Tokamak experiments at Institute for Plasma Research

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Introduction

THE quest for attaining controlled fusion power in the laboratory has been described as the most challenging scientific endeavour of the present century and the ultimate solution to the growing energy needs of the world. The notion that one is attempting to recreate the conditions of the interior of stars on earth has also led to such fanciful description of fusion energy as 'a bid to trap a star'. The imagery is quite apt, for one of the necessary first steps in the quest for achieving controlled thermonuclear fusion in the laboratory is the creation of an efficient trap to hold the reacting elements long enough for them to fuse. At the temperatures necessary for fusion (around 80 to 100 million degrees centigrade) the nuclear fuel is completely ionized and is in the form of a hot plasma. A convenient way to confine plasma is through the use of magnetic fields. Since charged particles execute tight helical trajectories around magnetic field lines, their transverse excursion is naturally restricted and they tend to stick to magnetic surfaces. Thus the basic idea is to create suitable configurations of magnetic surfaces that will confine the plasma most effectively. The tokamak is one of many such magnetic confinement devices designed to hold a plasma in physical and thermal isolation from the container walls. It is a toroidal magnetic trap and its name is a Russian acronym constructed from the words *toroid*, *kamera* (chamber), *magnit* (magnet) and *katushka* (coil). The concept was first proposed by Sakharov and Tamm and the first experiments on this concept were carried out at the Kurchatov Institute in Moscow by Artsimovich and his colleagues[1]. The basic configuration, illustrated in Figure 1, consists of a strong toroidal magnetic field created by a set of external coils wound around a toroidal vacuum vessel and a poloidal magnetic field created by inducing a toroidal current in the plasma itself with the help of another set of external coils. The combination of the two fields creates a set of

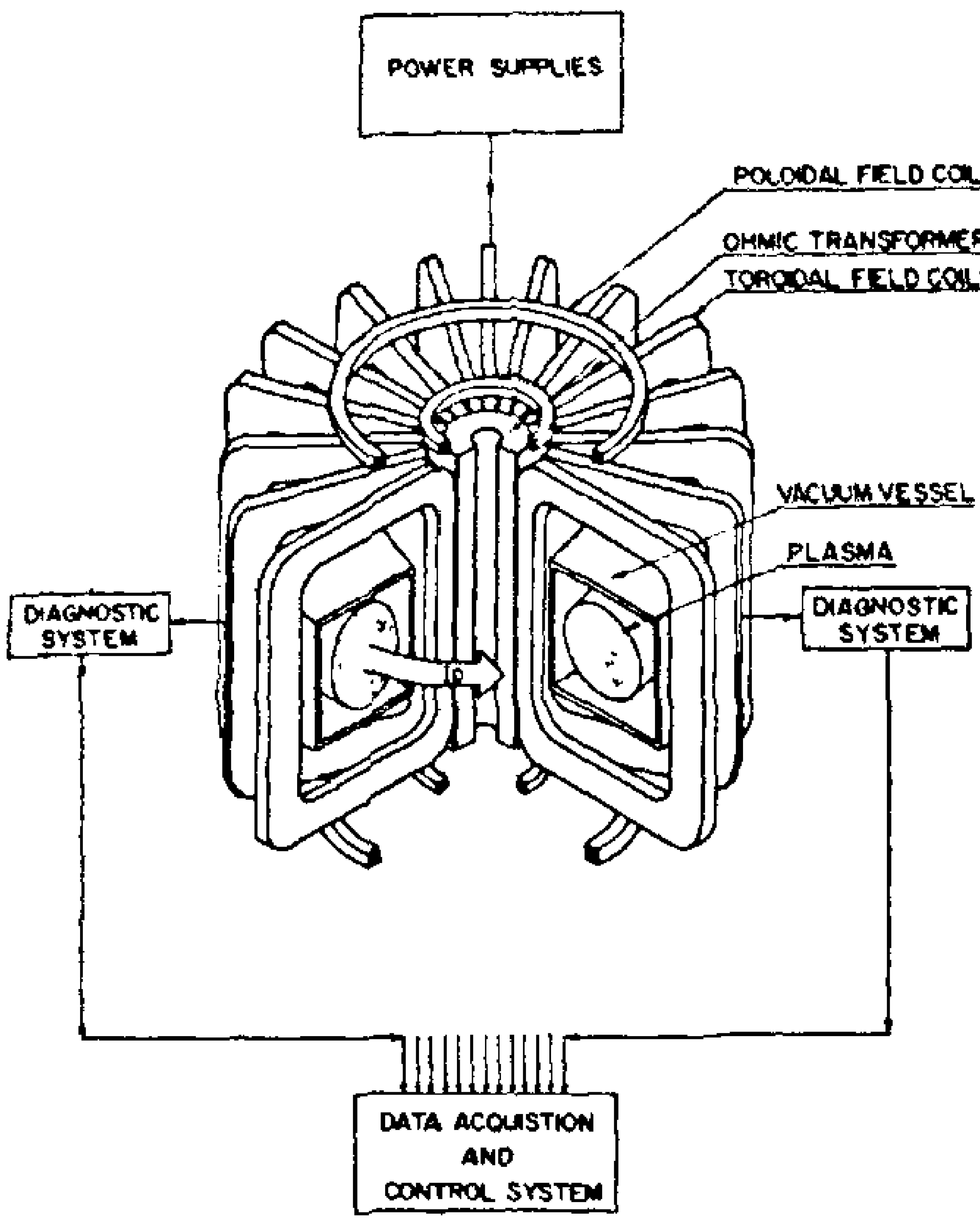

Figure 1. Schematic diagram of a basic tokamak machine.

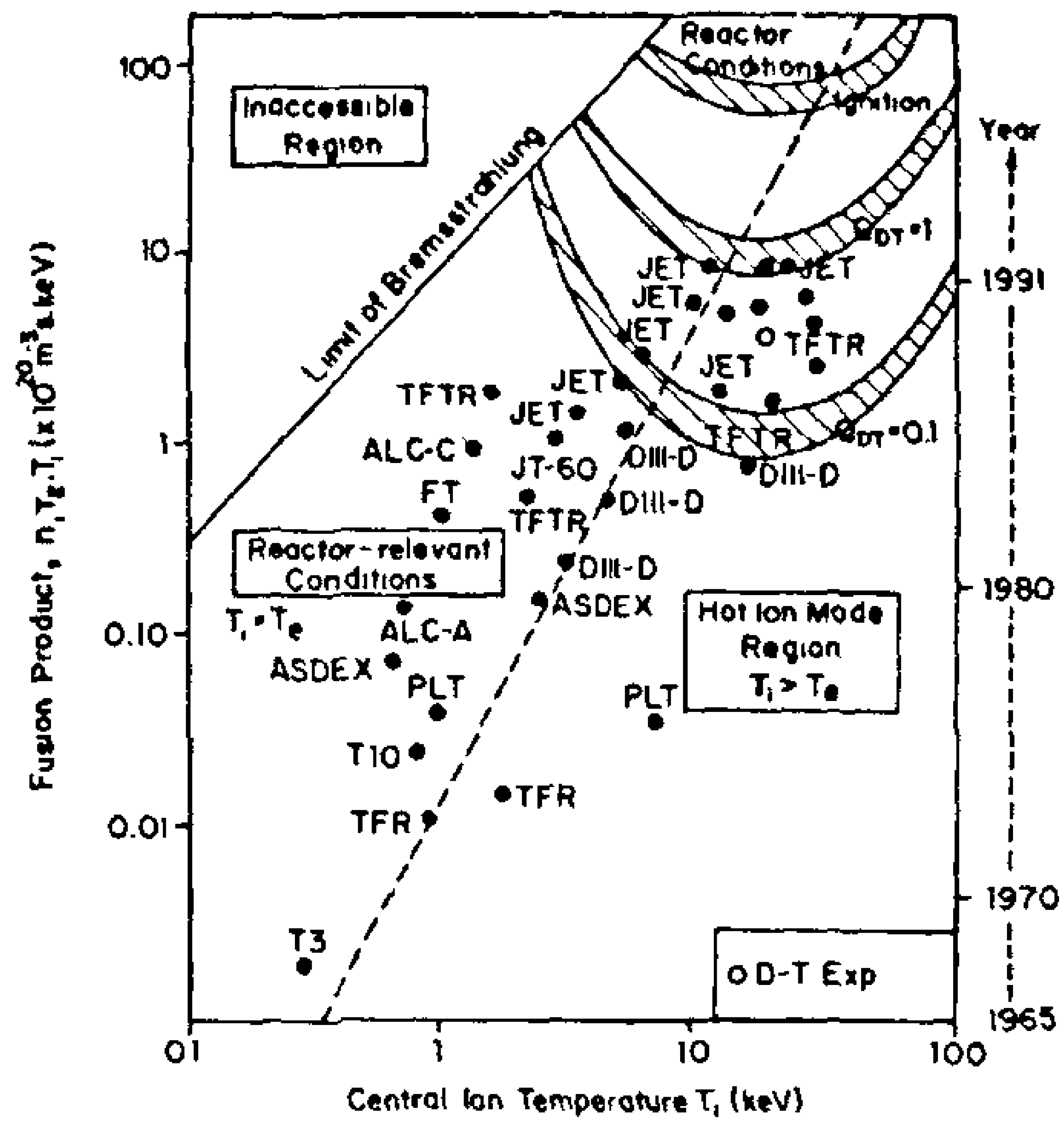

Figure 2. Parameter range achieved by various tokamaks.

nested toroidal magnetic surfaces to which the particle trajectories are confined. The toroidal current which creates part of the confining field also helps to heat the plasma by resistive Joule dissipation.

This simple configuration has several advantages over linear devices such as pinches and mirror machines which suffer from large end losses and early experiments clearly established this superiority. The T-3 device at Moscow showed a remarkable improvement in plasma confinement (a factor of about thirty over earlier results) and impressive electron temperatures. These early successes stimulated wide interest in tokamaks and soon they were built in many laboratories around the world. Today tokamaks are the most successful of the various fusion devices and have achieved the best experimental parameters to date. A convenient figure of merit to measure this progress is the quantity $n_i \tau T_i$, where n_i is the number density of the reactant ions, τ is the confinement time and T_i is the ion temperature. The condition for achieving scientific breakeven, i.e. when the amount of fusion reaction power equals the input power, is for $n\tau \sim 10^{14}$ cm^{-3} s and $T_i \sim 10$ keV. As Figure 2 illustrates, the progress in parameters for tokamaks from the early T-3 to the latest Joint European Torus (JET) is truly remarkable. Based on this success there is great optimism about building the first fusion test reactor as a tokamak device and designs of the International Thermonuclear Experimental Reactor (ITER) are in an advanced stage[2].

Physics issues

Although simple in concept the tokamak plasma is far from being that and constitutes a very complex nonlinear system which has continuously thrown up challenging physics problems. One of the major objectives of tokamak research, right from the early days, was gaining an understanding of the physics of confinement, i.e. the physics of transport of particles, heat and impurities. The rate at which energy leaves the tokamak is much faster than that predicted by classical collisional transport theory calculated for the tokamak magnetic field geometry (neoclassical transport theory). These 'anomalous' losses particularly of electrons and electron heat still remain a fundamental physics issue apart from posing a major obstacle to tokamak fusion. If these losses can be understood and therefore brought under control, one can confine plasmas for longer periods even in smaller devices and in principle build reactors from present sized tokamaks. The confinement problem is tied up with several other factors in a tokamak. In order to achieve thermonuclear temperatures, it is necessary to supplement the basic Ohmic heating in a tokamak by auxiliary heating using either energetic neutral beams or high power radio frequency waves. The additional heating however is found to lead to degradation of confinement properties. Some improvement in confinement can be achieved by tailoring the edge plasma whose properties seem to be deeply coupled to the confinement properties of the bulk plasma. The edge plasma is dominated by radiation processes, atomic physics and various plasma wall interactions. It is also the region where microscopic fluctuations are prone to acquire large values and their presence seems to have a direct influence on transport properties. Thus ultimately the 'anomalous losses' have to be understood in terms of the properties of the ambient plasma microturbulence — the nonlinear saturated spectrum of the various fine scaled collective oscillations of the plasma.

The tokamak plasma also exhibits a host of nonlinear phenomena associated with larger scale collective activities. These include Mirnov oscillations (associated with tearing mode activities), sawteeth oscillations (arising from thermal instability at the plasma core), disruptive instability (a catastrophic manifestation of the sawteeth instability in which the plasma discharge collapses), MARFE (a thermal-radiative instability occurring at the plasma edge), etc. These instabilities ultimately limit the operating parameter space of tokamaks. The operational limits of the tokamak, which have been empirically determined, are schematically shown in Figure 3 — the so-called Hugill diagram, in which the average current density ($1/q_a$) is plotted against the normalized electron density ($n_e R_0/B_T$). At a given plasma current there exist lower and upper limits for the plasma density in a tokamak discharge. At low densities, due to the lack of sufficient collisions, there is a tendency for the electrons to get accelerated to very high energies ('run-away' effect) by the inductive electric field and prevent a good discharge from building up. At high densities, the atomic processes (like radiation, charge-exchange, and ionization) at the edge become dominant leading to plasma

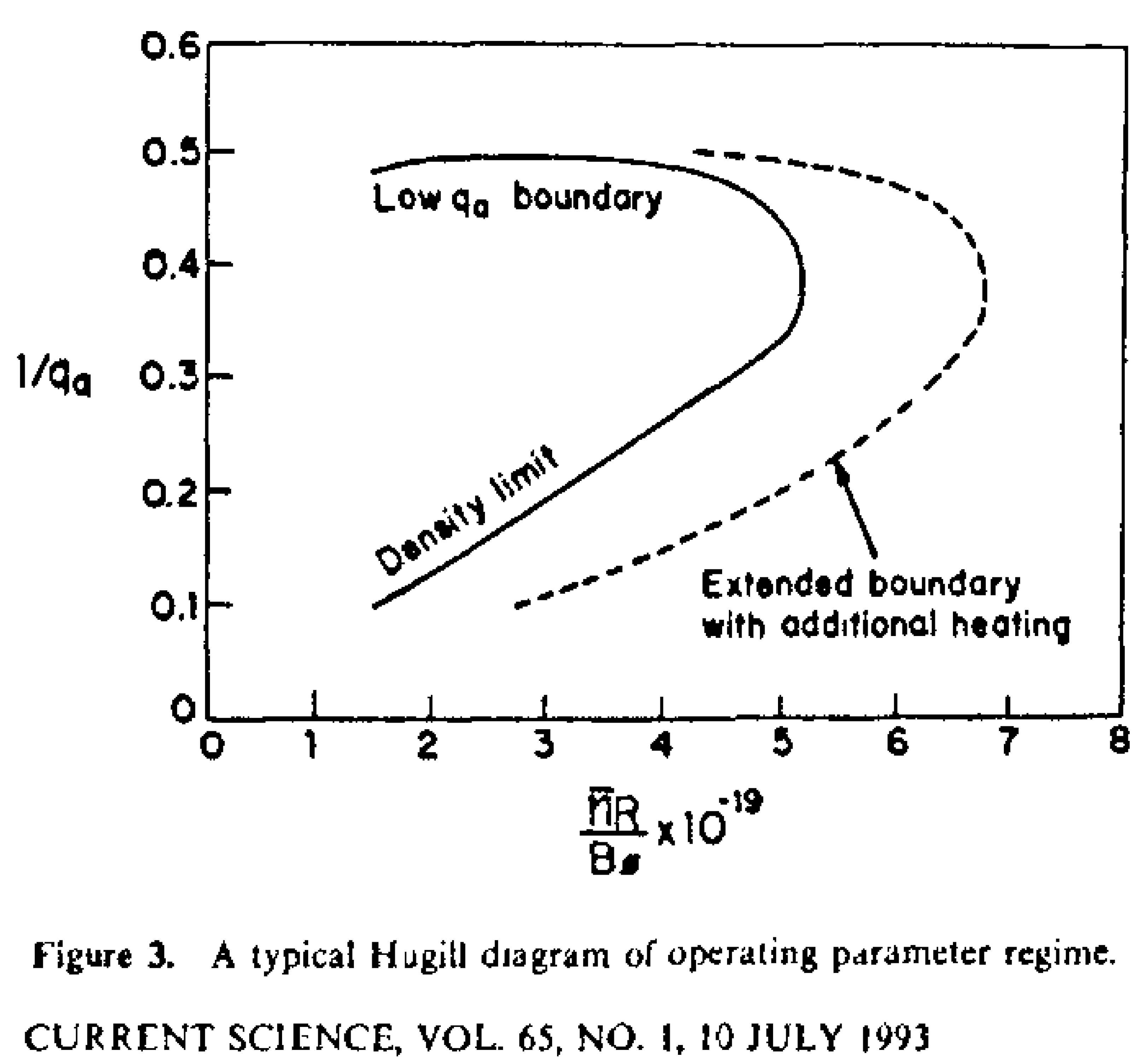

Figure 3. A typical Hugill diagram of operating parameter regime.

column contraction. Similarly for a given density, an upper limit to the plasma current is set by the MHD kink instability and a lower limit is set by the onset of MARFES. Although the broad premises of these limits are understood, their detailed understanding continues to remain a challenge. These limits severely constrain the operating regime of a tokamak and thereby have a serious impact on its suitability as a reactor device.

Two other important limitations of the tokamak are the pressure limit and its pulsed operating mode. The pressure limit is due to the onset of the ballooning instability — a 'bulging' of the magnetic surface in the outer regions where the convex curvature of the magnetic field lines tend to weaken the surface. The pressure limit is expressed in terms of β, the ratio of the plasma pressure to the magnetic field pressure and typically this ratio is limited to about 3%. For the most effective use of the magnetic field, it is necessary to raise the critical β value as much as possible. There are theoretical and some preliminary experimental evidence that indicate that it might be possible to steer the tokamak discharge into a high β regime which is stable to ballooning instabilities. The DIII-D tokamak at General Atomic in USA recently recorded the highest maximum β value at the magnetic axis of 44% and also the highest average β value of 10% (ref. 3). The pulsed operating nature of the tokamak arises because the plasma current in it is created and maintained with the help of an inductive current drive mechanism. This consists of varying the magnetic flux in a set of external coils to which the plasma ring is coupled like a secondary coil in a transformer arrangement. A pulsed operation would create severe mechanical problems in a reactor environment and hence it is necessary to develop noninductive current drive schemes that would enable a continuous operation. Various such schemes based on providing directed momentum to electrons with the help of electromagnetic waves or energetic neutral beams have been developed and appear promising. The best experimental results[4], obtained on the Japanese tokamak JT-60, show a current of 2 MA driven by electromagnetic waves at the lower hybrid frequency (a natural plasma resonance frequency).

Thus tokamak experiments, despite several areas of darkness, have shown remarkable all round progress — high temperatures, long confinement times and high densities. The large experimental data base has enabled the formulation of fairly detailed scaling laws which are being used to design the first experimental test reactor. Meanwhile, the quest for understanding the physics issues, which have the potential of providing major breakthroughs, continues worldwide on a host of machines, both big and small. It is against this background that the decision to start a tokamak-based experimental programme in India was taken. The aim is two-fold. The primary goal is to pursue a vigorous basic research programme that will address the many physics issues of high temperature magnetically confined plasmas in general and those of tokamaks in particular. An additional mandate is technological development — to create indigenous expertise in the construction of experimental devices for hot plasmas and to build an infrastructure within the country that will anticipate, critically evaluate and implement fusion technology if and when it is proven viable. Tokamak research is carried out at two laboratories in the country — the Institute for Plasma Research near Ahmedabad and the Saha Institute of Nuclear Physics, Calcutta.

Tokamak ADITYA

ADITYA is a medium-sized tokamak conceived, designed, and largely indigenously fabricated. It is commissioned and operational at the Institute for Plasma Research, Bhat, Gandhinagar. The chief scientific objectives of ADITYA are (a) investigation and control of edge phenomena for improving confinement properties; (b) investigation of density and current limits of a tokamak with special emphasis on interesting phenomena like MARFES, detached plasma, disruptive instabilities and their control, and (c) study of novel regimes of operation, e.g. H-mode obtained using bulk/localized heating by RF.

The device

The choice of machine parameters has been guided to a large extent by the desire to allow maximum flexibility to accommodate various needs of the proposed experiments as well as future experiments relevant to the physics of magnetically confined fusion plasma. A large number of sophisticated diagnostics have been deployed to meet the scientific objectives and one of the prime considerations was to provide ample access for these diagnostics and have a plasma with reasonable density, temperature and size for good confinement times. The preference was therefore towards a large volume low field design rather than towards a high field tokamak. The toroidal field at major radius was thus fixed at 1.5 T and the plasma minor radius chosen to be 25 cm. To optimize energy usage, a tight aspect ratio (major radius/minor radius) = 3 was chosen. This then leads to plasma currents ≈ 0.25 MA. The basic parameters of the plasma created in the device can be deduced with the help of available empirical scaling laws. Some of these estimated parameters are listed in Table 1 along with the basic machine parameters. Figure 4 shows the top view of ADITYA. Detailed technical description of various subsystems of ADITYA has been given by Bhatt *et al.*[5]

Table 1. ADITYA design parameters

Plasma major radius	R(m)	0.75
Plasma minor radius	a(m)	0.25
Toroidal field at plasma	B (tesla)	1.5
Safety factor	$q(a)$	2.5
Plasma current	I(MA)	0.25
Electron temperature	T(keV)	0.5
Ion temperature	T(keV)	0.2
Energy confinement time	t(ms)	≈ 5
Pulse duration time	t(s)	0 3

Figure 4. Top view of tokamak ADITYA.

The vacuum system of ADITYA includes one of the largest ultra high vacuum (UHV) vessel (capable of reaching an ultimate vacuum of better than 10^{-9} torr), designed and built in India. The vacuum vessel is a torus of major radius 75 cm with a square cross-section of side 60 cm. An inside view of a section of the vessel is given in Figure 5. The vacuum vessel is mainly pumped by two turbo-molecular pumps each having a pumping capacity of 2000 litres per second for air and backed by two rotary pumps of 60 cubic meters per hour pumping speed and two cryopumps, each having a pumping capacity of 10,000 liters/s for water vapour and condensible hydrocarbon. A view of one of the

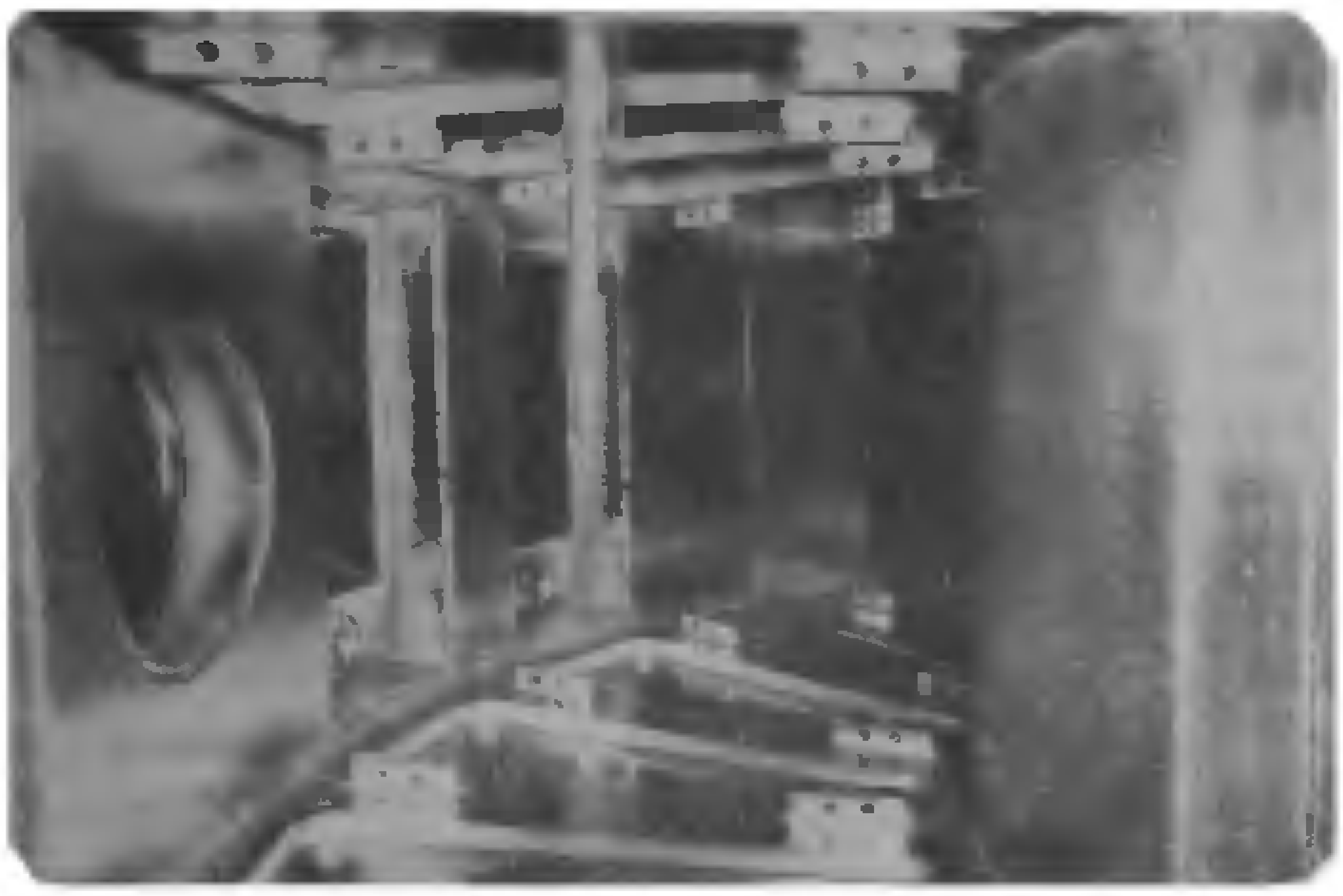

Figure 5. Inside view of a section of ADITYA vacuum vessel.

Figure 7. A limiter section with carbon tiles mounted on it.

pumping lines of the vacuum system is given in Figure 6. To ensure clean surface conditions of the vacuum vessel as required for reproducible plasma discharges, a large number of techniques, developed in-house, have been used. In ADITYA, a combination of glow discharge cleaning and low temperature pulsed discharge cleaning is used between the shots to maintain clean conditions. Based on the considerations of impurity generation and heat withstanding capability graphite has been chosen as the Limiter material for ADITYA which has two Safety Limiters, a Poloidal Limiter and four segmented Movable Limiters. Figure 7 shows a segment of the safety limiter of ADITYA.

Tokamak ADITYA employs three principal sets of magnetic field coils, the toroidal field (TF) coils, the ohmic transformer (TR) coils, and the vertical field (BV) coils. The TF coils produce the main toroidal field, the Ohmic transformer formed by TR coils produces the transformer flux required to produce the plasma and drive current through it, and the BV coils provide the vertical or equilibrium field that maintains plasma in equilibrium position during the course of a discharge. In addition, a set of feedback coils are used to control the plasma position. These large area, high field coils, have been designed and fabricated indigenously. Enormous electromagnetic forces appear on the magnetic field coils because of the large currents flowing through them and the plasma. The coils are therefore restrained by a carefully designed mechanical structure, which also provides an independent support for the vacuum vessel. Figure 8 gives a view of the magnetic field coils and the supporting structures.

Pulsed power is required to energize the magnetic field coils of ADITYA. Due to the high peak power and energy requirements of each of the coils per pulse, the overall demand for all loads exceeds a peak power level of 50 MW per pulse, drawing more than 60 MJ per pulse when run to full capacity. Two types of pulsed power systems are provided : (i) a triggered D.C. power supply cum capacitor based power system[6] for low parameter regime operations where plasma durations of about 25 ms, and peak plasma currents in the range of 25–60 kAmp can be obtained at a toroidal magnetic field of 0.25 tesla, and (ii) a controlled rectifiers cum

Figure 6. A view of ADITYA pumping system.

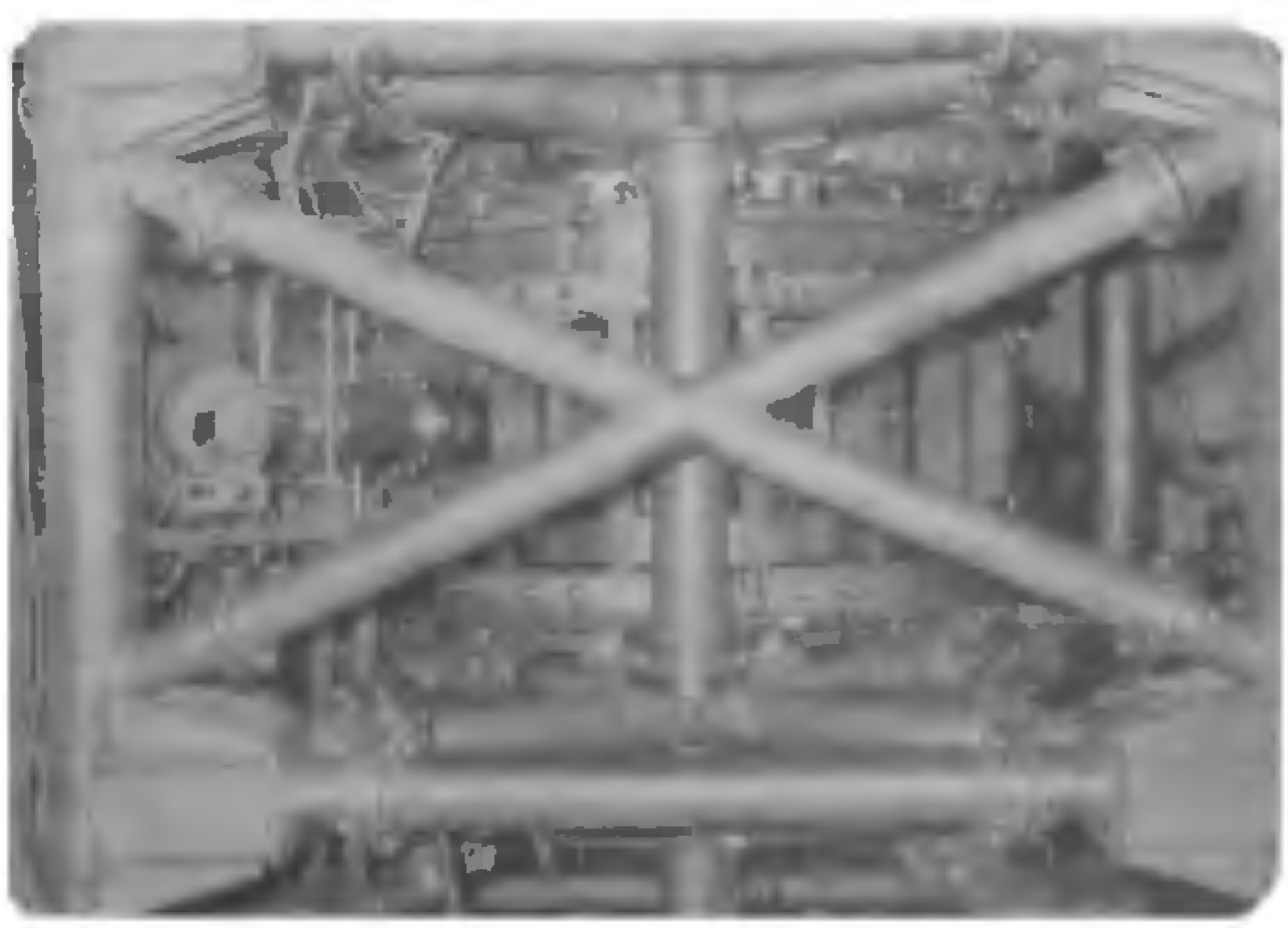

Figure 8. A view of coils and structure.

waveshaping circuits system drawing energy from the mains grid for full operational parameters.

Diagnostics and data acquisition

Since the purpose of Aditya is to provide fundamental information on plasma equilibrium, stability and confinement, a moderately high emphasis is placed on diagnostics of the plasma. The plasma performance is diagnosed by a number of passive and active probes. All these diagnostics are based on advanced physics and technology concepts. Various diagnostics deployed on ADITYA and parameters measured by each diagnostics are listed below.

A. Electromagnetic measurements:
   1. Rogowski coils for measurement of plasma current I_p.
   2. Voltage loops for measurement of loop voltage V_l.
   3. Magnetic probes for measurement of magnetic fluctuations $\tilde{B}$.
   4. Position probes for plasma position measurements.
B. Langmuir probe diagnostic for measurement of n_e, T_e, ϕ_e in edge region.
C. Visible detector for monitoring of impurity and ionization level.
D. μ-wave interferometry:
   1. Multichannel interferometer for measurement of density profile $n_e(r)$.
   2. Single channel interferometer for measurement of chord-averaged density n_e.
E. Laser-induced fluorescence for measurement of density and temperature fluctuations $\tilde{n}_e$, $\tilde{T}_e$.
F. UV and visible spectroscopy for measurement of impurity line intensities.
   1. Visible (VIS) spectrometer (1/2 m focal length, resolution 0.2 Å, 3000–7000 Å).
   2. Normal incidence monochromator (NIM) (1 m focal length, resolution 0.1 Å, 1100–3500 Å).
   3. Grazing incidence monochromator (GIM) (1/2 m focal length, resolution 0.3 Å, 50–700 Å).
G. X-ray tomography for estimation of T_e and monitoring of MHD activity.
H. Hard X-rays: Energetic electron flux.
I. Bolometer camera for estimation of radiation loss (2–5000 Å).
J. Charge exchange neutrals for measurement of ion temperature T_i.
K. Thomson scattering for measurement of electron temperature T_e.

Future developments include Faraday rotation measurements, pulse height analysis (PHA) of hard X-rays, crystal spectrometery, fibre-optics current probes, visible light tomography and electrocyclotron emission (ECE) spectroscopy. The diagnostics deployed and proposed to be deployed on ADITYA are very sophisticated in nature requiring very careful design and implementation. These diagnostics have been designed in-house and include sophisticated signal handling electronics and instrumentation. Figure 9 shows a view of the seven channel microwave interferometer diagnostics for measuring plasma density profile in ADITYA.

The entire experiment of plasma production, confinement and heating is a pulsed experiment lasting a few seconds only. In order to establish a systematic, reliable and safe operation for such experiment, complete automation of control, protection and data logging is essential. ADITYA, therefore, operates under the control of ADITYA data acquisition and control system (ADACS). Computer-automated measurements and control (CAMAC) concept has been adopted to meet requirements of reliability, expandability, interchangeability and different speeds of acquisition for different diagnostics. ADACS acquires real-time data

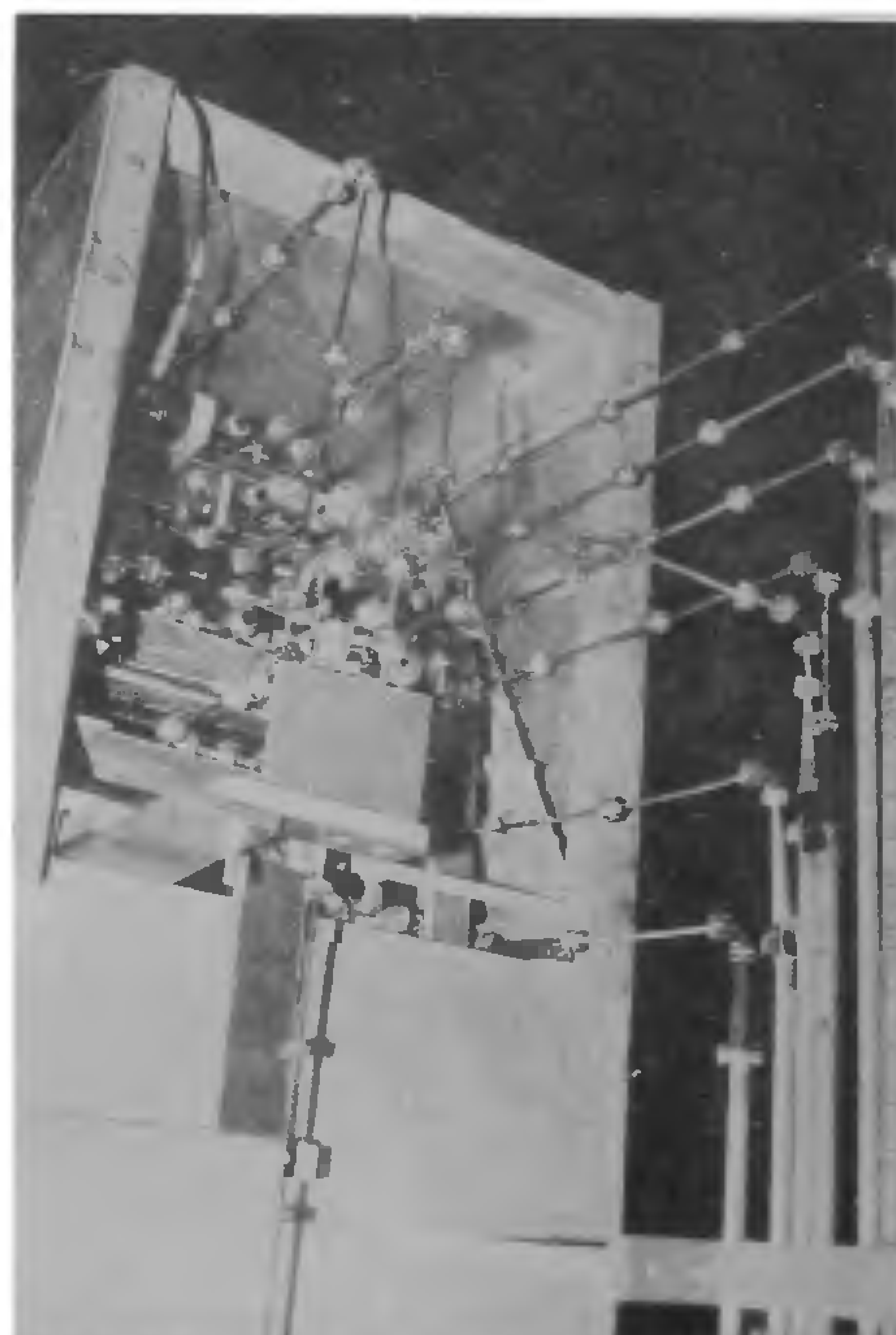

Figure 9. A view of seven channel microwave interferometer for measuring plasma density profile in ADITYA.

for various plasma and machine parameters, achieves the data along with other essential information and provides for retrieval, analysis and display of the discharge (shot) data on demand.

The design, fabrication, erection, and commissioning of ADITYA posed a major technological challenge to Indian scientists and the Indian industry. The successful completion of the project was achieved with the involvement of various agencies and organizations. The basic conceptual design was carried out by the IPR scientists and engineers in close collaboration with Tata Consulting Engineers, Bombay. Special advice was also provided by scientists from BARC, VSSC and SAC. The various subsystems were fabricated at IPR and at various industries including BHEL, Bhopal; Larsen & Toubro, Bombay; Godrej, Bombay; NGEF, Bangalore; and Hindustan Brown Bovery, Baroda.

Experimental results

Tokamak ADITYA was commissioned in September 1989 and several physics experiments have been carried out. At present, ADITYA is operated with DC power supply for TF coils and a capacitor bank for TR and BV coils[6]. After optimization of operating parameters like gas pressure, loop voltage, etc. and synchronization of the vertical magnetic field to the current rise, a plasma duration of 22 ms with a peak plasma current of 60 kA has been achieved. With appropriate shaping of the vertical field time profile, discharges lasting for 20–25 ms, with flat top currents in the range of 20–25 kA for durations of 10–15 ms have also been obtained. Initial results on conditions of gas breakdown, optimization of various discharge parameters and, measurements of average plasma density and temperature have been reported by Atrey *et al.*[7].

As mentioned earlier the study of the edge plasma forms one of the main objectives of ADITYA. We have carried out detailed study of the fluctuations in edge parameters. In the following we summarize some important results obtained from these experiments.

Characteristics of tokamak edge turbulence

Theoretical and experimental studies of turbulent edge region of tokamak plasma have received considerable attention in recent years because of a growing consensus that the edge turbulence affects the overall particle and energy confinement[8]. Detailed studies on the characteristics of the fluctuations of density and potential in the edge plasma of ADITYA have been carried out.

For these experiments, ADITYA is operated at the following parameters: toroidal field $B_T = 2.5$ kG; plasma current $I_p = 20$ kA; major radius $R = 75$ cm; minor radius $a = 25$ cm; average density $n_e = 5 \times 10^{12}$ cm^{-3} and electron temperature $T_e \approx 100$ eV. The plasma density and temperature as measured by Langmuir probes in the scrape-off-layer (SOL) are 5×10^{11} cm^{-3} and ≈ 15 eV respectively. The fluctuations in the plasma density and floating potential are measured using molybdenum Langmuir probes in SOL and rake probes in the edge region of main plasma. Figure 10 shows plasma current, loop voltage, ion saturation current and the SOL floating potential. The discharges typically last for 20–25 ms, out of which 10–15 ms is flat-top in plasma current.

The wavenumber (k) and frequency (ω) spectra, $S(k)$ and $S(\omega)$ respectively, for the density and potential fluctuations are obtained following the procedures involving estimation of wavenumber-frequency spectrum using fixed probe pair[9] and standard spectral analysis techniques. The spectra for density and potential fluctuations in the SOL plasma are shown in Figure 11 *a*. The $S(\omega)$ spectrum has nearly constant power at smaller frequencies and follows a power-law with index of ≈ -1.9 in the frequency range 20–125 kHz. The $S(k)$ spectrum also follows a power-law with an index -2 in the range $2 < k < 6$ cm^{-1}. Figure 11 *c* shows contour plots of $S(k, \omega)$ spectrum for the density fluctuation, illustrating the broadband nature of the spectrum. The spectral characteristics for the potential fluctuations are shown in Figure 11 *b*. The spectral index of $S(k)$ spectrum for potential fluctuation is ≈ -5. Both $S(k)$ and $S(\omega)$ spectra are broadband in nature similar to the ones for plasma density and indicate turbulence.

Fluctuation-induced particle transport

The spectral characteristics of fluctuation-induced particle transport in the scrape-off layer (SOL) region of

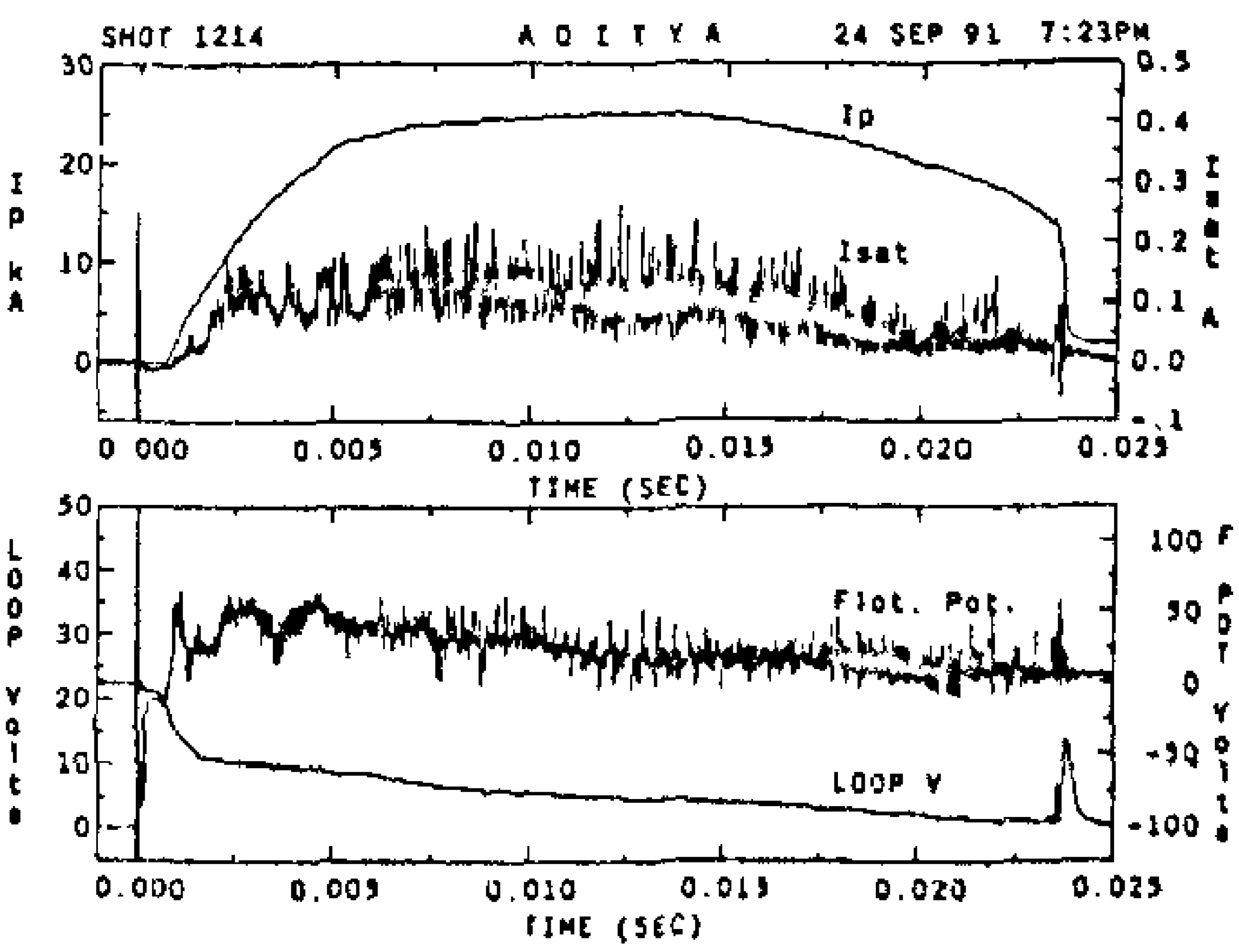

Figure 10. Plasma current, loop voltage, density and floating potential in the edge plasma.

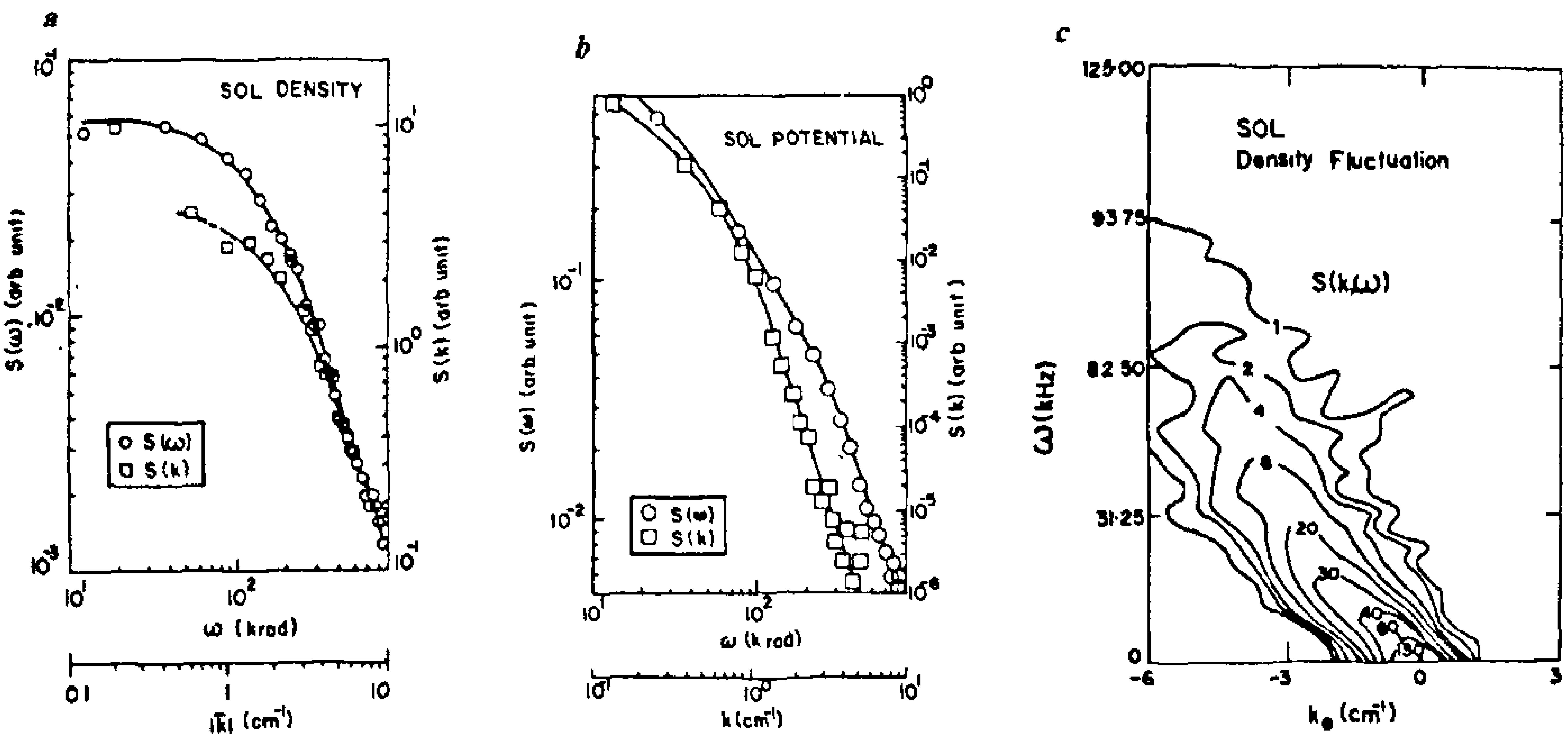

Figure 11. (*a*) $S(k)$ and $S(\omega)$ spectra for SOL density; (*b*) $S(k)$ and $S(\omega)$ spectra for SOL potential and; (*c*) Contours of $S(k,\omega)$ for SOL density.

ADITYA tokamak has been estimated from the observed fluctuations. For this purpose the density ($\tilde{n}_e = \delta n_e/n_e$) and potential ($\tilde{\phi} = \delta\phi$) are measured simultaneously using three Langmuir probes, arranged in Δ configuration, and located in the SOL plasma. Two of these probes, separated poloidally by 5 mm and at the same toroidal and radial locations, were used to monitor floating potential fluctuations (ϕ_1 and ϕ_2) while a third probe, located poloidally midway between the other two probes at the same radial location but shifted toroidally by ≈ 5 mm was used to monitor density fluctuations ($\tilde{n}_e$). The data are segmented into large number of ensembles (over 800), each containing 128 samples, and the ensemble-averaged power spectra and wavenumbers are obtained with a frequency resolution of 2 kHz. This enabled us to obtain frequency resolved particle transport with improved frequency resolution compared to previous measurements of particle flux[10].

The spectral particle flux is determined from the ensemble-averaged cross-power spectrum, $P_{n\phi}(\omega)$, and the phase difference $\alpha_{n\phi}(\omega)$, between the density and the potential fluctuations, and the wavenumber, $k_{\phi\phi}$ for the potential fluctuations. Usual procedures of ensemble averaging of the co-spectrum and the quad-spectrum, to obtain the averaged cross power and the wavenumber, have been followed. The estimated differential flux, $d\Gamma(\omega)/d\omega$, and the frequency integrated flux, $\Gamma(\omega)$, are shown in Figure 12. It is found that the $d\Gamma(\omega)/d\omega$ is inward at frequency less than 10 kHz and outward at higher frequency. Thus the overall transport due to fluctuations at frequencies ≤ 20 kHz is inward.

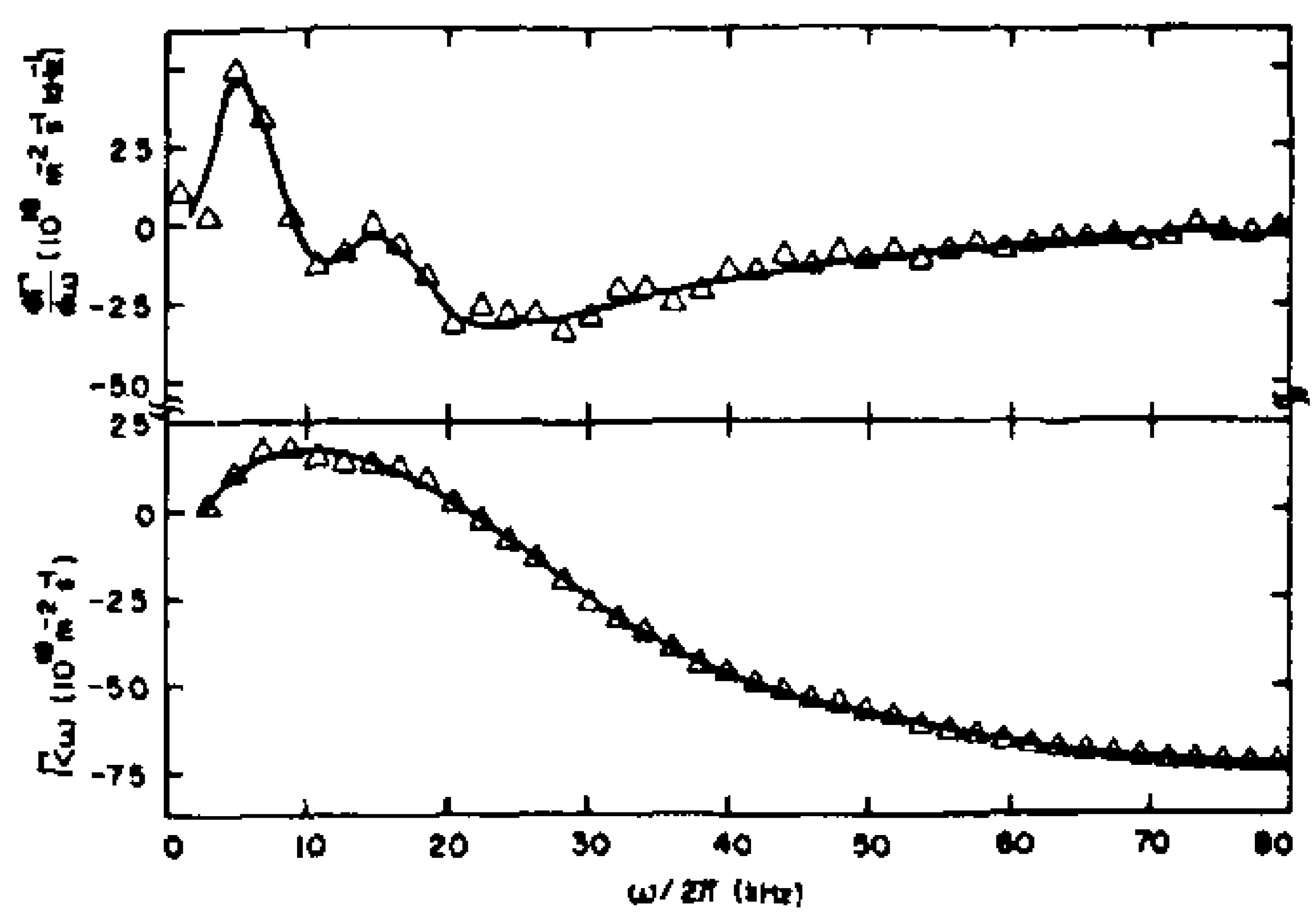

Figure 12. Spectrally resolved fluctuation-induced particle transport.

However, the net frequency-integrated flux, due to all the fluctuations together, is found to be outward. Further, we find that the phase $\alpha_{n\phi}(\omega)$ undergoes a change of sign[11] at $\omega/2\pi < 10$ kHz. The maximum negative phase at $\omega/2\pi \approx 10$ kHz is $-60°$, whereas positive phases at higher frequencies are in the range of $+60°$ to $+120°$. Such phase reversal appears to be the root-cause of the observed inward particle flux at lower frequencies.

Recent theoretical studies have shown that the drift wave fluctuations driven by ionization effects[12] are excellent candidates for generating inward particle transport. The wavelength regime for excitation of the ionization driven instability has been recently reexamined[13] and it has been shown that when one takes account of the perpendicular diffusion-induced damping of the excited modes, only long wavelength modes for which $k'_{\perp} L_{n} < 1$ can be driven unstable. For ADITYA parameters, this restricts the ionization instability to $m < 13$ and to $\omega/2\pi < 10$ kHz. Thus, the inward directed transport observed at low frequency and long wavelength end is consistent with the properties of the drift ionization instability. The shorter wavelength fluctuations ($m > 13$) must be ascribed to some other instability driven mechanisms associated with drift waves. Most of these instabilities lead to outward particle transport as observed in the experiments.

Intermittency in edge turbulence

In addition to the spectral analysis, a further analysis of the amplitudes of the fluctuations revealed a surprising phenomenon. The probability distribution functions (PDFs) of the amplitudes displayed marked non-Gaussian features indicating the presence of intermittency in the turbulent fluctuations[14]. Figure 13 shows the estimated PDF for the density and the potential fluctuations. The skewness and kurtosis of the PDFs are 0.9 and 6 respectively for the density fluctuations and, 0.5 and 5 for the potential fluctuations indicating strong deviation from a Gaussian. The density PDFs are asymmetric indicating preponderance of ion depletion regions. The asymmetry in the potential fluctuations may indicate stretching of streamlines. These non-Gaussian characteristics are seen to be similar in the scrape-off layer as well as in the plasma edge, a few centimetres inside the limiter. Our measurements also demonstrate that the intermittency evolves during the discharge, perhaps, due to evolution of critical parameters such as T_e, Z_{eff} even in the flat-top current phase.

These are the first ever measurements of intermittency in a tokamak edge plasma and can have far-reaching consequences in terms of our understanding of plasma turbulence and the nature of transport in tokamak plasmas. Intermittency implies the creation and destruction of spatially coherent structures over short periods. The short-lived coherent structures can arise through nonlinear processes, and then collapse due to secondary instability, leading to a new intermittent mechanism of energy dissipation which supplements direct coupling of unstable modes to the damped modes[15]. Another possible consequence of the intermittency is its influence on the dissipation mechanism of fluctuations and on the anomalous transport, both of which could acquire a 'bursty' behaviour. These and other exciting possibilities have triggered further experiments, not only on ADITYA, but on several other tokamaks in the world.

Current termination in low-q discharges

The disruption — a sudden termination of plasma current in a short span of time, typically few hundred microseconds — is known to limit the operating regime of a tokamak. With a view to understand the physics of disruption, we have carried out a systemetic study of the physics of current termination scenario in ADITYA[16].

A typical signature of current termination in tokamak discharges is a sharp positive spike in plasma current I_p, followed by a negative spike in loop voltage V_l. In ADITYA, for discharges with peak current values in the range of 30–60 kA, we have observed multiple disruption-like features (Figure 14). It is seen that these spikes are repeated 2–4 times each separated by about 1 ms. The delay in the positive current spikes and corresponding negative voltage spikes is 100–300 μs. The optical signals also show increase corresponding to

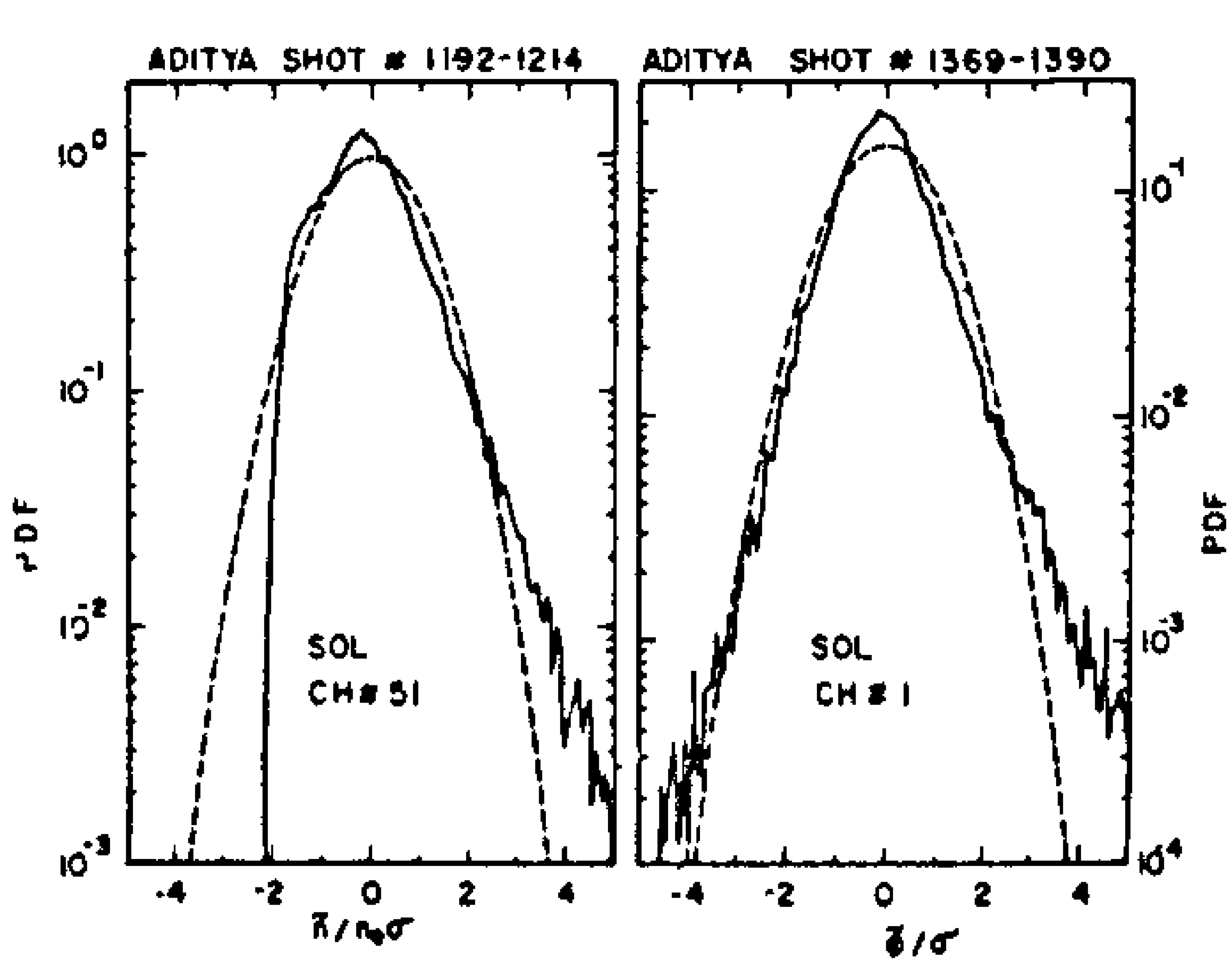

Figure 13. PDFs of SOL density and floating potential fluctuations.

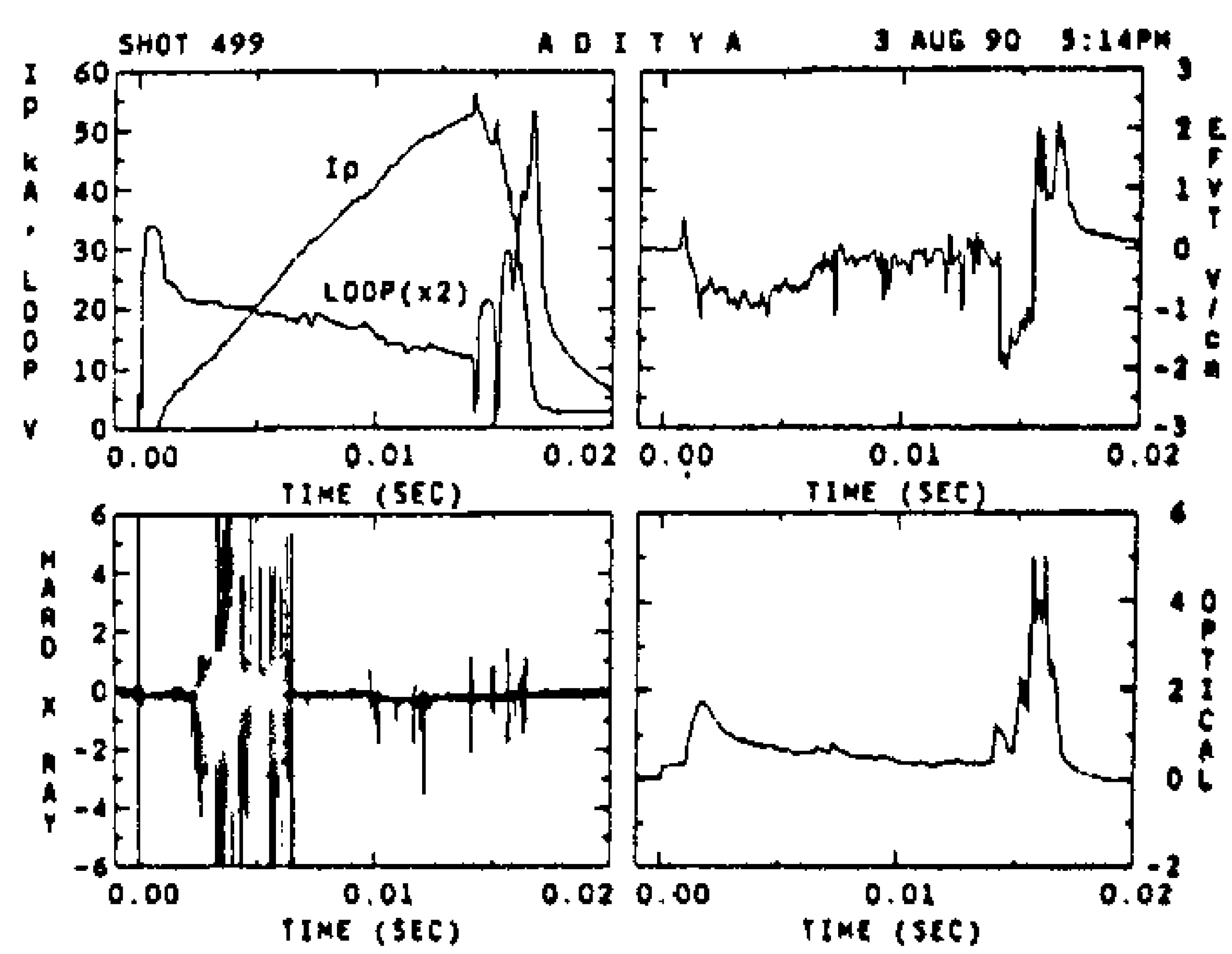

Figure 14. Current disruption characteristics.

these times. Hard X-ray bursts, synchronous with the current increase during these events are also observed. The sequence of events is repeated till the current channel crashes due to overcompensation of hoop force by the vertical magnetic field. The Langmuir probe signals show large vertical electric field across the plasma column, synchronous with the final current crash, indicating crashing of the current channel to the inboard side of the vacuum vessel. The appearance of the positive current spike is an indication of the flattening of the current profile. The subsequent appearance of the negative spike in loop voltage indicates the ejection of the poloidal flux. ADITYA current termination shows a sequence of profile flattening. The electrical field measurements indicate the movement of plasma to the outboard side of the vessel during this phase. The plasma resistivity increases as indicated by the rise in the average level of the loop voltage. The plasma current decreases until the external vertical field overcompensates the hoop force and causes the crashing of the current channel towards the inboard side of the vessel as indicated by the reversal and increase of the measured electric field.

A possible scenario for these multiple disruptions is as follows[16]: At plasma currents in the range of 30–50 kA the q-value at the limiter is in the range of 3–2. Hence in these discharges, the magnetic island width at $q=2$ surface grows and the island touches the limiter, leading to the loss of current and consequent redistribution of the current profile. The $q=2$ surface then moves in and the growth of island restarts. This sequence of events continues till the plasma current channel crashes due to the over-compensation of the hoop force by the external vertical field. Measurements using magnetic probes are in progress to verify the proposed scenerio.

Future experiments

In the next couple of years, a large number of experiments will be carried out on ADITYA to achieve the scientific objectives listed earlier. They can be broadly classified as edge plasma studies, disruption studies and RF heating experiments.

Edge plasma studies

Edge plasma studies are currently being pursued in order to gain an understanding of the driving mechanisms of fluctuations in the edge plasma and ultimately to suppress/control them through external means and achieve better confinement.

The observation of inward particle flux at low frequency ($\omega/2\pi < 20$ kHz) indicates that probably the fluctuations at low frequencies are caused by ionization driven drift-wave instabilities. On the other hand, fluctuations at higher frequencies may be caused by some other instability mechanisms which lead to an outward directed particle flux. A clear understanding of these separate instability mechanisms is necessary for they suggest a very interesting method of controlling particle confinement time by partial control of the long wavelength ionization instability. At present, detailed measurements of fluctuations in plasma density, temperature and potential are being carried out to delineate instability mechanisms in different regions of the edge plasma (e.g. SOL or inside the limiter) as well as in different ranges of frequencies and wavelengths.

Using fluctuation measurements feedback experiments to modify the edge plasma have been planned. ADITYA poloidal limiter has fourteen discrete carbon tiles, some of which will be used for the feedback modification of the edge region. This is proposed to be done by picking up fluctuation signals, filtering them appropriately, phase-shifting them and applying the same through suitable power amplifiers to the discrete tiles. Use of such methods has been found to reduce core fluctuations in mirror machine[17]. Recent experiments at IPR in a toroidal plasma device BETA have indicated interesting modifications in characteristics of the fluctuations using similar feedback method. This type of experiment has achieved additional significance in view of our observation of the inward directed transport due to low frequency fluctuations. If one could selectively suppress the higher frequency fluctuations and sustain low frequency component, a significant improvement of the particle confinement time could be achieved.

The observed intermittency in ADITYA edge plasma indicates the existence of shortlived spatial structures. Experiments are currently on to observe these short-lived coherent structures in the edge plasma turbulence. At present, fluctuation measurements are being made on a section (50 mm × 30 mm) of the poloidal cross-section in the edge plasma to determine these structures. Conditional averaging techniques[18] will be applied to determine the size and the lifetime of these structures. Another consequence of the intermittency effect is that the dissipation of plasma fluctuations and associated anomalous transport will also exhibit an intermittent behaviour. Wavelength dependence of the intermittency will be investigated together with the nature of the 'bursty' transport.

Disruption studies

Our preliminary study[16] indicates a certain scenario of current termination in which growth of MHD activity acts as a precursor. Experimental investigation of the role of MHD activity in causing disruption is being carried out at present. For this purpose, a poloidal array of magnetic probes has been installed on ADITYA which will allow determination of mode structure etc.

Recent experiments have shown that disruption has a longtime precursor (more than 10 times the energy confinement time) in the form of deterioration of particle confinement time[19]. A detailed study of this phenomenon is being carried out on ADITYA by measuring the short wavelength microturbulence, its spectral characteristics and its role as a precursor to both density limit and current limit disruptions.

RF heating experiments

Auxiliary heating is necessary to heat further the ohmically heated (OH) plasmas to achieve fusion temperature. Among various such heating schemes realized in tokamaks, radiofrequency (RF) heating methods around the ion cyclotron frequency appear to be quite attractive from the point of view of both physics and technology. Ion Bernstein wave (IBW) near the second harmonic of the ion cyclotron frequency has been effectively used to heat the OH plasmas in recent times[20].

Experiments have been designed to use IBW to heat the OH plasma in ADITYA. Ion heating at higher power is capable of providing plasma with improved particle and energy confinement times, reduced recycling of neutrals from the vessel wall, and suppression of low frequency fluctuations from the edge. Detailed comparative studies of IBW heated plasmas with the H-mode plasmas will be carried out.

Keeping in view the above experiments, a 200 kW RF heating system in the frequency range of 20–40 MHz is being fabricated. The RF power will be delivered to the antenna placed in the shadow of the limiter through a transmission line and an interface with matching impedance. The RF system has the capability to accommodate the conventional fast-wave heating scheme by changing the antenna.

Recent experiments using biased electrode have shown that a significant improvement in particle confinement time can be achieved by inducing radial current at the edge which results in a poloidal rotation[21]. An experiment is planned on ADITYA to achieve this H-mode like discharge by selective ion cyclotron resonance heating of minority species in the edge plasma. A stripline toroidal antenna will be used to launch an evanescent mode at the plasma edge. This wave will generate a radial current by preferentially accelerating the resonant minority ions and ejecting them on to unconfined orbits. Thus, a non-invasive technique of inducing poloidal rotation and H-mode like behaviour in high grade tokamak plasmas will be tested.

Future direction

As the first experimental device of the country, ADITYA has been a rich source of learning experience —both technological and scientific. With its first novel results on edge turbulence it has already made its scientific impact in the world fusion research programme. If the present experiments are any indication, there is good reason to hope that ADITYA will continue to provide a happy hunting ground for exploring new ideas and gleaning new insights into the behaviour of tokamak plasmas.

However, what of the future? In the long term, as we look at future fusion reactors, we have to look at long-pulse or steady state machines in which currents are maintained by RF waves, impurities are kept under control by the use of divertor or pumped limiter systems, magnetic fields are produced by liquid nitrogen cooled copper or superconducting coils, power systems are supplemented by rotating machines, fueling is done by pellet injectors, and power exhaust is controlled by gas flow near divertor plates and so on. This will lead us to the development of a host of new technologies which can be brought to a focus only by designing and fabricating our own next generation device. The investments required will be high but the challenges it offers and the capabilities it develops for our scientific and industrial infrastructure will be well worth it.

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