-exchange ion temperature. Power losses from the ions and electrons include convection, conduction, classical ion-electron coupling, charge-exchange, and radiation. From the stored energies of the plasma and the beam, the code computes both the diamagnetic and equilibrium β -values, yielding remarkably good agreement with those values obtained experimentally. A comparison between the TRANSP code diamagnetic β and that measured directly from the flux loop for a number of discharges is shown in Fig. 4, where excellent agreement is seen for all except the highest $\langle \beta_1 \rangle$ values. Results from the code also corroborate the fact that the differences between the equilibrium and diamagnetic measurements were due primarily to the calculated beam anisotropy. In FY82, a total of 85 well-documented

experimental conditions were analyzed using the code.

To study the statistical trends of the β -scaling experiments, data from over 1000 PDX discharges, selected on the basis of shot quality and diagnostic coverage, were incorporated into a computer data base. Linked to the data base are several interactive programs enabling the user to read, sort, and display the stored parameters in a variety of fashions, as well as to fit the data to regression curves.

The majority of the β -scaling experiments were carried out in the rail limited circular cross section plasmas limited by the top water-cooled rail limiter, with $R = 138$ - 143 cm, $a = 40$ - 44 cm, and $\bar{n}_e = 3$ - 6×10^{13} cm $^{-3}$. The toroidal field was varied from 0.7 - 1.5 T and the plasma current from 240 - 450 kA to study the dependence of $\langle\beta\rangle$ on these variables. Deuterium injection was used in order to maximize the absorbed power. Some size, configuration, and plasma and beam species scalings were performed. Figure 5 is a plot of the distribution of data as a function of B_T and I_p used in the β -scaling studies, and is a subset of the data points contained in the data base. Each circled cluster of points represents a fully documented, controlled power scan at fixed B_T , I_p , and if possible, $\bar{n}_e$. Note that this plot contains both the set of points obtained in FY82 for the β -scaling as well as data obtained in FY81 at 2.2 T ($I_p = 200$ - 480 kA). No diamagnetic β -values are available for this latter set of data. The family of data points is bounded on two sides. The 10% orbit loss boundary occurs near $I_p = 280$ kA, and represents the boundary below which $> 10\%$ of the injected beam ions are lost from the plasma due to classical guiding center bad orbits. For very low q_a , disruptions prohibited successful discharges, consequently the $q_a = 1.7$ boundary represented a true physical limit on the machine operation. Within this operating regime $\langle\beta_i\rangle$ values of up to 3.2% and β_{pol} values up to 1.8 were achieved.

The β -scaling investigations began with an analysis of the behavior of $\langle\beta_1\rangle$ with input power and current at fixed toroidal field. At high toroidal field and/or low input power (0-2 beams), $\langle\beta_1\rangle$ was found to increase linearly with input power and current. However, as the toroidal field was reduced and input power increased, there appeared to be some saturation in $\langle\beta\rangle$ with beam power. An example of this result for the 1.2 T power/current scan is shown in Fig. 6a, where $\langle\beta_1\rangle$ is plotted as a function of the ohmic and absorbed beam input power for four different currents. The saturation of β is apparent. What is also evident in Fig. 6 is that while the values of the $\langle\beta_1\rangle$'s at fixed input power depend strongly on current, the degree of saturation, i.e., the shape of the curves, is the same for all currents. This feature is common to results obtained at different B_T as well, implying that $\langle\beta_1\rangle$ scales favorably with I_p (or $1/q_a$).

An analysis of the electron profile data, obtained from a 56 point Thomson scattering system, suggests a similar behavior for the electrons alone. At fixed B_T and beam power, improved electron heating and thus higher electron stored energy with lower q is evidenced by higher and broader T_e profiles at the higher current. It is also apparent from these temperature measurements that at fixed I_p there was only a minimal change in central T_e or profile going from two to four beams, reflecting the stored energy saturation. The electron density profile shapes were relatively unaffected by variations in either beam power or plasma current.

Having established that the degree of saturation was the same for all I_p at fixed B_T , but that the amplitude of $\langle\beta_1\rangle$ depended on I_p , the next question to be addressed related to the variation of the degree and amplitude of the saturation with B_T . It was found that the self-similarity of the curves shown in Fig. 6a was retained at the different toroidal fields;

however, the degree of saturation varied with B_T such that the greatest saturation was observed at the lowest field (0.7 T), and almost no saturation of $\langle \beta_I \rangle$ with input power was observed for $B_T > 1.5$ T. The data points from the full range of power scans could be reduced into a single "universal" curve by plotting the product $\langle \beta_I \rangle q_a^{1.1}$ as a function of $P_{TOT}/B^{1.4}$ for each point, as shown in Fig. 6b. The choice of the variable $\langle \beta_I \rangle q_a^{1.1}$ reflects the fact that at fixed B_T , $\langle \beta_I \rangle \propto I_p$ and that the degree of saturation for all currents is the same (see Fig. 7), while the abscissa, $P_{TOT}/B^{1.4}$, reflects the degree of saturation as a function of input power and toroidal field. The curve in Fig. 9 reveals that saturation occurred for $\langle \beta_I \rangle q_a^{1.1} \gtrsim 4.5\%$.

Although a systematic study of β -scaling with regards to plasma size, configuration, and beam and plasma species has not yet been made, some conclusions can nevertheless be drawn from the data that are available. When sorted by inverse aspect ratio, $\epsilon = a/R$, the major and minor radius scaling data in the saturated region were consistent with the limit of $\langle \beta_I \rangle q_a^{1.1}$ varying as ϵ . In the divertor configuration ($a/R = 37/134$ cm), the values of $\langle \beta_I \rangle q_a^{1.1}$ were similar in the unsaturated region to those of limited plasmas for $D^+ + H^+$ injection, while in the saturated region the values were lower, consistent with the dependence on inverse aspect ratio. Deuterium targets seemed to have higher values of $\langle \beta_I \rangle q_a^{1.1}$ in the unsaturated regime than did hydrogen target plasmas; however, there was no difference in the saturated regime. Finally, there appeared to be no difference in either the saturated or unsaturated regime for hydrogen instead of deuterium injection.

Because the kinetic transport analysis yields an accurate description of the thermal and beam power flow under the assumptions of the model, we were able to extract from the code a scaling of the thermal confinement times with the parameters of interest. Measured $\langle \beta_I \rangle$ values included the stored energy

contributions from the thermal plasma as well as the beam, so only a gross, not thermal, energy confinement could be calculated from $\langle \beta_{\perp} \rangle$. The global thermal confinement time is defined to be the stored energy in the thermal plasma divided by the net power flow out of the thermal plasma:

$$\tau_E = \frac{\text{Stored Energy}}{\text{Input Power} - d/dt (\text{Stored Energy})}$$

The confinement scaling was carried out for cases in the unsaturated regime, $\langle \beta_{\perp} \rangle q_a^{1.1} \lesssim 4.5\%$.

The confinement scaling for a complete power scan (0-4 beams) at four different currents and $\bar{n}_e \approx 4 \times 10^{13} \text{ cm}^{-3}$ is shown in Fig. 7 for two different toroidal fields. The abscissa is heating power, defined to be the total beam power into the plasma (absorbed beam power less bad orbit and charge exchange losses) plus the ohmic heating power. Points with $P_{\text{heat}} < 1 \text{ MW}$ were heated solely ohmically. What emerged for both toroidal fields and all currents is a clear degradation in the confinement time for neutral beam heated discharges, although there was little variation of τ_E for discharges with > 2 beams at a given current. There was essentially no variation of thermal confinement with toroidal field, but there was a strong scaling of τ_E with plasma current, confinement being greater at the higher current in close to linear fashion. These results confirm the trends in the scaling of gross confinement derived directly from the diamagnetic measurements.

To understand the difference in confinement as a function of current, the volume integrated energy balance at $r = a/2$ for $B_T = 1.2 \text{ T}$ is plotted as a function of plasma current for the 2-beam shots in Fig. 8. We see that the

dominant energy loss for the ions was classical ion-electron coupling: since the ions were hotter than the electrons, power flowed to the electrons via Coulomb collisions. The electrons absorbed this power, but promptly lost it via anomalous thermal heat conduction, making the electrons the dominant energy loss channel. The ion thermal conduction loss is at most $\sim 25\%$ of that of the electrons. Since the power balance changed very little with current, and the stored energy and thus confinement increased with current, it appears that the electron thermal conductivity transport coefficient, χ_e , must change with plasma current. Because electron conduction is the dominant loss, understanding how the electron thermal conductivity scales is the key to understanding the process giving rise to the confinement scaling.

The electron thermal diffusivity profile was determined from the calculated conduction loss and the measured electron temperature profiles. Once determined, the diffusivity profiles were compared to those predicted by a large number of published theories of anomalous electron thermal transport to test the applicability of those theories. Poor agreement between the calculated and theoretical thermal diffusivities was found. However, another approach to understanding the diffusivity scaling is to perform a regression analysis to find the variables which best order the conductivity values. The results of this analysis suggest that a simple scaling of χ_e with r , plasma radius, and B_p , poloidal field was sufficient to fit the χ_e variation to a high degree of accuracy ($\chi_e = 10^{11} r^{1.67} B_{pol}^{-1.79}$). In fact, these two parameters proved to be the dominant scaling parameters; adding variables such as temperature, density, or toroidal field did not improve the regression fit. Although the magnitude of the values χ_e were greater, their dependence on B_p is that expected from classical diffusion theory. The inverse dependence of χ_e with poloidal field supports the scaling of increased

confinement with increased current.

B. Fishbone activity

During high power NBI (3-4 beams) in discharges with low toroidal fields ($B_T < 1.2$ T), a new type of instability was observed on several diagnostics. The instability was observed on all discharges in the saturated β regime, that is, with $\langle \beta_{\perp} \rangle q_a^{1.1} > 4.5\%$. One of its more distinctive signatures was seen on the Mirnov coils, probes that measure fluctuations in the poloidal magnetic field. The activity was typified on the Mirnov coils by growing bursts of large amplitude $m > 2$ oscillations with a period of 2-6 msec and a frequency of ~ 10 -15 kHz within each burst. Because of its appearance on the Mirnov signal, these distinctive oscillations have been dubbed "fishbone activity" (see Fig. 9).¹⁰

The fishbone activity as observed on the Mirnov coils showed interesting characteristics. The fishbone repetition period varied linearly with toroidal field from $\tau_{fb} = 2$ msec for $B_T = 0.7$ T to $\tau_{fb} = 6$ msec at $B_T = 1.6$ T, and the mode growth rate varied inversely with B_T . The mode, which originated in a region around the $q=1$ surface, was found to rotate in the direction of ion precession. Both the mode rotation and the ion precession speed scaled as E/I_p . A precursor oscillation at much higher frequency, 50-150 kHz, was sometimes observed at the onset of each fishbone spike; this signal was also believed to have been localized near the $q=1$ surface. The amplitude of these poloidal field fluctuations was about 10 times larger in 250 kA discharges than in 400 kA discharges.

Observations of fishbone activity were not confined to the Mirnov coils. Figure 10 is a composite of fishbone activity on two different time scales as observed on the central chord of the soft X-ray (SXR) array, the

Mirnov coils, the fast ion charge-exchange detector (FIDE), and a fast neutron flux detector. The SXR signal in the left panel shows the repetitive bursts of the $m=1$ activity characteristic of the fishbone activity. Similar behavior was also observed in the TiXIX radiation from the center of the discharge. Under low- β conditions ($\langle \beta \rangle q_a^{1.1} < 3.5$) the fishbones were not present; instead a growing envelope of $m=1$ activity just before the fall of the sawtooth was observed. During the fishbone activity, the SXR signal exhibited an inverted sawtooth behavior near the plasma edge, similar to that behavior seen in both CIII and $L\alpha$ radiation. The sawtooth inversion radius occurred near $r=20$ cm ($\sim a/2$). Also seen in the figure is a burst of fast charge-exchange neutrals which escaped from the plasma at the time of each fishbone spike. Coupled to this was a drop in the neutron emission. The right hand panel in Fig. 10 shows, in an expanded time plot, the phase relation between the signals during one fishbone burst.

During each fishbone burst, line integral density fluctuations of magnitude $\sim 4\%$ were observed by the 2 mm microwave interferometer, along with $\delta n_e/\bar{n}_e$ fluctuations of order 5-10% as seen by both the microwave and CO_2 laser scattering diagnostics. The fluctuations as observed by the CO_2 scattering exhibited typical frequencies of 8-20 kHz and wavelengths > 6 cm; with frequency spectra typically having less harmonic structure than those from SXR. Precursors associated with the fishbones with frequencies from 20 to 150 kHz were sometimes observed. An interesting feature of the CO_2 scattering results is that the observed density fluctuations sometimes occurred without concomitant SXR fluctuations; however, when they did occur together, the fluctuations were in phase. Results from a transport analysis of injected scandium indicated an enhanced inward "diffusion" during fishbone activity. Other measurements in which fishbone activity was observed were the H_α

radiation, impurity radiation, and the thermal charge-exchange energy flux.

As previously mentioned, large bursts of escaping fast charge-exchange neutrals, originating from the beam ion distribution, were phase-coupled to the fishbone spikes. In general, the escaping neutrals were found to localize in the near-perpendicular region of the angular scan (from counter-parallel to counter-perpendicular). Because of the choice of angular scan, the escaping particles detected suffered charge-exchange on the counter (inner) leg of their banana orbits. Two examples of charge-exchange spectra are shown in Fig. 17. The top panel exhibits a near-perpendicular spectrum during a 2-beam discharge without fishbones, and is typical of beam slowing down through classical collisions. The bottom panel is an example of a spectrum observed during a 4-beam discharge with severe fishbones. Fishbone spikes are seen throughout the spectrum. Note that the FIDE detector sweeps in energy as a function of time, so the observed spikes were actually localized in time rather than in energy. Several features of the fishbone spectrum are worth mentioning. First, depletion of beam ions is seen from just below the injection energy ($E_{inj} = 45$ keV) down to $E \sim E_{inj}/2$. Also seen are spikes of escaping neutrals whose energies are $> E_{inj}$. In the most severe cases of fishbones, spikes from thermal energies up to ~ 75 keV were observed. In these cases spikes were also observed in the low energy target species neutrals. During the more moderate cases the spikes and particle depletion were confined to energies between $E_{inj}/2$ and E_{inj} .

Because of the apparent effect of the fishbones on the beam ion distribution, a phenomenological model of ion loss due to fishbones was developed in which a loss cone of near-perpendicular particles with energies > 35 keV was opened during each fishbone. Using a bounce averaged Fokker-Planck code with this fishbone loss model, the ion loss rate was adjusted to bring

the predicted charge-exchange signal into agreement with the measured one. Simultaneously, neutron emissivity calculations were carried out, the results of which showed agreement with the observed drop in neutrons, thus reinforcing the empirical validity of the ion loss model. This loss model was then incorporated into the transport analysis code in an attempt to reexamine the cases for $\langle \beta_T \rangle q_a^{1.1} > 4.5$ in which poor agreement was found between the observed and predicted $\langle \beta_I \rangle$ (Fig. 4) when only a classical beam ion loss process was considered. For these severe fishbone cases it was estimated that up to 40% of the beam heating efficiency was lost during the fishbone oscillations. Because of this loss, the predicted $\langle \beta_I \rangle$'s in these cases were reduced to a point where they agreed well with the observed $\langle \beta_I \rangle$'s.

The theory of these fishbone oscillations is still under development. Results from the PEST (Princeton Equilibrium, Stability, and Transport) code,¹¹ indicate that the onset of the fishbone activity occurs close to the critical β of the ideal internal $n=1$ mode. Furthermore, the critical β for ballooning modes ($n=\infty$) is close to the maximum β achieved in the PDX experiments. Measurements showed also that the observed mode rotation frequency was close to the precession frequency of beam ions with energies near the injection energy, suggesting that a resonance between the beam ions and the MHD mode could be responsible for the loss of these fast ions. Unless the fishbone activity can be controlled, this loss of heating power is potentially damaging for perpendicular neutral beam heated tokamak reactors.

C. Fluctuations and disruptions

A summary of the types of MHD activity observed on PDX is given in Table II. For normal discharges at low $\langle \beta_I \rangle q_a$ (≤ 1) very little MHD activity is observed on the Mirnov coils, while sawteeth are observed on the

SXR signal. At these low $\langle \beta_1 \rangle$ values a very low level of $m=1$ activity was seen on the SXR signal just before the drop of the sawtooth oscillation. For $\langle \beta_1 \rangle q_a \sim 2.5\%$, the sawteeth increased in amplitude and period. The $m=1/n=1$ mode, with a frequency of 1-5 kHz as observed by the central SXR array just before the crash of the sawtooth, is phase-coupled to $m \geq 2$ oscillations observed on the Mirnov coils. The frequency of the oscillations is about five to ten times greater than that in ohmic plasmas and rotated in the ion diamagnetic direction. The sawtooth period is ~ 20 -50 msec, while that in the ohmic plasmas is generally 5-10 msec. The energy content of the plasma is systematically larger at the top of the sawtooth than at the bottom, with a 5-10% variation in electron temperature and 2-5% variation in density observed over the course of a sawtooth.

The amplitude of $m=1$ oscillations appeared to have no further effect on the gross energy confinement of the NBI plasma, as calculated from the measured $\langle \beta_1 \rangle$ values, which was already degraded from that of ohmic. The situation with $\langle \beta_1 \rangle q_a > 4.5\%$ has already been discussed; strong bursts of $m=1$ on the SXR sawtooth were coupled to $m \geq 2$ bursts observed on the Mirnov coils giving rise to the fishbone activity (see section on fishbone activity for further details).

During low- q operations $m=2$ oscillations were sometimes observed on the SXR signal. During ohmic discharges a continuous $m=2$ oscillation in the SXR's and Mirnov coil led to a degraded energy confinement. During NBI, sawtooth oscillations in the SXR signal broke into a continuous $m=2$ oscillation, with an associated $m=2$ signal in the Mirnov coils. When this happened, a clear decrease in the $\beta_p + \beta_i/2$ signal as well as energy confinement time was observed. This $m=2/n=1$ activity modulated the line average density, the neutron emission, and the charge-exchange flux. No fast ion loss was

observed. The mode was found to rotate in the electron diamagnetic direction at a frequency between 0.2 and 2.0 kHz.

Higher frequency oscillations (3-200 kHz) in the PDX tokamak have been studied using small-angle scattering of CO₂ laser radiation to measure the chord averaged wave-vector and frequency spectra of density fluctuations in both circular and diverted plasmas. Typical wave vectors of the fluctuations were $k = 3 \text{ cm}^{-1}$ with a broad frequency spectrum ($\Delta\omega \sim 100 \text{ kHz}$ at $k = 3 \text{ cm}^{-1}$). For ohmic plasmas there was a slight trend of increased wave number k with increasing line averaged density. The spatial distribution of the density fluctuations peaked at the minor radius corresponding to that of the limiter or separatrix, and then decreased monotonically from the edge to the center of the plasma (Fig. 12). The peak at $\sim 10 \text{ cm}$ in the upper panel of Fig. 12 was determined not to be real. Typical fluctuation amplitudes were $\delta n/n = 0.5$ at the boundary in both circular and diverted plasmas and $\delta n/n \leq 0.02$ near the plasma center. In divertor plasmas $\delta n/n$ decreased less rapidly away from the edge than in circular plasmas. There was little change in fluctuation amplitude, frequency, and wave vector spectra from OH to NBI; $\delta n/n$ increased to within a factor of ≤ 2 and showed little dependence with beam power.

A method has been developed to determine the group velocity of the wave from the CO₂ scattering signals by measuring the time delay of the maximum in the crossed beam correlation signal as a function of vertical position. The slope of the plot of x_0 , the inter-beam spacing, versus τ , the time delay giving rise to the maximum of the cross-correlation function, yields the group velocity of the waves. For divertor plasmas a propagation velocity of $\sim 1.3 \times 10^5 \text{ cm/sec}$ in the electron diamagnetic drift direction was observed.

The 2 mm microwave scattering system became operational in October

1981, and has been used to measure density fluctuations in the drift wave parameter regime ($f < 500$ kHz, $k < 15$ cm⁻¹, $k\rho_i \gtrsim 1$). During low field operation, scattering was observed primarily for $k_{\perp} < 7$ cm⁻¹. Since the size of the scattering volume increases with decreasing $k_{\perp}$, the spatial resolution of the measurements has been restricted. Average fluctuation levels of $\delta n/n \sim 0.002$ - 0.006 have been measured. It was found that $\delta n/n$ did not change between the inside and the outside (in major radius), with the wave spectrum being narrower on the inside than on either the outside or the center of the plasma. Doppler shifts in the direction of the electron diamagnetic drift were seen on the outside during ohmic plasmas, and additional Doppler shifts, consistent with the direction of toroidal rotation, were observed during both co- and counter-injection ($v_{\phi} \sim 5 \times 10^5 - 2 \times 10^6$ cm/sec). No Doppler shifts were seen on the inside. The density fluctuations increased up to a factor of two with NBI from their ohmic level, consistent with the CO₂ scattering results, but showed little dependence with injected power at constant line averaged density.

During inner wall operations of PDX, a very fast disruption ($\dot{i}_p > 45$ kA/ms) was often observed at $q_a \approx 3$. Analysis of the Mirnov coil signals indicated that an $m=3/n=1$ mode was present in these cases, as compared to the $m=2/n=1$ mode that was generally responsible for slower disruptions in the rail limited plasmas ($R \approx 140$ cm). In the rail limited plasmas, the $m=2/n=1$ mode if not controlled, resulted in disruptions when $q_a = 2$. A comparison of these two cases revealed that the magnetic fluctuations of the two modes were comparable; $\delta B/B \sim 1$ - 2% for the $m=2/n=1$ mode while $\delta B/B \sim 0.5$ - 2% for the $m=3/n=1$ mode. The mode frequencies were 1.8 - 2.2 kHz for $m=2/n=1$ and 2.6 - 4.0 kHz for $m=3/n=1$. A significant difference between the two modes was the growth rate of the instabilities, with the $3/1$ mode growing about three

times faster than the 2/1 mode. Also, the duration of the saturated state was much longer in the 2/1 than in the 3/1 case. At present it is believed that the speed of the disruptions on the inner limiter may have been due to the motion of the plasma against the limiter.

D. Impurities and impurity transport

A general monitor of impurity radiation in PDX discharges has been the 19 channel bolometer array. However, this diagnostic is sensitive to both impurity radiation and charge-exchange flux. To determine the fraction of the bolometer signal due only to impurity radiation above, several techniques were employed. In Fig. 13, spectroscopic measurements of impurity radiation are compared with bolometric measurements as a function of beam power in circular discharges with carbon limiters. For ohmically heated plasmas good agreement was observed. In beam heated discharges the spectroscopic measurements accounted for only half of the bolometric power.

This conclusion concerning the bolometer signal was corroborated by two other independent techniques. The first technique utilized the beam orbit simulation in the transport analysis code, TRANSP, to determine the beam charge-exchange efflux dependence on toroidal viewing angle. The charge-exchange efflux varies strongly with this angle because of the velocity space anisotropy of the beam particles. By contrast, radiated photon power should vary only slowly with toroidal viewing angle. Using bolometric measurements at two closely spaced toroidal angles, and comparing these with the predicted toroidal dependence of charge-exchange efflux, the charge-exchange was determined to be a substantial fraction ($> 50\%$) of the bolometric signal.

The third method of estimating the contribution of charge-exchange to bolometric measurements relied on exposing two samples of poly-methyl-

methacrylate (PMMA) a polymer sensitive to both radiation and impinging particles during neutral beam heating experiments. The two samples of PMMA were identical; a $1.6\mu\text{m}$ layer of PMMA on top of a silicon wafer exposed through a mask. On one sample, however, an additional $0.07\mu\text{m}$ layer of chromium was evaporated. Additional information on the contribution of the charge-exchange flux could be extracted since the attenuation coefficients of radiation and charge-exchange neutrals are different through the chromium film. The samples of PMMA were exposed near the limiter, and the principle conclusion of the experiments indicates that, at this location, $\sim 80\%$ of the incident energy was due to beam charge-exchange neutrals.

The most abundant impurities in tokamak plasmas are usually low-Z elements such as carbon and oxygen. At electron temperature $> 1\text{ keV}$, these elements are completely ionized and consequently cannot be observed with conventional spectroscopic techniques. Time and space resolved detection of the fully stripped impurities was made possible by a technique utilizing the radiation induced by charge-exchange collisions between the fully stripped ions and neutrals from a highly collimated, low power (5-10 keV) diagnostic neutral beam (DNB) in the field of view of a grazing incidence spectrometer.¹² Square wave modulation of the DNB allows discrimination of the desired signal from the background. The DNB can vary its angle of injection so that the beam is scanned across the plasma midplane in the line of sight of the spectrometer.

An example of steady-state radial profiles of C^{6+} and O^{8+} measured by the charge-exchange recombination technique in PDX ohmic discharges is shown in Fig. 14 along with model calculations. A noticeable feature of the O^{8+} and C^{6+} profiles is that both species have appreciable densities beyond the limiter radius at 30 cm. Since the electron temperature at the plasma edge is

too low to produce C^{6+} and O^{8+} , impurity transport across the magnetic field in times shorter than the recombination times of 50 to 100 ms must be responsible for the presence of these ions at large radii. The amount of cross-field transport necessary to account for the observed profiles can be estimated by comparing the measured radial profiles to those calculated by a multispecies impurity transport code.¹³ The curves in Fig. 14 show results of steady-state calculations assuming either coronal equilibrium or relatively simple diffusion models. Both the C^{6+} and O^{8+} show clear deviations from coronal equilibrium, especially at large radii. The measured profiles are in much better agreement with profiles calculated assuming a constant diffusion coefficient of $D = 10^4 \text{ cm}^2/\text{s}$, which is the value obtained from analysis of earlier scandium injection experiments.

E. Asymmetries, spatial structure, and plasma rotation

Electron temperature and density profiles in PDX discharges are measured by a 56-channel Thomson scattering system with a spatial resolution of 1.6 cm. During high power heating, both the temperature and density profiles undergo substantial changes, as shown in Fig. 15. In the ohmic phase, both the temperature and the density profiles show a great deal of symmetry about the magnetic axis, with the temperature profile somewhat hollow. When the beams were turned on, the density profile primarily broadened, while the electron temperature profile increased symmetrically. Further into the beam pulse, both the temperature and density profile exhibit a skewing to the outside, the skewing being more pronounced for the density profile. The bottom panel of Fig. 15 shows the profiles ~ 130 msec into the beam pulse; the skewing of the temperature profile is consistent with the Shafranov shift for this particular discharge while the density profile

asymmetry exceeds this shift. It was found that while the amount of density profile asymmetry was independent of beam power, toroidal field, and plasma current, it depended heavily on the plasma density, with a greater asymmetry at a higher density.

Several possible explanations for such asymmetries have been examined. The first explanation was that the centrifugal force due to plasma rotation would cause the particles to move outward. Calculations indicate that to account for the outward shift, toroidal rotation velocities of order 5×10^5 m/s were needed. Plasma rotation measurements were made on PDX using the Fast Rotating Mirror System which detected Doppler shifted UV light of the magnetic dipole transitions in high-Z elements. To facilitate such measurements titanium injection by laser blowoff was necessary. Measured rotation velocities were at least four times lower than those necessary to give rise to the observed asymmetries, ruling out this mechanism as the sole cause. A second method which has been examined quantitatively using TRANSP is the classically expected poloidally asymmetric trapping of beam particles. Either alone or in conjunction with the observed rotation velocities this mechanism is also insufficient to produce the observed asymmetries.

Vertical asymmetries in the low-ionization states of carbon (CII, CIII), whose source is not poloidally symmetric and is either the limiter or the wall of the vacuum vessel, have also been observed. These impurities were often observed to peak outside the limiter radius, where they were most asymmetric, and the asymmetry itself propagated inward through some anomalous radial diffusion mechanism thus affecting higher charge states. Potential explanations for this observation include an asymmetry in the neutral hydrogen profile, due to local recycling and gas valves, influencing the CII and CIII profiles through charge exchange. A quantitative prediction of asymmetry due

to this process is difficult. Another mechanism is the vertical drift of the carbon impurities out of the plasma and onto a limiting surface. Three other potential sources for the asymmetry are also worth noting: asymmetric gas puffing, changes in edge temperature, and poicidally asymmetric radial diffusion coefficients. Studies of all the possibilities are presently underway.

F. Power handling and limiter conditioning, edge studies, and plasma fueling

These studies were carried out in the three different plasma configurations shown in Fig. 1 in order to assess impurity control techniques as well as the power handling capabilities of the various limiting surfaces. To maintain densities of $1-4 \times 10^{13} \text{ cm}^{-3}$ in rail limited discharges, a gas flow rate of $\sim 10-12 \text{ Torr-l/s}$ was required.¹⁴ Pressures measured at the limiter and in the divertor chamber were $\sim 1-2 \times 10^{-5} \text{ Torr}$ and $< 10^{-7} \text{ Torr}$ respectively, for these discharges. The bolometrically measured power accounted for $\sim 30\%$ of the total input power with approximately 50% of the bolometer signal accounted for by impurity radiation. Negligible amounts of central radiation ($< 100 \text{ MW/cm}^3$, $\sim 10\%$ of central input power) were measured by the 19-channel bolometer array. The impurities responsible for the impurity radiation over the entire plasma were oxygen:titanium:carbon in a 50:40:10 ratio, while titanium alone accounted for the majority of the central radiation ($< 50 \text{ MW/cm}^3$). With the onset of neutral beam heating, Z_{eff} rose from ohmic values of $\lesssim 2$ to values $\lesssim 4$, the rise in Z_{eff} presumably due to an influx of carbon and oxygen. Long-term conditioning of the limiting surfaces was inferred by measuring the Z_{eff} of the plasma over the long-term operation. The cleanup was clearly observed as Z_{eff} approached unity after

six months of FY82 operation. In comparison, low Z_{eff} was obtained in divertor operation with minimal conditioning.

A detailed study of short-term limiter power handling and conditioning was performed making use of an array of thermocouples which measure absorbed power 0.5 cm below the surface of the top limiter and a nonlinear cooling model with known thermal properties of graphite to infer the limiter surface temperature and energy flux. The limiter is made out of graphite with a surface area of 800 cm²; it is copper-bolted to a water-cooled backing plate, the copper bolting providing a high thermal conductivity path. During a typical 4-beam discharge the limiter surface temperature reached a value of ~ 1600°C at a peak power flux of 3.5 kW/cm². After temperature equilibrium was reached, the limiter cooled with a time constant of 326 sec corresponding to a cooling coefficient of 9.2 W/°C. Figure 16 shows the fraction of input energy that was deposited on the limiter over a range of toroidal fields and plasma currents. The fraction of input energy that went to the limiter was a function of beam power, initially increasing then decreasing, but there was little change with current and toroidal field. On the other hand, the bolometer power was a constant fraction of the input power, indicating that the change in the fraction of energy going to the limiter was perhaps due to the local plasma-limiter interaction and not due to changes in other energy loss channels.

To study the energy flux into the edge plasmas a combined electrostatic calorimeter probe was utilized.^{15,16} The active elements of the probe are 0.5 x 3 x 8 mm metal plates with thermocouples bonded to the back and tightly secured to a ceramic probe head. The probe operates in one of two modes. In the nonrotating mode, the elements are shielded in such a way that they accept heat flux exclusively from the entire hemisphere into which they face. In the

rotating mode, the probe is shielded by an enclosing cap of graphite or boron nitride permitting selection in velocity space (energy and pitch angle). The probes are mounted in the PDX midplane generally 180° from the rail limiters. During neutral beam heated PDX discharges, absorbed powers of $\sim 1 \text{ kW/cm}^2$ have been measured; integrating the heat flux around a toroidal annulus yields $\sim 1.1 \text{ MW}$. In addition, the angularly resolved heat fluxes showed a peak in velocity space consistent with the heat flux being carried by particles with energies of $\sim 45 \text{ keV}$ and pitch angles of $\alpha = \tan^{-1} v_{\perp}/v_{\parallel} = 81^\circ$, precisely the pitch angle and energy of the injected beam ions. The 1.1 MW of power is consistent with that power level predicted for classical bad orbit loss of beam ions. Thus these edge studies indicate that a substantial portion of the edge heat flux was carried by beam particles and that these ions underwent prompt bad orbit loss with a minimum of pitch angle scattering.

The PDX inner wall bumper limiter, a prototype for the TFTR limiter, consists of TiC coated graphite tiles (70% of total wall area) with a water-cooled stainless steel backing. The rest of the wall is made of titanium tiles except for small breaks for diagnostic access. To maintain a constant density discharge ($\sim 2.5 \times 10^{13}$), gas flow requirements increased by $\sim 60\%$ over the rail limiter cases. The gas pressure at the limiter for the inner wall discharges remained at $\sim 1 \times 10^{-5}$ Torr while that in the divertor jumped to $\sim 2 \times 10^{-6}$ Torr as the plasma contacted the bumper limiter. Thus, a significant fraction of the input gas flowed to the divertor chamber contributing to the observed pressure rise and implying a delocalization of recycling. This last inference was supported by the delocalization of the H_α emissions, also an indicator of recycling. No special conditioning of the inner wall was needed, as $q_a > 3$ discharges were obtained with ease. However, as previously discussed, fast disruptions occurred when q_a was lowered. A few

discharges with $2.5 < q_a < 3$ were eventually obtained. To achieve and maintain inner wall discharges with $q_a \sim 2.5$, it was necessary to ramp the current up $\sim 40\%$ faster than for $q_a = 3$ discharges.

The impurity levels and radiated power in the inner wall discharges were difficult to obtain because of limited diagnostic access; thus the global averages from spectrometers and bolometers were unreliable. Using the charge-exchange recombination technique, the levels of O^{8+} and C^{6+} were found comparable to those in rail limiter discharges. Titanium levels in the core were also comparable to those in rail limiter discharges, as were ohmic and NBI values of Z_{eff} .

A toroidal average of total power incident on the bumper limiter, measured by an array of thermocouples around the machine, indicated that 40% of the input power was deposited on the wall. With a six-minute shot cycle, the Ti section reached a steady-state bulk temperature of 50-100°C, with a surface temperature of $\leq 200^\circ\text{C}$ with high power injection. Typical load levels with $P_{inj} = 6$ MW were estimated to be ~ 0.2 kW/cm². The vertical power deposition profile was measured by a vertical thermocouple array behind a beam armor section. The vertical profile of the change in bulk temperature per shot for both ohmic and NBI discharge exhibited a vertical asymmetry consistent with the direction of ion magnetic drifts.

Inside D diverted plasmas produced cleaner plasmas with less conditioning than either the rail or bumper limiter, with Z_{eff} approaching 1. Responsible for this lowered Z_{eff} was a decrease in the amount of low-Z impurities in the discharge (carbon and oxygen): their levels were 50% of those in rail limiter discharges. However, the radiated power measured by the bolometers was still 30-40% of the total input power, similar to that for circular plasmas. The amount of titanium in the core for medium densities ($\sim$

$3 \times 10^{13} \text{ cm}^{-3}$) was similar to that for circular discharges, implying that there was little extra titanium influx due to power loading of the neutralizer plates. The titanium in both circular and divertor discharges must have originated from the same source, presumably the protective titanium liners close to the main plasma.

An important difference between diverted and limited plasmas is that it took approximately five times the gas flow (45-50 Torr-l/s) to maintain a constant density in the diverted than in the circular plasmas. Pressures in the divertor chamber, at $\sim 1 \times 10^{-4}$ Torr during OH and $\sim 4 \times 10^{-4}$ Torr during NBI, were also significantly higher. This along with three times greater H_α emissions indicates the potential for high recycling in a moderate-high density inside D configuration. In addition to the 40% of input power measured by the bolometers, $\sim 60\%$ of the remaining energy was measured by the thermocouples on the neutralizer plates. The energy deposition profiles on the neutralizer plates were quite narrow for $\bar{n}_e \sim 3 \times 10^{13} \text{ cm}^{-3}$, only a few centimeters wide, and peaked at the intersection of the separatrix and the neutralizer plates. Peak power loads for these plasmas with NBI were $\sim 0.75 \text{ kW/cm}^2$, $\sim 1/4$ of the peak load on the limiters for circular discharges.

When the density was raised to $\bar{n}_e \sim 6 \times 10^{13} \text{ cm}^{-3}$, the energy deposition profile on the neutralizer plates was found to broaden, and the peak power loading was reduced, as shown in Fig. 17. These profiles were obtained on the outer neutralizer plate of the inner divertor which absorbed $> 90\%$ of the power incident on the divertor plates. Pressures $> 1 \times 10^{-3}$ Torr were measured in the dome. In the high density case, the core titanium density was reduced by a factor of five with respect to the lower density case, indicating improved shielding by the divertor scrape off in the main chamber, reduced penetration of titanium from the divertor region, or the

presence of a cold plasma causing reduced sputtering of Ti from the protective liners.

Measurements of power flow in the scrape-off plasma made by the biased calorimeter probe are shown in Fig. 18. The figure shows the radial variation of the heat flux for a variety of PDX discharge configurations with and without neutral beam heating. The heat flux of both Inside D and circular scrape-off plasmas had e-folding distances of ~ 2.5 cm. Neutral beam heating increased the edge conduction and convection power flow in proportion to the total energy input. The observed invariance of the e-folding length from ohmic to NBI discharges suggests that radial transport of heat increased to offset any increase in parallel edge particle velocity. A summary of the power loading studies is given in Table III.

To study alternate methods of plasma fueling, a gas driven hydrogen pellet injection system built by the Oak Ridge National Laboratory Technology Division and capable of producing and injecting four pellets per discharge was tested. The pellets are driven by high pressure helium gas. The gas load delivered to the tokamak by the pellet injection system is reduced using large baffled chambers and Teflon guide tubes. Four straight guide tubes are used, each having a length of ~ 75 cm and an aperture of 5 mm. This guide tube system has improved the coupling of the injector and the tokamak, resulting in highly reliable injector operation.

Injection speeds of ~ 470 m/s and penetration lengths of ~ 16 cm were achieved in ohmic discharges using pellets each containing $\sim 1.3 \times 10^{19}$ hydrogen atoms. Such pellets, when injected on the midplane, are capable of increasing the discharge column density by $\Delta \bar{n}_e l = 1.2 \times 10^{14} \text{ cm}^{-2}$. An example of a four pellet injection event is shown in Fig. 19. Given in the left-hand panel is the plasma column density as measured by the midplane microwave

interferometer. Each pellet is clearly seen by an increase in the column density. In the right-hand panel is a photograph taken through a near tangential viewing port showing the ablation signatures of the four pellets.

Changes in plasma parameters were similar to those found during previous single pellet experiments,¹⁷ indicating that the pellet mass could be increased. Modifications to the injector were begun to increase the pellet mass to 12×10^{19} hydrogen atoms per pellet. When complete, these larger pellets will be used in conjunction with neutral beam heating experiments.

G. Counter-injection studies

Because of the apparent connection between the fast ion precession velocity and plasma rotation during fishbone activity, a counter-injection experiment was performed to study the effect of the ion precession in the opposite direction. For similar discharge conditions ($B_T \sim 0.9$ T, $q_a < 2$) approximately the same $\langle \beta_1 \rangle$ values were achieved in the counter- as in the co-injection case; however, no fishbone activity was observed with counter-injection. This last result may have been due to the opposite rotation directions of the fast ions and the plasma. On the other hand, there was evidence that the q in the center of the discharge was higher for counter- than for co-injection. It is believed that larger $q(0)$ may result in greater stability of the MHD mode giving rise to the fishbone activity.

A transport analysis of the counter-injection case was performed, and a comparison of the results between the counter and co- case is given in Table IV. We see in the table that even though the beam heating power was much lower in the counter case due to the greater classical bad orbit loss of counter-going beam ions, the measured $\langle \beta_1 \rangle$ values were comparable. This implies, as the table shows, a thermal confinement time for the counter-

injection case which is a factor of ~ 2 greater than that in the co- case, even in the absence of fission activity.

H. Diagnostic development

FY82 has been a fertile year for the development of tokamak plasma diagnostics on PDX. Diagnostics developed for PDX will certainly be adapted for use on larger machines such as TFTR.

The SPRED system¹⁸ (Survey, Poor Resolution, Extended Domain) is a compact, time-resolved, multichannel spectrograph developed to provide routinely impurity spectra information for tokamak plasmas. It operates in the vacuum ultraviolet spectral region; in high temperature plasmas, with central electron temperatures reaching several keV, the most intense emission lines fall within this 100-1700 Å wavelength regime. For instance, the major transitions of frequently observed low-Z impurities such as carbon and oxygen, which radiate at the plasma edge, fall in the 500-1600 Å range. The 100-400 Å range contains line radiation of highly ionized medium-Z impurities such as chlorine, titanium, and iron, elements sputtered from the limiters and the walls.

The ability of SPRED to monitor routinely machine impurities is made possible by its design; it is compact with a minimum of moving parts, moderate spectral resolution of 2 Å, high luminosity, and ultra high vacuum compatibility with synchrotron radiation sources for sensitivity calibration. A full spectrum may be recorded in 16 msec, or six or more spectral lines and their backgrounds can be monitored with sampling rates up to 1 kHz.

The SPRED instrument has three basic functions. First, it is used as a machine monitor for observing intrinsic impurities. An interesting result of the SPRED observation is that only the lower stages of ionization of carbon and oxygen (e.g., CIII at 977 Å and OII, OIII at 844 Å, OIV at 554 Å) showed large increases with the onset of NBI, while higher stages of ionization such

as OV at 630Å and OVI at 1032Å, which presumably reside in annular shells somewhat inside the separatrix, showed hardly any increase with NBI. The difference could be due to changes in the edge plasma (e.g., higher n_e in the plasma scrape off) or to enhanced recycling or shielding of impurities with NBI in the D plasmas. The increase in emission from highly ionized titanium was due to both the rise in $T_e(0)$, which results in more of the central T_i ionizing up to TiXIX and TiXX, and to rises in n_e and n_{Ti} .

The second operating mode of SPRED, polychromatic monitoring of the impurity radiation with ≤ 1 kHz resolution, was used in order to study impurity transport following, for example, laser injection of a trace of scandium into a PDX discharge. That the impurity is nonperturbing is borne out by noting that no appreciable change was observed in either the bolometer radiation or the line average Z_{eff} . The initial injection and subsequent decay of the scandium lines were clearly observable even with only a trace injection. Using either simple or more sophisticated impurity transport codes and known rate coefficients, impurity confinement times and transport coefficients may be calculated.¹⁹ Lastly, as already discussed, line intensities from SPRED have been used to estimate impurity radiation losses.

Besides providing information about the level of low-Z impurities in the plasma core, as already described, the charge-exchange recombination technique is capable of fulfilling several additional diagnostic needs of near thermonuclear plasmas. For instance, this charge-exchange technique offers a means of monitoring the space and time evolution of fusion products such as He^{++} . Pilot studies have been carried out in ohmically heated PDX discharges which show that even a small concentration of helium ($n_{He}/n_e \sim 3\%$), can be readily detected. Ion temperature profiles may be obtained by using the charge-exchange recombination technique to measure the Doppler broadened

profiles of low-Z impurity spectral lines.²⁰ This technique overcomes the necessity for the presence of high-Z heavy atoms to measure emission lines as well as the limitations of neutron and neutral charge exchange measurements in high temperature plasmas for ion temperature measurements. Thus, charge-exchange recombination is particularly useful for plasmas close to thermonuclear limits. In addition to measuring the Doppler broadening of the spectral lines, measurement of the shift of these lines allows a determination of the plasma rotation velocity.

A monitor of the overall impurity level in the discharge is the measurement of Z_{eff} , the effective or average charge of the combined ion species within the discharge. Simple estimates of Z_{eff} may be made through measurements of the resistive loop voltage, plasma current, electron temperature, and an assumed electrical resistivity model such as that given by Spitzer. Another method has been to use the charge-exchange recombination measurements of fully stripped carbon and oxygen, taking these elements to be the dominant impurity species, and noting that lower charge states of these atoms have a small (~ 5%) contribution to the overall impurity level in the discharge. The third method has been to use visible continuum radiation measured by double interference filtered detectors at 5235Å which view the plasma through 19 fiber optically coupled telescopes. Central chord data from this diagnostic have been combined with the appropriate model for electron density and temperature profiles to derive a line averaged Z_{eff} , with the results displayed to the operator as a "Z-meter" for real time monitoring. Data from the entire 19-channel poloidal array are Abel inverted to yield both temporal and radial profiles of Z_{eff} in the discharge. A comparison of these Z_{eff} measurement methods for ohmically heated PDX discharges is shown in Fig. 20. Good agreement between all the methods is seen.

Techniques to measure radial profiles of Z_{eff} and q using the modulated diagnostic neutral beam and fast neutral charge-exchange analyzer (FIDE) have been developed and are presently being refined. The Z_{eff} measurement is made by observing the competition between the thermalization and pitch angle scattering of the injected ions. This is accomplished by aiming different channels of the analyzer at different radii to look over a range of pitch angles, and by sweeping the detection energy over a small range around the injection energy to get a local slowing down spectrum of the beam ions in the plasma. A theoretical slowing down spectrum is then calculated by putting the known plasma parameters into a time dependent linearized Fokker-Planck equation. Since the amount of pitch angle scattering depends on the local Z_{eff} , the value of Z_{eff} is varied in the Fokker-Planck calculation until the experimental and theoretical spectra agree. This gives a local measurement of Z_{eff} .

The measurement of q is also made by injecting ions from the DNB into the plasma. Conservation of angular momentum implies that the injected ions will shift from the toroidal flux surface that they were injected on by an amount proportional to their parallel velocity and q . For a given accelerating potential of the DNB, ions will be injected at energies equal to the full, half, and third injection energy. The FIDE analyzer then observes these three different components at various radii, and the relative shift of the components yields a local measurement of q independent of the position of the magnetic axis.

An electron cyclotron heating (ECH) system is nearly completed for PDX to study localized electron heating and electron current drive. The waves will be launched from 60 GHz gyrotron oscillators located 70-100 feet away from PDX ports, with a maximum power of 400 kW and a maximum pulse length of

100 msec. Ordinary mode waves will be launched from the low magnetic field side directed radially inward at the horizontal midplane. The transmission of the wave energy will be monitored with an array of receivers. Mirrors will be used to launch waves from the high magnetic field side of the vacuum vessel. A ray tracing code, TORAY, is being developed which, when coupled to the transport analysis code, TRANSP, will aid in understanding and optimizing conditions for heating and current drive.

Presently under development for PDX is a diamagnetic loop capable of measuring the diamagnetic flux to within the required accuracy of $\leq 10^{-4}$ and of effectively decoupling the poloidal magnetic flux from the measured signal. Standard diamagnetic loops compensate for the poloidal flux by energizing the various sources of this flux in the absence of a plasma and determining the coupling between the sources and the loop. However, ignored in such a compensation is the poloidal flux generated by the plasma itself. Since the poloidal flux due to the plasma is opposite in sign from that produced by the coils, ignoring this flux may result in an overcompensation and incorrect result for the plasma diamagnetism. The technique used in the loop being developed makes use of a pair of saddle-shaped coils running very close to the diamagnetic loop which essentially see the same poloidal flux as the diamagnetic loop. In this way, all the poloidal flux can be rejected regardless of its origin.

A new diagnostic was developed to measure in situ the species yield of an NBI system under operating conditions, using an electrostatic analyzer and microchannel plate detectors to detect beam particles Rutherford backscattered 136° from the PDX inner wall armor in the absence of a plasma. Instead of relying on species yields deduced from past neutral beam performance, the species distribution can now be determined directly, between plasma shots, for

comparison with the slowing-down spectra obtained with the FIDE analyzer. Furthermore, the technique allows fast systematic measurements of the species yield, across a beam power profile, as a function of beamline operating conditions.

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Table I: PDX Milestones for Fiscal Year 1982

October 1981	Low-q operation in ohmic plasmas.
November 1981 - December 1981	Machine maintenance and diagnostic repair and upgrade.
March 1982	Low-q inner wall operation. Neutral beam injection into circular plasmas ($B_T = 1.2$ T).
April 1982	Low-q rail limiter operation. NBI into circular ($a/r = 40/140$) plasmas ($D^+ + H^+$, D^+); β -scaling, fishbone, and MHD studies, $B_T = 0.9-1.4$ T, $q_a = 1.8$, $\langle \beta_i \rangle = 3.2\%$ achieved.
May 1982 - July 1982	NBI into circular ($a/R = 35-46/125-140$ cm) and Inside-D plasma; β -scaling, fishbone, and MHD, counter-injection studies. $B_T = 0.7-1.5$ T, 140 kJ of stored energy achieved.
August 1982 - September 1982	Installation of new divertor hardware, scoop limiter, ECH system, vertical soft x-ray array.

Table II.

MHD ACTIVITY IN PDX WITH NEUTRAL BEAM INJECTION

	<u>Normal Discharge</u>			<u>m=2 Discharges</u>	
	<u>OH</u>	<u>Beams</u>	<u>Beams</u>	<u>OH</u>	<u>With Neutral Beams</u>
Activity Observed on soft X-ray signal
$\langle \delta \rangle_1$ q_a range	~ 0.01	~ 0.025	~ 0.045	~ 0.01	~ 0.025
X-rays Mirnov coil	Sawteeth and m=1; very low level of MHD Activity	Large sawteeth coupling to Mirnov coil at fall of sawtooth $m \geq 2$	Fishbone Activity, Strong coupling of m=1 to Mirnov coil, $m \geq 2$	m=2 mode m=2	Sawteeth, then m=2 Large m=2
Neutron Emission		Decrease (Fast Ion Disruption)	Decrease $\sim 40\%$		m=2 oscillations observed on neutron emissivity
Charge-Exchange Flux		Sawteeth observed on charge exchange, and spike at drop of sawtooth	Large burst of charge exchange during fishbone and depletion in slowing down spectrum		m=2 activity on charge exchange flux. No depletion of the fast-ion slowing down spectra
Conclusions	No transport effects.	No clear global transport effects observed	A loss of fast ions from the plasma. A drop in the calcu- lated I_E .	Decrease in confinement time is observed when m=2 activity is present	Confinement time is $1/2 + 2/3$ the confinement with sawteeth or fishbones

Table III. Power Loading in PDX Discharges With Neutral Beam Heating

	Graphite Rail (FY82)	Bumper Limiter (FY82)	Divertor (FY81)
Beam Species	$D^+ + H^+$	$D^+ + H^+$	$H^+ + D^+$
Δt_{Beam} (msec)	150	150	150
P_{inj} (MW)	4.4	4.1	4.5
P_{absorbed} (MW)	3.4	3.2	3.6
P_{OH} (MW)	0.2	0.3	0.2
$\bar{n}_e$ (cm^{-3})	2.5×10^{13}	2.5×10^{13}	4.3×10^{13}
Limiting Surface Material	C	TiC/C (70%) T_1 (30%)	Ti
$\frac{E_{\text{Limiting Surface}}}{E_{\text{IN}}}$	0.19	0.49	0.60
P_{Load} (kW/cm^2)	1.9	0.2	0.75
z_{eff} (OH-NBI) (Spitzer)	2.0-4.5	1.8-3.9	1.3-1.7
$\frac{P_{\text{Bolometer}}}{P_{\text{IN}}}$	0.30	N/A	0.40

Table IV.

	<u>Co-Injection</u>	<u>Counter-Injection</u>
P_{injected}	5.0 MW	4.7 MW
P_{heat}	3.5 MW	1.8 MW
β_{pol}	0.96	0.9
β_{tor}	2.95%	2.85%
$\tau_E(a)$	15 ms	35 ms

FIGURE CAPTIONS

- FIG. 1 Outer flux surfaces for different operating modes of PDX. The surfaces are defined for a) water-cooled graphite rail limiters ($R = 135\text{--}155$ cm). b) water-cooled titanium-carbide coated graphite bumper limiter for inner wall operation ($R = 125$ cm). c) Inside D diverted configuration ($R \approx 138$ cm).
- FIG. 2 a) PDX Neutral Beam Operating History; b) PDX Neutral Beam Ion Source Fault History.
- FIG. 3 Between shots confinement analysis output (CAAN).
- FIG. 4 Comparison of $\langle \beta_{\perp} \rangle$ as measured by the TF coil diamagnetic flux loop and as calculated in the kinetic transport code (TRANSP).
- FIG. 5 Range of I_p and B_T covered by PDX β -scaling experiments. Circled points represented fully documented, controlled power scans, $R = 143$ cm, $40 < a < 44$ cm. The $q_{cyl} = 1.7$ line is drawn for $a = 42$ cm.
- FIG. 6a $\langle \beta_{\perp} \rangle$ vs. P_{TOT} at $B_T \approx 1.2$ T for different currents. Curves have been fitted to the data to guide the eye.
- FIG. 6b $\beta_{\perp}^{1.15}$ versus $P_{TOT}/B_T^{1.4}$. Universal curve for all beam heated PDX diamagnetic data at $R, a = 143, 40\text{--}42$ cm.
- FIG. 7 Global thermal energy confinement times as a function of total heating power for two toroidal field cases and four currents for each case.
- FIG. 8 Ion and electron power balance as a function of current for the 2-beam, $B_T = 1.2$ T case.
- FIG. 9 Mirnov coil signal during fishbone activity. The "outline" is a result of the artist's imagination and was not measured on the Mirnov coil.

- FIG. 10 Typical signatures of strong fishbone activity on central soft X-rays, Mirnov coil, charge-exchange flux, and neutron emission on slow (left panel) and fast (right panel) time scales.
- FIG. 11 Perpendicular charge-exchange spectra for a 2-beam (no fishbone) and a 4-beam (strong fishbone) case.
- FIG. 12 Density fluctuations amplitude as measured by CO_2 scattering as a function of plasma radius.
- FIG. 13 Total power radiated from PDX plasma as a function of neutral beam input power. The dark circles are from the bolometer, and the open circles are from VUV spectroscopy. The inset shows radiation profiles as measured by the bolometer array.
- FIG. 14 Radial profiles of fully ionized oxygen and carbon during the steady-state phase of the discharge. Closed circles $R > R_{\text{major}}$, open circles $R < R_{\text{major}}$. The solid lines are radial profiles of C^{5+} and O^{8+} calculated from an impurity transport code. C.E. = distribution expected from coronal equilibrium assuming a constant impurity density. D = constant impurity diffusion coefficient.
- FIG. 15 Thomson scattering electron temperature and density profiles before (top panel) and ~ 130 ms into (bottom panel) neutral beam injection.
- FIG. 16 Fraction of input energy that is deposited on limiter as a function of input power for a power and current scan ($I_p = 250\text{--}450$ kA, 0-4 beams).
- FIG. 17 Energy deposition onto the neutralizer plates for an Inside D discharge as a function of plasma density. Position denotes height above bottom of neutralizer plate $P_{\text{inj}} = 4.5$ MW.

FIG. 18 Radial dependence of best flux in PDX: floating probe: inside D: $R = 138$ cm, $a = 38$ cm, $\bar{n}_e = 2.3 \times 10^{13}$ cm $^{-3}$, $I_p = 300$ kA, $B_T = 1.7$ T: ■ = Ion Side, □ = Electron Side, ohmic heated, ◇ = Ion Side during 1.7 MW NBI. Circular: $R = 138$, $a = 45$, $\bar{n}_e = 2.6 \times 10^{13}$ cm $^{-3}$, $I_p = 300$ kA, $B_T = 1.7$ T: ● = Ion Side, ○ = Electron Side. Grounded probe (all ion side): inside D: $R = 138$ cm, $a = 38$ cm, $\bar{n}_e = 2.4 \times 10^{13}$ cm $^{-3}$, $I_p = 360$ kA, $B_T = 1.7$ T: V = ohmic heated, X = during 1.7 MW NBI, ◆ = during 3.2 MW NBI. Four-Null Square: $R = 144$, $a = 42$, $\bar{n}_e = 2.9 \times 10^{13}$ cm $^{-3}$, $I_p = 360$ kA, $B_T = 1.2$ T: Δ = ohmic heated, ▲ = during 1.5 MW NBI, ⊕ = during 3 MW NBI. + = $R = 143$, $a = 42$, $\bar{n}_e = 3.9 \times 10^{13}$, $I_p = 420$ kA, $B_T = 1.2$ T ohmic heated.

FIG. 19 Four pellet injection into a PDX ohmic discharge (left panel) column density measured by midplane microwave interferometer. Arrows indicate the times of pellet injection. Photo of pellet ablation taken from a near-tangential viewing port (right panel).

FIG. 20 Comparison of central Z_{eff} values derived from charge-exchange measurement of low-Z impurity densities, visible bremsstrahlung measurements, and estimates from inferred plasma resistivity.

Figure 1

Figure 2

PDX MD CONFINEMENT ANALYSIS 29-JUL-82 53154
 CO 46.5 KEV D0 --> H+ SPIKES = 0 A/R = 40.0/142.6 15.1KG

Figure 3

Figure 4

Figure 5

Figure 6

Figure 7

Figure 8

#83X0023

Figure 9

Figure 10

Figure 11

Figure 12

#82X0466

Figure 13

Figure 14

Figure 15

Figure 16

82X0467

Figure 17

Figure 18

INCREMENTAL INCREASE IN
PLASMA LINE INTEGRAL DENSITY

NEAR TANGENTIAL VIEW
OF PELLET ENTRY

Figure 20

EXTERNAL DISTRIBUTION IN ADDITION TO TIC UC-20

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