Chapter 2
Tokamaks and ADITYA Tokamak

This chapter introduces the basic physics of tokamaks and ADITYA tokamak with its subsystems and operational diagnostics.

2.1 Tokamaks

In thermonuclear fusion research using magnetic confinement, tokamak is the most promising candidate to demonstrate fusion as achievable energy source. A tokamak (Торoidalная Камера Магнитная Катушка or "Toroidal Vessel with Magnetic Coils") is a toroidal device which uses a strong toroidal magnetic field, \( B_\phi \) to confine high temperature plasma within the torus for a sufficiently long time. Final goal of the tokamak research is to reach fusion of deuterium and tritium nuclei for production of electricity.

In a tokamak toroidal magnetic field magnitude has gradient and curvature in \( -\hat{R} \) direction. Charged particles (electrons and ions) gyrate under the influence of these fields and feel drift (Fig. 2.1). Because of grad- \( B \) drift and curvature drift ions drift in \( \hat{z} \) direction and electrons drift in opposite direction. This charge separation produces an electric field perpendicular to toroidal magnetic field (in \( -\hat{z} \) direction). The force \( qE \) of electric field along with \( B_\phi \) results in outward (in \( \hat{R} \) direction) collective drift \( v_{ExB} = -E_z \times B_\phi / B^2 \) on charge particles and makes plasma unstable.

The outward drift tendency of charged particles in a toroidal magnetic field is compensated by adding a poloidal magnetic field (\( B_\theta \)) generated by toroidal plasma current (\( I_r \)) (Fig. 2.2). The resultant magnetic field lines (\( B = B_\phi + B_\theta \)) are twisted into helical shapes. Electrons flowing along field lines neutralize the charge-separation and instability originated from grad- \( B \) and curvature drifts are removed.
**Figure 2.1.** Opposite $\hat{z}$-direction drift in ions and electrons caused by grad-$B$ drift and curvature drift results a charge polarization, which induces an electric field. Both ions and electrons drift outward $(E \times B)$. Consequently, plasma is not confined in a simple toroidal magnetic field.

**Figure 2.2.** Helical structure of magnetic field: Plasma current $(I_P)$ generates a poloidal magnetic field $(B_\theta)$, which together with toroidal magnetic field $(B_\phi)$ produce helical magnetic field lines $B = B_\phi + B_\theta$. In helical field lines polarization is short circuited, as charged particles can move freely along the line of force.

In a tokamak a fast time varying current in Ohmic transformer (OT or TR) coils produce a time varying magnetic flux, which in turn produce a toroidal electric field (generally expressed in terms of loop voltage). Under the influence of electric field plasma is produced
inside the vacuum vessel. Plasma current ($I_p$) induced via this transformer action is understood as secondary circuit of transformer. Plasma is confined by toroidal magnetic field, which is produced by external electric currents flowing in poloidal coils (toroidal field (TF) coils) wound around the torus. Poloidal magnetic field caused by plasma current ($I_p$) and vertical magnetic field produced by a set of vertical magnetic field coil (Bv coils) is used to keep plasma in equilibrium. A detailed description of coil arrangement and plasma discharge is discussed in section 2.4.1.1 and 2.4.2.

2.1.1 Equilibrium of tokamak plasma

On short timescales tokamak plasmas show a variety of oscillations and turbulent phenomena. On sufficiently long timescales the plasma behaviour is governed by diffusive losses, gradual changes in the magnetic configuration and changes caused by plasma heating. The timescale in which the plasma pressure and the magnetic forces balance, tokamak plasma is understood to be in “equilibrium”. In ideal MHD, which treats the plasma as an ideally conducting fluid, Maxwell equations are subject to the low frequency and Ampere’s law shows no displacement current ($j = \nabla \times B$). Ideal MHD equilibrium satisfies the force balance:

$$\nabla p = j \times B$$  (2.1)

From which we have, $B.\nabla p = 0$, i.e., there is no pressure gradient along a field line; magnetic flux surfaces are surfaces of constant pressure. Also from eqn. 2.1, we have $j.\nabla p = 0$, so that current lines lie in a magnetic surface. Therefore, in equilibrium configurations, $B$ and $j$ lie on constant pressure surfaces, which if closed, appear as continuous windings of intersecting magnetic field and current lines; these are said to lie on ‘magnetic surfaces’.

Safety factor

As discussed in a tokamak, plasma is confined by a toroidal magnetic field lines. Plasma current generates poloidal magnetic field, which in combination with toroidal field lines forms helical field lines (Fig. 2.2) and keeps plasma in equilibrium. Field lines lay on nested surfaces centred on magnetic axis. These nested surfaces are termed as magnetic flux surface. In an axi-symmetric equilibrium each magnetic field line has different twisting, which is expressed by its ’$q$’ value and termed as ‘rotationl transform’ or ‘safety factor’.

General definition of safety factor can be expressed as the pitch of the field: $q = \frac{d\phi}{d\theta}$, where $\phi$ and $\theta$ are toroidal and poloidal angle, respectively. If after change of $\Delta \phi$ toroidal
angle a field line returns to its poloidal position (i.e., after $\Delta \theta = 2\pi$) at some poloidal plane, its $q$-value is defined as $q = \Delta \phi / 2\pi$. In other words, safety factor is defined as number of toroidal turns in one poloidal turn.

Field lines associated with rational flux surfaces with $q = m/n$, where $m$ and $n$ are integers, the field line joins up on itself after $m$ toroidal and $n$ poloidal rotations around the torus. Rational values of $q$ play an important role in stability. Effect of magnetic shear on mode stability and its localization at mode rational surface ($r = r_s$) is of great interest. Generally, higher value of $q$ leads to greater stability. In tokamaks, the radial profile of $q$ usually has its minimum value at, or close to, the magnetic axis and increases outwards. Its behaviour is determined by the toroidal current density profile $j_\phi(r)$.

### 2.1.2 Grad-Shafranov equation and the Shafranov shift

In studying tokamak equilibria it is convenient to introduce the stream function $\psi$ to reduce equilibrium equations to a single partial differential equation in one unknown. The stream function is proportional to the poloidal flux function $\psi = \psi_{\text{pol}} / 2\pi$ within each surface.

In cylindrical coordinates $(R, \varphi, z)$, an axi-symmetric toroid, $(\partial / \partial \varphi = 0)$, poloidal magnetic flux can be expressed in terms of the toroidal component of the vector potential $A_\varphi$ alone, i.e., $\psi(R, z) = RA_\varphi$. Assuming that there is no pressure gradient along a field line, the equation for pressure balance in equilibrium is given by $\nabla p = j \times B$. With these assumptions ideal magnetohydrodynamic equations reduce to an elliptical partial differential equation for the poloidal magnetic flux, also known as Grad-Shafranov equation [2.1, 2.2]

$$-\mu_0 R_j_\varphi = R \frac{\partial}{\partial R} \left( \frac{1}{R} \frac{\partial \psi}{\partial R} \right) + \frac{\partial^2 \psi}{\partial z^2}$$

Above equation is solved in order to reconstruct the magnetic flux surface inside an isotropic plasma.

#### The Shafranov shift

For low-$\beta$, large aspect ratio tokamak plasmas of nearly circular cross section when toroidal effects are included, it is seen that magnetic flux surfaces form non-concentric circles, i.e., the centers of the magnetic flux surfaces are displaced with respect to the center of the bounding surface (Fig. 2.3). This displacement is called the Shafranov shift ($\Delta_s$). The origin
of Shafranov shift can be easily understood. Current is bent in axi-symmetric tokamak plasma. So, average value of the major radius is smaller on the inside surface than the outside surface. A given amount of poloidal flux $\psi$ on the outside of the torus must be squeezed into a smaller cross sectional area on the inside (small $r$). Therefore, poloidal magnetic field at large $r$ is smaller. Magnitude of the magnetic field is greater on the in-board side than on the out-board side and the lines are packed more closely together on the inside [2.3].

Figure 2.3. Circular flux surface is displaced by a distance ($\Delta_s$) with respect to boundary flux surface whose centre is at a distance $R_0$ from the major axis. The outwards displacement ($\Delta_s$) is known as the ‘Shafranov shift’.

Since an excessive shift will inefficiently lose the plasma either to the limiter or to the wall of the vacuum vessel, it is customary in tokamak devices to control the shift with an externally provided vertical magnetic field. The necessary vertical field to maintain the tokamak plasma in equilibrium, is given by [2.4]

$$B_v = -\frac{\mu_0 I_p}{4\pi R_0} \left( \ln \frac{8R_0}{a} + \Lambda - \frac{1}{2} \right),$$

(2.3)

where $I_p = 2\pi a B_{0\theta}(a)/\mu_0$ is the plasma current and $
\Lambda = \beta_p + \frac{l_i}{2} - 1$.

$\beta_p$, the poloidal beta, is given by $\beta_p = \frac{8\pi \langle p \rangle}{B_{0\theta}^2}$. 

26
Chapter 2: Tokamaks and ADITYA Tokamak

The interaction of the toroidal plasma current with vertical magnetic field, $B_V$, pushes the plasma inwards in major radius to balance the outward forces. Necessary condition for vertical magnetic field to keep plasma in equilibrium is discussed in section 2.4.1.1.

2.1.3 Tokamak confinement

As described in the section 2.1.1, in a tokamak plasma is confined in the nested magnetic surfaces. In the zeroth order approximation, while moving along the nested surfaces, trajectories of particles on the inner surfaces do not mix up with those on the outer surfaces. In the central region high temperature plasma is confined and separated from the cold plasma at the edge. Plasma parameters (temperature and density) gradually decrease from centre to the edge. But this inhomogeneity in the plasma brings main problem of plasma confinement. In the first approximation, plasma energy and density is lost due to temperature gradient, density gradient, electric fields and gradients in magnetic field. Particles migrate across $B$ to the walls along the gradients.

In straight cylindrical plasma diffusion understood in terms of Coulomb collisions is termed as ‘classical transport’. In classical transport particles suffer collisions with a collision frequency $\nu$ and collision allows the particle to step length equal to the Larmour radius, $\rho$. This gives a diffusion coefficient $D_\perp \sim \nu \rho^2$ and hence diffusion coefficient across the magnetic field scaled as $1/B^2$. To keep the distinction clear, the collisional transport in a toroidal geometry is termed as ‘neoclassical transport’. In toroidal plasma, transport fluxes enhance above the straight cylinder geometry because of internal convective flows and loss of poloidal symmetry because the toroidal field is stronger on the inside than on the outside of the torus.

Based upon the collisional transport tokamak plasma can be differentiate in three regimes: (1) banana regime, (2) Pfirsch-Schluter regime and (3) plateau regime.

In collisionless regime trapped particles exist in the weaker region of the tokamak magnetic field due to inhomogeneity of magnetic field and average time between collisions is long compared to the time required for a particle to complete a banana orbit and plasma is said to be in a banana regime. Thus in banana regime effective collision frequency is smaller than the bounce frequency: $\nu < \varepsilon^{3/2} v_{th}/Rq$.

If the time between collisions is less than the time required for a particle to complete an untrapped orbit, then the form of the trapped orbit cannot be relevant to the diffusion process and the plasma is in collisional regime: $\nu > \varepsilon^{3/2} v_{th}/Rq$. In the collisional regime, particles
move along the field lines between collisions. The transport in highly collisional fluid regime is called Pfirsch-Schluter transport.

The regime of intermediate collisionality, connecting low collisional banana regime to the highly collisional Pfirsch-Schluter regime is called plateau regime ($e^{3/2}v_{th}/R_q < \nu < v_{th}/R_q$). In this regime diffusion coefficient has no dependence on the collision frequency.

The measured values of transport across the magnetic field in tokamaks often exceed the predicted values calculated using the classical or the neo-classical theories. These enhanced losses are termed as anomalous transport. The anomalous transport in plasma is believed to arise from turbulent diffusion caused by electrostatic or electromagnetic fluctuations caused by micro-instabilities.

Confinement of plasmas is among the most important subjects in fusion research. To achieve thermonuclear conditions the plasma in a tokamak has to be confined for sufficient time. The energy confinement time is defined as the ratio of plasma energy content to the plasma energy loss rate. The energy confinement time is usually defined via the energy balance equation as

$$\frac{dW_p}{dt} = P_{in} - \frac{W_p}{\tau_E},$$

where $W_p$ is the total plasma energy and $P_{in}$ is the power supplied to the plasma.

Experiments have revealed that the energy confinement time turns out to be strongly dependent on the plasma temperature. High temperature plasmas are generated by use of the intense heating. The confinement behaviour can be conveniently put into three categories. The first covers Ohmically heated plasmas and the other two relate to plasmas with additional heating. These two basic modes of auxiliary heated plasma confinement are the so-called L (for low) and H (for high) confinement regimes. The scaling laws for Ohmically heated plasmas and those for externally heated plasmas are summarized in Table 2.1 and 2.2.

### 2.1.4 Heating

Plasma heating is essential problem in controlled fusion. In tokamak plasma the energy losses are balanced by the plasma heating. Fusion reaction rate is strong function of temperature and is negligible at low temperatures. Thus, to reach the temperature required for ignition it is necessary to provide some form of heating. Neutral beam injection (NBI) and radio-frequency heating (RFH) are commonly used for additional heating task.
Neutral injection heating

The concept of neutral injection as a heating technique is the most successful to date. In this technique high energy neutrals are injected into magnetically confined plasma as they are unaffected by the confining fields. These neutrals are converted into ions inside the plasma. The fast ions that result are then slowed down by Coulomb collisions, transferring most of their energy to the plasma particles if they are confined long enough, causing heating of both electrons and ions. Since heating the center of the plasma is desirable, injection of neutrals normal to the field lines is generally preferred, distance of penetration of a neutral beam into plasma before “ionization” increases with the energy of the beam particles.
Radio frequency heating

Since the collisional process scales as $T_e^{-3/2}$, plasma heating by the direct collisional mechanism saturates with temperature and becomes weak in high temperature plasmas. Electromagnetic waves in plasmas are subject to resonant absorption which is collisionless process and produces strong heating. Radio frequency heating transfers energy from an external source to the plasma by means of electromagnetic waves. When an electromagnetic wave propagates through plasma the electric field of the wave accelerates the plasma ions and electrons which then heat the plasma through collisions. Because of the nonuniform magnetic field and density in most plasmas, the non-collisional absorption of energy of electromagnetic waves in a magnetized plasma is allowed at variety of resonance frequencies by many different radio frequency heating schemes. However, all these schemes consist of the same general layout, namely, an efficient high power generator remote from the plasma, a low-loss transmission line and an efficient antenna which couples the electromagnetic energy to the plasma.

Radio frequency heating has been used to heat magnetically confined plasmas since the early days of fusion research. Three schemes have emerged as the most successful, namely, ion cyclotron, lower hybrid and electron cyclotron heating.

Ion Cyclotron Resonance Heating ($\nu \sim 50MHz$)

In a tokamak, the toroidal field falls off with major radius as $R^{-1}$ and the ion cyclotron resonance is localized at a value of $R$ for which the wave frequency, $\omega$, satisfies $\omega = \omega_{ci}(R)$. Ion cyclotron heating in tokamaks is accomplished by launching a fast magnetosonic (compressional Alfven) wave to transport energy from the antenna to the absorption region of plasma, where $\omega_{pe}^2 / \Omega_e^2 \leq 1$ and $\omega_{pi}^2 / \Omega_i^2 \geq 1$.

Lower Hybrid Resonance Heating (LHRH) ($\nu \sim 1-8GHz$)

In dense plasmas of interest, $\omega_{pi} >> \omega_{ci}$, and the lower hybrid resonance frequency is then given to a good approximation by $\omega_{LH} = \omega_{pi} / \sqrt{\left(1 + \omega_{pe}^2 / \omega_{ce}^2\right)}$, which lies typically in the range about 1-8GHz. The use of lower hybrid waves was originally proposed with the object of heating the ions. For this purpose it is essential to choose conditions such that the lower hybrid resonance occurs in the plasma. The condition for this is
Chapter 2: Tokamaks and ADITYA Tokamak

\[ n_{res} = \frac{2.3 \times 10^9 A_i f^2}{(1 - 2.3 A_i f^2 / B_0^2)} m^{-3} \text{ (} f \text{ is in GHz),} \tag{2.5} \]

where \( n_{res} \) is the electron density at the lower hybrid resonance, \( A_i \) is the atomic mass number of the ions, \( f \) is the frequency, and \( B_0 \) is the magnetic field.

**Electron Cyclotron Resonance Heating (ECRH)** \((\nu \sim 50 - 200 \text{GHz})\)

Since \( \Omega_e \geq \omega_{pe} >> \omega_{pi} \), only the electrons can respond to waves in the 50 GHz to 200 GHz ECRH frequency range, but ion heating can result from collisional energy transfer from the heated electrons. Electron cyclotron heating has been made possible by the invention of the gyrotron millimetre wave source. When the source frequency is comparable to the electron gyro-frequency or its harmonics, then cyclotron damping by electrons can take place when the resonance condition is satisfied. Because of the variation in the tokamak magnetic field with major radius, \( R \), this limits the spatial extent of the resonance region. Because extremely high frequencies employed, electromagnetic waves propagate freely into the plasma until quite high densities are reached and power can fed simply and efficiently to the plasma from wave-guides.

**2.1.5 Instabilities**

MHD instabilities and low-frequency micro-instabilities of magnetically confined plasmas are two important areas of modern plasma physics research. Tokamaks are subject to a variety of macroscopic instabilities. These strong instabilities are mainly MHD modes and arise from (1) current gradients and (2) pressure gradients in combination with adverse magnetic field curvature. The resulting instabilities are divided into two categories: (i) **Ideal modes**- Instabilities which would occur even if the plasma were perfectly conducting, and (ii) **Resistive modes**- which are dependent on the finite resistivity of the plasma.

Instabilities in which wavelength of the fluctuations is comparable to the ion Larmor radius are termed as micro-instabilities. Fine scale plasma turbulence can be explained by micro-instabilities. Therefore, these are important for understanding plasma turbulence.

In confined plasma there are processes by means of which the plasma evolves towards a state of lower free energy. In this process free energy is converted into kinetic energy to drive instabilities. The generalized energy conservation relation from the linearized MHD equations is given by
Chapter 2: Tokamaks and ADITYA Tokamak

\[
\frac{\partial}{\partial t} \left( \frac{1}{2} \rho_o V.V + \frac{B.B}{8\pi} + \frac{1}{2} \frac{J_o}{c} \cdot (\xi \times B) \right) + \frac{1}{2} \frac{p_o}{\rho_o} (\nabla \cdot \xi)^2 + \frac{1}{2} (\xi \cdot \nabla p_o)(\nabla \cdot \xi) \right) d^3 r
\]

\[
+ \int \left( \frac{c}{4\pi} E \times B + pV \right) dS = 0
\]

(2.6)

for a plasma in which the pressure is isotropic and bounded by a rigid, perfectly conducting wall, where the boundary conditions are:

\[
\mathbf{n} \cdot V = 0 \quad \mathbf{n} \times E = 0 \quad \mathbf{n} \cdot \frac{\partial \mathbf{B}}{\partial t} = 0 \, ,
\]

(2.7)

where \( \mathbf{n} \) is the unit normal to the boundary, \( V \) is fluid velocity, \( \xi \) is linear displacement vector defined by \( d\xi = V dt \), \( dS \equiv \mathbf{n} S \) and \( S \) is the surface bounding the volume of integration.

Under boundary conditions (2.7) surface integral vanishes and eqn 2.6 reduces to

\[
\frac{\partial}{\partial t} (K + \delta W) = 0 \, ,
\]

(2.8)

where \( K = \int \frac{1}{2} \rho_o V.V dV \) is the total kinetic energy of the plasma. \( \delta W \) is identified as the potential energy of the perturbation and given by

\[
\delta W = \int \left( \frac{B.B}{8\pi} + \frac{1}{2} \frac{J_o}{c} \cdot (\xi \times B) \right) + \frac{1}{2} \frac{p_o}{\rho_o} (\nabla \cdot \xi)^2 + \frac{1}{2} (\xi \cdot \nabla p_o)(\nabla \cdot \xi) \right) d^3 r
\]

(2.9)

A number of conclusions can be drawn from equation for the conservation of energy (eqn 2.8). Since we have assumed that there is no flow without perturbation in equilibrium, it follows that \( K \) is always positive definite. From the definition of instability \( K \) increases unboundedly in time. If \( \delta W > 0 \), system must be absolutely stable. In order to have instability, \( \delta W < 0 \) such that \( |\delta W| \) grows in time so as to balance exactly the increase in \( K \).

Equation 2.9 shows that the energy required to perturb the equilibrium magnetic field (either field line bending or compression) is positive definite and therefore stabilizing. The third term is also positive definite so that incompressible perturbations may be expected to be the most unstable. The second and fourth terms are potentially destabilizing ones. The driving mechanism in the second term is due to the equilibrium current and in the fourth term is due to the equilibrium pressure gradient.

With some algebraic manipulations equation 2.9 can be derived into following form

\[
\delta W = \frac{1}{2} \int d^3 r \left( \frac{1}{4\pi} \left| \mathbf{B} \right|^2 + 4\pi \frac{B_\parallel}{4\pi} - \frac{B_\theta \xi \cdot \nabla p_o}{B_\theta^2} \right)^2 + \frac{1}{2} \frac{p_o}{\rho_o} (\nabla \cdot \xi)^2 + \frac{J_o \cdot B_\theta}{|B_\theta|^2} B_\theta \times \xi \cdot B - 2\xi \cdot \nabla p_o \xi \cdot \mathbf{k}
\]

(2.10)
where \( \perp \) and \( \parallel \) denote components of \( B \) perpendicular and parallel to \( B_\theta \), \( \kappa \) is the normal field line curvature, \( \kappa = \hat{\mathbf{B}} \cdot \nabla \hat{\mathbf{B}} \), where \( \hat{\mathbf{B}} \) is a normal vector parallel to \( B_\theta \). In equation (2.10), first three terms are always positive and are the energies associated with the shear Alfvén, magneto-sonic and sound waves. The fourth term is the energy – driving kink modes and is created by the presence of the plasma current. This term can be negative. The fifth term is the energy – driving interchange modes. It can be negative if the curvature is in the same direction as the pressure gradient. If the pressure gradient is anti-parallel to \( \kappa \), the last term is always positive.

From the above discussion it is quite clear that the instabilities in tokamak plasma are mainly attributes of equilibrium toroidal current and equilibrium pressure gradient. At high \( \beta \) they start to interact but at low \( \beta \) they can be studied separately.

**Instabilities due to magnetic energy of plasma current**

Gradient in toroidal current drives instabilities in tokamak plasmas. These instabilities grow at rational surfaces where helicity of the perturbation matches with that of the magnetic field. These MHD instabilities may be ideal or resistive. Kink instability is a strong ideal MHD instability. Kink instability with low mode number is driven by radial gradient of the toroidal current density. It is so named because it leads to the kinking of the magnetic surfaces and the plasma boundary. Tearing modes are also driven by radial gradient of the equilibrium toroidal current density. These modes grow at rational surfaces due to tearing and rejoining of magnetic field lines as consequence of finite resistivity.

**Instabilities due to pressure gradient**

High-\( n \) modes tend to be strongly stabilized by the energy required to perturb the magnetic field. A necessary condition for the stability for a large aspect ratio tokamak of circular cross-section was given by Mercier criterion

\[
\frac{rB_\phi^2}{8\mu_0} \left( \frac{q'}{q} \right)^2 > (-p')(1 - q^2)
\]  

(2.11)

The term \( p'q'^2 \) represents the stabilizing contribution of the average curvature of the toroidal magnetic field. For \( q > 1 \) this is sufficiently large that the resultant curvature is good and a negative pressure gradient is stabilizing.
Chapter 2: Tokamaks and ADITYA Tokamak

In the usual case with \( (dp/dr) < 0 \), both the pressure gradient and shear term are stabilizing if \( q(r) > 1 \). For toroidal current density distributions that peak at the centre of the plasma and decrease with minor radius, \( r \), the safety factor, \( q(r) \), increases with \( r \). Thus, \( q(0) > 1 \) is sufficient to ensure the stability of localized, high-\( m \) interchange modes, as long as \( (dp/dr) < 0 \).

At sufficiently large values of the plasma pressure, the pressure gradient may become large enough in regions of bad curvature in a tokamak to produce “ballooning” instability. The deformation is flutelike and is larger on the outside of the torus. This deformation bends the field lines, which provides a restoring force. If the driving force, which is proportional to the product of the pressure gradient and the curvature (inverse major radius), is greater than the restoring force due to the resistance of the field lines to bending, then these ballooning modes will limit the achievable plasma pressure.

**Non-linear instabilities**

These instabilities are not fully understood but they can be attributed to identifiable MHD modes. The three major types of non-linear activities related to the MHD perturbations are Mirnov oscillations, Sawtooth oscillations and Disruptions.

**Mirnov Oscillations**

There are small magnetic fluctuations, which can be detected by placing magnetic pick-up loops just outside the plasma. These magnetic oscillations are observed as the island passes under the pick-up loop at the drift frequency (typically 1-20kHz). These small oscillating helical perturbations in the poloidal magnetic field can be separated into their Fourier components around the plasma in poloidal direction.

**Sawtooth oscillations**

Observations of x-rays emitted from the centre of the plasma often reveal oscillations, having a period of the order of the millisecond. The emission increases slowly through most of this period and falls back rapidly. X-rays observed from the outer part of the plasma have the inverse profile, a slow decay being followed by a rapid rise. The interpretation of these oscillations is that there is a tendency of tokamak plasma to a unstable concentration of current in the central, hotter region. As a result of this the axial current density \( j_0 \) rises. Thus for sufficiently large \( j_0 \), \( q_0 \) falls below unity and a \( q = 1 \) surface appears. The presence of
Chapter 2: Tokamaks and ADITYA Tokamak

$q = 1$ surface now allows $m = 1$ instability to occur, producing a magnetic island. The value of $q$ on this island is larger than unity.

**Disruption**

The most serious problem caused by magnetic islands is the major disruption. Disruption is sudden and complete loss of plasma confinement and a collapse of the plasma current, which result in large electromagnetic and pressure forces in the surrounding structure. These generally occur at the sufficiently high plasma density or high plasma current.

One mechanism for the disruption is nonlinear coupling of two islands which generate a large stochastic magnetic region within the plasma volume destroying the plasma confinement. Magnetic islands, when uncoupled or only loosely coupled to each other, grow on a time scale of milliseconds. This growth rate is too slow to explain the sudden nature of the hard disruption which happens on a microsecond time scale. If, however, two magnetic islands of different helicity were to grow and overlap, the resulting region engulfed by the islands may explosively grow into a stochastic magnetic region of poor confinement, and if the initial islands were large enough, a large percentage of the plasma confinement would be destroyed thus terminating the discharge.

**2.2 Resistive tearing instability**

Plasma instabilities may be divided into two general categories: magneto-hydrodynamic (MHD) and kinetic instabilities. The former may pose the most stringent limitations to plasma performance and detrimental effects on plasma confinement, and are described by a magneto-hydrodynamic model of the plasma. Instability drive may arise predominantly from plasma current or pressure gradients.

The poloidal field magnetic energy excites the helical instabilities owing to the helical perturbations. Perturbations may be decomposed into different modes. A growing perturbation increases the internal energy of the plasma. If all of them are stable in the plasma, they will attenuate and the perturbation will vanish. On the contrary, if some modes are unstable, they will grow in time. Perturbation having a form $\exp(i m \theta - i n \phi)$, is periodic in $\theta$ and $\phi$ angles and represents a sum of harmonics, where $m$ and $n$ are integers. This perturbation is called the $m/n$ mode. $m$ and $n$ specify the poloidal and toroidal mode numbers, respectively. As discussed in Section 2.1.1, in a tokamak device, along different flux surfaces, the ratio between the poloidal and toroidal magnetic fields changes. Each field
Chapter 2: Tokamaks and ADITYA Tokamak

line is defined as function \( q = m/n \). In general modes of greatest interest and importance are those whose perturbation structure lies along the direction of the magnetic field lines somewhere inside the plasma. Around these magnetic surfaces the pitch of the perturbation matches to that of the magnetic field. They are then said to be resonant at that point, and if they possess a field component perpendicular to the equilibrium magnetic surfaces, they will cause the field topology to change and thereby the formation of magnetic islands.

2.2.1 Ideal and resistive instabilities

Plasma instabilities that occur when resistivity (\( \eta \)) is null are called ideal instabilities. In ideal MHD theory unstable modes are expected to be evolve in a very fast Alfven time scale

\[
\tau_A = \sqrt{\frac{\mu_0 \rho a}{B}}
\]

where \( \rho \) is the plasma density, \( B \) is the magnitude of the magnetic field, and \( \mu_0 \) the magnetic permeability of vacuum.

In an ideal plasma with zero electrical resistivity, taking into account Ohm’s law \( \mathbf{E} + \mathbf{v} \times \mathbf{B} = 0 \),

the evolution of the magnetic field is governed by the following equation:

\[
\frac{\partial \mathbf{B}}{\partial t} = \nabla \times (\mathbf{v} \times \mathbf{B}),
\]

(2.13)

where \( \mathbf{v} \) is the plasma flow, \( \mathbf{B} \) is the magnetic field, \( \mathbf{E} \) is the electric field. Eqn. 2.13 equation describes the convection of the magnetic field with the plasma flow. The significance of above equation is that in the absence of resistivity, magnetic flux is \textit{frozen} to the plasma fluid. This frozen flux relation implies that magnetic flux through an arbitrary surface that moves with the plasma fluid is constant, that is, the topology of the magnetic surfaces does not change and island formation is inhibited.

In a resistive-MHD plasma, taking into account Ohm’s law

\[
\mathbf{E} + \mathbf{v} \times \mathbf{B} = \eta \mathbf{j},
\]

(2.14)

the time variation of the magnetic field is given by

\[
\frac{\partial \mathbf{B}}{\partial t} = \nabla \times (\mathbf{v} \times \mathbf{B}) + \frac{\eta}{\mu_0} \nabla^2 \mathbf{B},
\]

(2.15)

The first term in the RHS describes the convection of the magnetic field with the plasma flow, i.e., \textit{frozen in} condition of magnetic field lines with the fluid. The second term (known as diffusive or inertial term) in the RHS is the resistive diffusion of the field across the plasma, i.e., resistivity allows the plasma to diffuse relative to the magnetic field lines. The
rate at which the magnetic field lines diffuse through the plasma is proportional to the plasma resistivity, gives rise to another time scale defined by \( \tau_r = \mu_0 a^2 / \eta \) (resistive diffusion time).

Validity of the ideal MHD approximation is expressed in terms of the magnetic Reynolds number or *Lundquist number* (S), which is the ratio of Alfven transit time and resistive diffusion time. Ideal instabilities grow in Alfven transit time \( (A\tau) \) typically measured in micro-seconds for tokamaks, whereas, resistive diffusion time \( (r\tau) \) in seconds. Resistive tearing instabilities grow on time-scales that are intermediate between the very short MHD time-scale, \( A\tau \), and the very long resistive time-scale, \( r\tau \). These modes grow faster than the resistive diffusion time scale \( (r\tau = \mu_0 a^2 / \eta) \) but slower than the Alfven time scale.

### 2.2.2 Tearing modes

If Reynolds number (S) is small then the diffusive (inertial) term becomes important and the magnetic field lines can move through the plasma and even tear and reconnect to form a completely new topology. Because the magnetic Reynolds number is very high for normal tokamak operation, one could say that the motion conforms to ideal MHD. There are certain thin layers (*tearing layer*) within a tokamak plasma that have very short system length scales and a small S value, which means that resistivity is important to consider and that magnetic reconnection can become favourable. Because of this, these regions can become susceptible to the growth of a type of resistive plasma instability called a neoclassical tearing mode (NTM).

The non-zero \( \eta \) in the resistive layer allows magnetic field lines to "tear" or break and reconnect across a closed magnetic field line at rational surface which supports a resonant perturbation, via a finite value of \( B \), to form a new and different magnetic configuration, called magnetic islands. A magnetic island is characterized by its helicity number and its width. The helicity numbers, which are the \( m \) and \( n \) numbers indicate the island location and with what magnetic surface it interacts with. The relationship between the \( m \) and \( n \) numbers is \( nq - m = 0 \).

Radial magnetic field perturbation is needed to produce magnetic islands or to change it in time. The width of the island is defined as the maximum radial width across the island region in the poloidal plane. It is approximately

\[
w = \frac{r_s L_s B}{m B_z}, \tag{2.16}
\]
where \( r_s \) is the radius of the resonant surface, \( \tilde{B}_r \) is the perturbed radial magnetic field, and 
\[
L_s = q^2 R_0 / q r_s,
\]
both evaluated at the rational surface of the mode.

The major effect of magnetic islands is enhanced transport. Since the islands have a width, they provide a transport short circuit by the destruction of closed magnetic surfaces. Magnetic islands cause the flattening of the temperature and the density profiles in the plasma interior. When islands of different helicity overlap a stochastic magnetic field region is produced. This results in a region with no confinement since the particles will traverse this region on a time scale of the thermal velocities, rather than the diffusion time scale. During the tokamak discharge, there occur sawtooth oscillations and Mirnov oscillations and disruptions, which are understood to be due to tearing modes.

### 2.2.3 Growth of tearing modes

The linear stability analysis of the tearing mode was originally given by Furth et al [2.10], which is based on the boundary layer theory around the rational surface. The growth rate of the linear tearing mode is obtained by matching the ideal MHD solution for the outer layer and the resistive MHD solution for the inner layer.

In resistive MHD, tearing mode stability is determined by parameter ‘stability index, \( \Delta \)’, which is defined as the discontinuity in the radial magnetic field perturbation across the rational surface \( (r_s) \):

\[
\Delta' = (\tilde{\psi}_s + \varepsilon - \tilde{\psi}_s + \varepsilon)/(a \tilde{\psi}(r_s)), \text{ where } \varepsilon \rightarrow 0
\]

(2.17)

A positive \( \Delta' \) implies instability. Its value is related to the growth rate of the mode by:

\[
\gamma \approx 0.55 r_R^{2/5} r_A^{2/5} (\Delta')^{4/5} \left( \frac{a}{n R} \frac{aq'}{q} \right)^{2/5}
\]

(2.18)

where \( \gamma \) is the growth rate of the mode, \( q' \) is the radial derivative of \( q \), and \( n \) refers to the mode number \( (m, n) \). The expression above shows that the development of the instability takes place in a hybrid time scale, faster than the resistive diffusion time scale but slower than the Alfvén time scale.

### 2.2.4 Linear and non-linear tearing mode growth

As long as the island width is much smaller than the resistive boundary layer thickness, island grows exponentially. The mechanism for saturation of mode growth can be explained as follows [2.11]:

...
As the perturbation on $B$ diffuses around the resonant surface, a similar diffusion occurs for the perturbation on $j$. The cross product of the former with the equilibrium current density promotes the occurrence of a perturbation in the fluid velocity field. The convection caused by the perturbation of velocity causes an increase in the magnetic field and its perturbation. In turn, the increase in the magnetic perturbation will generate an increase in the velocity perturbation. This process will account for an exponential growth in the early development of the perturbation.

Rutherford [2.12] showed that when the magnetic island width grows to be as large as the boundary layer width, a new force becomes important and the growth rate of the instability is greatly reduced and finally the island width saturates. Instead of growing like an exponential function in time, the island width starts to grow like a linear function in time. At this time, the growth of the island is related to $\Delta'$ by

$$\frac{dW}{dt} = 1.66\Delta' \frac{\eta}{\mu_0} \left[ 1 - \frac{W}{W_s} \right]$$

(2.19)

where $W$ is the island width, and $W_s$ is the saturation island width.

### 2.2.5 Effect of sheared flows on growth of tearing modes

Shear flow has significant influence on tearing modes by changing the value of stability index $\Delta'$. In the case when flow shear is very small inertial term dominate the convection term. When flow shear is larger than the magnetic field shear the convection term overtakes the inertial term, kinetic energy overpowers the magnetic energy and growth rate of mode is changed. In this case the flow freezes the magnetic field and suppresses the tearing instability. An explanation for stabilizing effect of magnetic shear and flow shear on tearing mode instability was presented by La Haye [2.13], which says that-

Magnetic shear (with no flow shear) varies the field line pitch with $q$ increasing with major radius. Thus the singularity at the rational surface is limited in radial extent. The resonant condition $(1-nq/m)$ becomes $-(r-r_s)/L_q$ ($L_q$ is the magnetic shear length $q/(dq/dr)$ and $r_s$ is radial location of rational surface), which makes tearing “harder”, i.e., it takes more energy to bend field lines, and this makes the linear classical $\Delta'$ more negative, i.e., more stable. When flow shear is added, the island is distorted as shown in Fig. 2.4 due to the viscous drag of a sheared flow and the flow shear provides an additional means to limit the radial extent of the resonance.
2.3 Disruptions in tokamak

Tokamak plasma can last up to hours within some operational limits. Beyond these limits major plasma instabilities result termination of discharge on the millisecond timescale, known as disruption. Plasma density, current and pressure are limited by MHD instabilities and any attempt to raise these parameters above the stable value ends in a disruption. Disruption is preceded by growth of MHD modes, which destroy magnetic flux surfaces creating magnetic field lines, which fill the whole plasma in a stochastic manner. As a consequence of the ergodicity of the magnetic field the plasma energy confinement is lost. This phase is called thermal quench and is followed by the increase of the plasma resistivity and by the Ohmic dissipation of the plasma current. A disruption causes the loss of significant thermal loads and mechanical forces on the plasma facing components and therefore must be avoided.

2.3.1 Causes of disruptions

There are several identified causes of disruptions. Operating limits for steady operation are imposed by both low-q disruptions and density limit disruptions. These are thought to be due to unstable current profiles which lead to large amplitude tearing modes. These two operational limits can be combined in a diagram which was found by Hugill [2.14]. In this diagram (Fig. 2.5) the boundary of operation as limited by disruptions is plotted against $1/q_a$ and Murakami parameter $(\bar{n}R/B_\phi)$ [2.15].

\[\text{Figure 2.4. (a) Magnetic island with magnetic shear with no flow shear and } q \text{ increasing with major radius } R. \text{ (b) Magnetic island with flow shear added.}\]
Chapter 2: Tokamaks and ADITYA Tokamak

Figure 2.5. Hugill diagram shows limitation of stable operational regime of the tokamak in a region of \((q_a, \bar{n})\) space, where \(q_a\) denotes the safety factor at the edge of the plasma and \(\bar{n}\) denotes the mean electron density. In this diagram operating regime is shown as a function of \(1/q_a\), which is proportional to plasma current \((I_p)\), and of the Murakami parameter \(\bar{n}R/B_\phi\). ‘low-\( q \) disruptions’ and ‘density limit disruptions’ limit this region. Operating regime is reduced for contaminated OH-plasmas and extended beyond the \(Z_{eff} = 1\) boundary when auxiliary heating is employed.

**Density limit disruptions**

The plasma density achievable in a tokamak is roughly proportional to the toroidal current density. Highest achievable density for a stable tokamak operation is known as ‘density or Murakami limit’ [2.15]. Above this critical plasma density disruptions occur. An empirical formula for this critical density in tokamaks is

\[
\bar{n}_{crit} \approx 10^{20} B_\phi A_i^{1/2} / q_a Z_{eff} R_0 (m^{-3}) ,
\]

where \(A_i = \text{(ion mass)}/\text{(proton mass)}\) [2.16]. Attempt of increasing the density with gas puffing leads to further cooling of cold high density plasma edge due to rise of radiation. Cooling is more localized on the high field side, which is called MARFE (multifaceted asymmetric radiation from the edge). In this process resistivity increases at the edge, the toroidal current contracts and leads to steepening the profile. In this scenario peaked current
Chapter 2: Tokamaks and ADITYA Tokamak

profile may develop tearing modes on rational flux surfaces, which form magnetic islands. These islands grow and trigger major disruption. The cooling of the edge plasma, i.e., the shrinkage of the temperature profile can be postponed by additional heating hence shifting the density limit to higher values.

**Low-q disruptions**

High $\beta$ plasmas can be achieved either by increasing the plasma current or by decreasing the external magnetic fields. In both the cases when $q_a = 2$ approaches at the plasma boundary and $q_0 \leq 1$, the $m/n=2/1$ tearing mode, driven to large amplitude at the $q(r) = 2$ surface, alters $J,(r)$ so that neighbouring modes (3/2) or (5/3) also become destabilized and island overlap and field line becomes stochastic, magnetic flux surfaces are destroyed, particles flowing along magnetic field lines cross through the entire region and confinement is spoiled, leading to plasma disruption.

When the plasma current is concentrated along the plasma centre, then $q(0)$ at the centre becomes less than 1 even if $q_a$ is much larger than 1. It has been observed that the tearing mode grows at the rational surface $q_a = 1$ and triggers disruptive instability.

**Beta limit or ballooning modes limit disruptions**

Ballooning modes are pressure-driven instabilities. These are local bulges in the plasma surfaces. These are high-mode-number interchange modes. Plasma flux surfaces bulges out due to the ballooning instabilities created by the high plasma pressure at the place where the curvature of magnetic fields are worst, such as at the tips of a vertically elongated plasma. These instabilities lead to disruptions and hence set upper limit to the plasma pressure or plasma beta for stable tokamak operation.

Stability against ballooning depends upon the magnitude of the curvature and on the connection length, which are related to $q$ and $R_o / a$. For circular plasma in an axisymmetric torus, a simplified theoretical estimate is $\beta \leq a / R_o q^2$ for stability.

**2.4 ADITYA Tokamak**

ADITYA is a medium size tokamak [2.17, 2.18]. This is a poloidal limiter tokamak with circular cross-section plasma and with air core transformer. It has minor radius $a = 25$cm and major radius $R = 75$cm. ADITYA design parameters are listed in Table 2.3.
## Chapter 2: Tokamaks and ADITYA Tokamak

### Design Parameters

<table>
<thead>
<tr>
<th>Design Parameters</th>
<th>Design value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Major radius; $R$</td>
<td>75cm</td>
</tr>
<tr>
<td>Minor radius; $a$</td>
<td>25cm</td>
</tr>
<tr>
<td>Toroidal magnetic field; $B_\phi$</td>
<td>1.5T</td>
</tr>
<tr>
<td>Plasma current; $I_p$</td>
<td>250kA</td>
</tr>
<tr>
<td>Edge safety factor; $q_\alpha$</td>
<td>2.5</td>
</tr>
<tr>
<td>Central electron temperature; $T_{eo}$</td>
<td>400eV</td>
</tr>
<tr>
<td>Length of the pulse duration</td>
<td>300 ms</td>
</tr>
</tbody>
</table>

*Table 2.3. Design parameters of ADITYA tokamak*

### 2.4.1 ADITYA tokamak subsystems

Major subsystems of ADITYA tokamak are: magnetic coils, vacuum vessel, pumping systems, gas puff systems, wall conditioning systems, limiters, pulsed power system and data acquisition and control system.

#### 2.4.1.1 Magnetic coils in ADITYA

In a tokamak electromagnetic coils are used to produce magnetic field for various purposes. Ohmic heating (OH) power generates the transformer flux that produces and heat the plasma. Toroidal magnetic field ($B_\phi$) confines the plasma and vertical or equilibrium field keeps the plasma in an equilibrium position.

**Ohmic Transformer (OT or TR) coils**

In a tokamak plasma current is generated by the transformer action. Ohmic transformer initiates plasma discharge and drive plasma current. Plasma currents forms poloidal magnetic field ($B_\theta$) and heats plasma Ohmically. In ADITYA in this set of coils central solenoid TR1, which acts as primary of transformer produces total flux swing $V_{Loop} \times \Delta t \sim 1.2V\text{-sec}$ required to produce and maintain plasma current $I_p \sim 250kA$ for a duration $\sim 300$ms. Stray fields produced by TR1 in the plasma region are minimised by compensating coils TR2, TR3, TR4 and TR5. Vertical cross-section of OT (or TR) coils is shown in Fig. 2.7.
Chapter 2: Tokamaks and ADITYA Tokamak

**Toroidal Field (TF) coils**
Toroidal magnetic field causes charged particles to spiral around field lines. Plasma particles are lost to the vessel walls only by relatively slow diffusion across the field lines. Toroidal magnetic field is produced by 20 numbers of equidistant rectangular coils. These coils are designed to generate a magnetic field of 1.5T at the plasma centre. Discreteness of coils causes <2.5% ripple in toroidal magnetic field. Vertical cross-section of TF coils is shown in Fig. 2.7.

**Vertical Field (B_V) Coils**
Under the uniform plasma kinetic pressure and non-uniform magnetic field pressure on a flux surface plasma current ring expands as net pressure is stronger at the in-board side than on the out-board side. It is necessary to eliminate this net force by adding a vertical field. This radially outward directed force on the plasma is balanced by vertical magnetic field generated by quadrupole of poloidal magnetic field. Necessity of vertical magnetic field (B_V) coils can be explained by following argument.

**Stability relative to displacement along the axis of symmetry (êz)**
With a purely vertical dipole magnetic field, \( B_V = B_z \), the plasma is neutrally stable relative to its displacement along vertical direction (êz). However, if the line of force of the confining field are slightly concave towards the major axis (Fig. 2.6) and \( B_v = B_z + B_R \), then the accidental displacement in vertical direction is opposed by restoring force \( F \sim I_p \times B_R \) produced by interaction of the \( R \)-component of the field with the current and plasma position is stable with regard to up and down motion.

The concavity of vertical field line is written as decay index. \( n = -\frac{R}{B_z} \frac{\partial B_z}{\partial R} \) is radial decay index of the externally applied magnetic field. It is linked to the curvature of \( B_z \) and the vertical variation of its radial component.

The condition of stability of the equilibrium position of the plasma column relative to vertical displacements can be written as \( n > 0 \) [2.19, 2.20].

**Stability relative to horizontal displacement (êR)**
To maintain plasma column in equilibrium against horizontal axisymmetric displacement, the interaction of the \( z \)-component of the external field and the plasma current produces a radially
Chapter 2: Tokamaks and ADITYA Tokamak

inward directed force that just offsets the imbalance in the plasma pressure and poloidal magnetic field pressure,

\[ F_R = -I_p R B_v \]  
\[ (2.21) \]

The variation of this force with radial position is given by

\[ \frac{\partial F_R}{\partial R} = -IB_v \left( \frac{R}{I_p} \frac{\partial I_p}{\partial R} + 1 - n \right) \]  
\[ (2.22) \]

where \( B_v \) is vertical magnetic field in absence of plasma. For horizontal stability the condition on the field index is given by \( n < 3/2 \) to ensure \( \partial F_R / \partial R < 0 \). Hence the condition for stability of the equilibrium against displacements is expressed in terms of the field index as: \( 0 < n < 3/2 \).

\[ \text{Figure 2.6. Restoring force produced by concave field lines against vertical (} \hat{z} \text{-direction) displacement} \]

In ADITYA to maintain plasma column in equilibrium position a set of two vertical magnetic field (\( B_v \)) coils, \( B_v1 \) and \( B_v2 \) are placed symmetrically around the mid-plane. These coil position and Ampere-turns are chosen such that \( n > 0.4 \) and \( < 1.2 \). Vertical cross-section of \( B_v \) coils is shown in Fig. 2.7.

2.4.1.2 Vacuum vessel and associated systems

ADITYA vacuum vessel is a ultra-high vacuum compatible and made out of SS-304L material with major radius 75cm and square cross-section of side 60cm. Vessel is assembled
in four quadrants. At one toroidal location electrical discontinuity is provided to allow fast poloidal magnetic field penetration into the vessel. Base pressure of $\sim 1 \times 10^{-7}$ Torr is achieved by three turbo-molecular pumps and one cryogenic pump. Each turbo-molecular pump has pumping capacity of 2,000 l/s for N$_2$. Cryogenic pump has pumping capacity of about 9,500 l/s for water vapour and condensed hydro-carbons.

2.4.1.3 Wall conditioning

Normally, vacuum vessel surface is covered with a layer of adsorbed gas atoms with weak binding energy. These impurities can be desorbed by incident ions, neutrals, electrons and photons back into the plasma. This process is called particle recycling. Desorbed atoms radiate away the energy and result cold boundary plasma. In order to minimize recycling due to plasma–wall interaction during plasma discharge various wall treatment procedures have been adopted before plasma discharge. In order to condition ADITYA vacuum vessel wall (1) glow discharge cleaning, (2) pulse discharge cleaning in combination with glow discharge cleaning and (3) electron cyclotron resonance cleaning are carried out.

Figure 2.7. Vertical cross-section of TF coil, OT (or TR) coil, B$_v$ coils in ADITYA
Chapter 2: Tokamaks and ADITYA Tokamak

2.4.1.4 Gas puff system
Hydrogen gas is injected from a constant pressure reservoir through fast response piezo electric valve. There are two stages of gas filling: (1) Pre-filling of torus to a specific pressure prior to the initiation of a tokamak discharge, (2) Pre-programmed gas feed for building up plasma density and to control HXR. At working pressure of $\sim 1 \times 10^{-5}$ Torr plasma is produced.

2.4.1.5 Limiter
Limiter is the first material surface to come into contact with the hot plasma. Limiter receives particle and heat load. ADITYA has two poloidal limiters of graphite material- working limiter and safety limiter. Working limiter receives all heat loads and has graphite tiles mounted on two semi-circular rings. This limiter has radius of 25cm and defines plasma size. Safety limiter generally does not receive significant particle and heat load. It is designed to take the full load in the event of failure of working limiter.

2.4.1.6 ADITYA Pulsed Power System (APPS)
ADITYA Pulsed Power System (APPS) mainly consists of three pulsed power supplies. Its main function is to deliver current pulses of specified shape, amplitude and duration to Ohmic transformer (OT or TR) coils, TF magnet coils and vertical field ($B_V$) coils.
APPS consists of two major sub-systems namely the 132kV/ 11kV sub-station including the reactive power compensation system and the DC system comprising of the line commuted converters, pulse shaping units and the control instrumentation.
The power system consists of 12-pulse thyristor-based wave-shaping dc converter system for Ohmic power supply ($\pm 25$ kA, 2000 V), TF power supply (25 kA, 425 V), the ripple frequency is 600 Hz and voltage ripple is 6%. With total load inductance is 6mH, the current ripple is estimated 0.0045%.

**Ohmic power supply**
Ohmic power supply initiates the gas breakdown and drives the plasma current. To ionize the gas and start current build-up a relatively large voltage is required, which is followed by a lower loop voltage for slower plasma current changes and less runaway electron generation. To produce a desired plasma loop voltage and current an active wave shaping circuit is connected between the dc power converter and the OT coils.
Chapter 2: Tokamaks and ADITYA Tokamak

OT power supply, which is also called converter consists of four thyristor bridges. Two bridges are in series, which are in parallel of other two bridges in series. Each thyristor produces 3 pulse output and each series combination gives 12.5kA. In this way the converter is 12 pulse 25kA output.

**Vertical field power supply**

Vertical field power supply is a thyristor based power supply, which delivers power to $B_V$ coil. This power supply can be operated by two methods- (a) by pre-programming and (b) by $I_p$-control or feed- back control. In second method current in $B_V$ coil is supplied in such a way that it produces $B_V \propto I_p$.

**Toroidal field power supply**

Toroidal field power supply (50 kA, 425 V) is a simple thyristor based converter. A 132/11 kV transformer of 50 MVA rating supplies the input power to the converter system.

![Figure 2.8](image)

*Figure 2.8. Anti transformer arrangement to nullify mutual inductance between Ohmic Transformer (OT) coil and Vertical Field (Bv) coil

**Anti-transformer and its significance**

In ADITYA tokamak OT (or TR) coil and $B_V$ coil are very close to each other. So there is mutual inductance between the two coils $M = k \sqrt{L_{OT} L_{BV}}$, where $k$ is the coupling coefficient between the two coils, $L_{OT}$ is the inductance of the OT coil, and $L_{BV}$ is the inductance of the $B_V$ coil. To nullify effect of mutual inductance an anti-transformer is used. Primary coil of anti-transformer is connected in series with OT coil and secondary coil is
connected in series with $B_V$ coil. Flux linkage between primary and secondary coil is such that it nullifies the flux linkage due mutual inductance between OT and $B_V$ coil.

### 2.4.1.7 ADITYA Data acquisition system and control system

The data acquisition and control of the entire tokamak system is carried out using control system based on Computer Automated Measurements And Control (CAMAC) concept. It is designed to provide various channels with selectable sampling rates and different selectable time slots synchronized for pulsed experiments.

The analog signals coming from various diagnostics are passed through opto-isolators and sent to digitizers. The digitizers are software/hardware settable for appropriate sampling rate and memory length depending upon the frequency of interest and viewing time interval during plasma discharge. The time synchronisation between the plasma discharge and acquisition is achieved with the help of appropriate trigger signal derived from control process and timer modules. During the interval between successive discharge the local memory of digitizer is read by computer and the data is stored on hard disk for immediate and future processing.

### 2.4.2 Typical ADITYA tokamak discharge

As discussed in previous section, in ADITYA toakmak vacuum vessel is evacuated to a base pressure $\sim 10^{-7}$ torr. At $\sim$-2s TF power supply is turned on (Fig. 2.9). At $\sim$-0.5s current in TF coil reaches to flat–top, which remains flat till $\sim$ 0.2s. Toroidal magnetic field during TF current flat–top is maintained at $\sim$0.70-0.8T. Ohmic power supply is used for plasma breakdown and current drive. At $\sim$ -1.1s Ohmic transformer (OT) power supply is switched on which reaches to maximum value at $\sim$-0.3s. At $t$- -0.2s H$_2$ gas is filled up to produce $\sim 1 \times 10^{-5}$ torr pressure. At 0ms fast change in current $(dI_{OT}/dt)$ induces toroidal electric field (loop voltage) inside vacuum vessel. In this electric field H$_2$ gas plasma is produced. Simultaneously, vertical field power supply is switched on. Vertical field is applied for the control of radial motion of plasma. It is maintained as proportion to plasma current all the time so that the plasma can be kept at center.

Toroidal electric field $E_r = V_{loop} / 2\pi R$ accelerates free electrons, which are always present due to the cosmic radiation. But their amount is not sufficient to produce fast discharge. Additional free electrons are produced by pre-ionization filament to produce fast and reproducible discharge. At breakdown $H_\alpha$ intensity is at its maximum value. After
breakdown, $H_n$ intensity reduces and electron density $n_e$ increases. Temporal profile of typical ADITYA tokamak discharge (#22761) is shown in Fig. 2.10.

![Temporal profile of current signal](image)

**Figure 2.9.** Temporal profile of current signal in (a) toroidal field magnetic coil ($I_{TF}$), (b) Ohmic transformer coil ($I_{OT}$), (c) Vertical magnetic field coils ($I_{BV}$) for a typical ADITYA tokamak discharge (#26571). Time of hydrogen gas pre-fill is shown by red dotted line.

In ADITYA rate of plasma current increase ($dI_p/dt$) just after breakdown is kept ~7–9MA/s. During current rise phase run-away electrons are generated which are detected by HXR detectors. These initial run-away electrons are removed by puffing H$_2$ gas. After the removal of these run-away electrons part, plasma current takes a dip and then rises again. After the plasma current reaches to plateau (~60-90kA), ~2-3V loop voltage is sufficient to drive plasma Ohmically.

It is observed that during plasma disruption negative spikes occur. These negative spikes are due to plasma inductance decrease. The sudden decrease of the inductance is due to expansion of the plasma radius across the strong magnetic field [2.21 – 2.23].
Figure 2.10. Temporal profile of typical ADITYA tokamak discharge (#22761). (a) Loop voltage ($V_L$), (b) plasma current ($I_P$), (c) Vertical magnetic field ($B_V$), (d) $H_\alpha$ line radiation, (e) SXR signal, (f) HXR signal

2.4.3 Operational diagnostics in ADITYA tokamak

Diagnostics for tokamaks have generally been developed to run tokamaks by measuring plasma parameters and to study particular topics in tokamak research.

2.4.3.1 Magnetic measurements

Magnetic measurements in tokamaks are carried out to setting up stable plasmas, investigate MHD instabilities, determine energy confinement time etc.
2.4.3.1.1 Loop voltage measurement

The Loop voltage also called the Volts per Turn or Surface Voltage, is used in calculating the Ohmic power input to the plasma. It also allows a calculation of the plasma parallel resistivity. Loop voltage is simply measured by a single turn pickup coil toroidally wound close to the plasma. In ADITYA tokamak during plasma current flat-top loop voltage remains ~2-3V.

2.4.3.1.2 Rogowski coil

Plasma current is measured by a ‘Rogowski coil’, which is multi- turn solenoid completely enclosing the current to be measured. The transient plasma current generates a voltage

\[ \varepsilon = \frac{\mu_0 N \pi r^2}{2\pi R} \frac{dI_P}{dt}, \]  

(2.23)

where \( N \) is number of turns, \( r \) and \( R \) are minor and major radius of the Rogowski coil. \( I_P \) is deduced from the time integration of equation (2.23). In ADITYA plasma current is measured by a Rogowski coil with square cross-section of \( 4 \times 10^{-4} \text{ m}^2 \) and side length 0.64m [2.18]. It has 4600 turns and made of Teflon insulated copper wire.

2.4.3.1.3 Mirnov coil

In tokamaks it is expected that magnetic islands play a role in determining transport. Their structure is expressed in the form of helical Fourier modes \( e^{i(m\phi+n\theta)} \) (\( m \) and \( n \) being poloidal and toroidal mode numbers), and they are located at rational surfaces. Mirnov [2.24] first studied their presence using array of \( B_\theta \) coils in a single poloidal plane, measuring \( dB_\theta /dt \) outside the plasma. Poloidal mode structure of such perturbation is determined by Mirnov coils. Toroidal mode structure is determined from a toroidal array of \( B_\theta \) coils. In ADITYA a poloidal array of 30 magnetic probes (15 \( B_\theta \) coils and 15 saddle coils) [2.25] mounted inside the vacuum vessel at \( r = 27 \text{ cm} \) allows measurements of fast \( dB_\theta /dt \) and \( dB_\phi /dt \) oscillations. Frequency response of these probes probes linear up to ~30kHz.

2.4.3.1.4 Diamagnetic loop

Energy confinement time measurement is a subject of particular interest. This is carried out by determining the energy content, \( W \), using a diamagnetic loop and calculating the confinement time from \( \tau_E = W/P \), where \( P \) is the input power to the plasma. The quantities
which require to be measured to determine the thermal energy and internal inductance (or poloidal beta) are the magnitude and direction of the magnetic field at the measuring surface, and the diamagnetic flux. The diamagnetic flux is the difference between the total toroidal flux with plasma and that in the absence of plasma. This flux is measured with loop enclosing the plasma, encircling it poloidally, called diamagnetic loop.

Relation between the diamagnetic flux and the poloidal beta derived from simplified equilibrium relation

\[
\beta_p = 1 + \frac{8\pi B_{\phi_0} \delta \phi}{\mu_0 I_p^2}
\]

(2.24)

where \( B_{\phi_0} \) is the toroidal magnetic field in the absence of the plasma which can be obtained by the magnetic probe,

\[
\beta_p = \frac{8\pi^2 a^2 \langle p \rangle}{\mu_0 I_p^2}
\]

(2.25)

for a circular cross-section, \( I_p \) is the plasma current which can be obtained by the rogowski coil.

\[
\delta \phi = \pi a^2 (\bar{B}_\phi - B_p(a))
\]

(2.26)

is the change in the toroidal flux due to plasma pressure and \( \bar{B}_\phi \) is the average toroidal magnetic field. The energy confinement time \( \tau_E \) can be calculated from \( \beta_p \) and plasma resistance \( R_p = \frac{P}{I_p^2} \) using the relation

\[
\tau_E = \frac{3}{8} \mu_0 \beta_p (R/R_p)
\]

(2.27)

In ADITYA diamagnetic flux is measured by a single-turn loop (diamagnetic loop) [2.26] and a compensating loop of six turns made of polyimide-insulated shielded wire. Its shield ensures complete immunity to electrostatic pick-up. Flux measurement channels use low pass filters at 157Hz. Signal is converted from analog to digital by CAMAC module at sampling rate of 5 kHz.

2.4.3.2 Microwave interferometry

Microwave interferometer technique is a well known method of measuring plasma density. This method gives line averaged electron density along the line of sight through the measurement of the phase shift in the microwave beam due to the plasma. Microwave beam during its passage through plasma suffers a phase shift (\( \phi \)) due to the change in the refractive
index of the medium. For propagation path length, \( L \), average density is given as
\[
\bar{n} = 2\varepsilon_0 n_e c \omega \phi / e^2 L,
\]
where \( \omega \) is frequency of microwave.

In ADITYA a seven chord millimetre wave interferometer at 100GHz homodyne interferometer system is used to measure the radial electron density profile \([2.27]\). Each chord is separated by a distance of 7cm at the median plane of plasma. The phase changes are detected by Schotky diode.

**2.4.3.3 Soft X-ray measurements**

Two identical detectors with different thickness of thin metallic foils, usually Beryllium, provide the possibility of measurement of electron temperature \((T_e)\) by means of the absorber foil technique \([2.28]\). The time evolution of the X-ray flux provides information about MHD activity also.

*The two foil method or absorption method*

The two foil method of determining \( T_e \) relies on the fact that the continuous X-ray power spectrum depends upon \( \exp(-E/T_e) \), and the transmission properties of X-ray through thin foils. The transmission coefficient through a thin metal foil is strongly energy dependent and a foil of particular thickness will effectively have a cut-off energy, \( E_1 \), below which it is opaque to incident X-rays. A thicker foil will have a higher cut-off energy, \( E_2 \), and so if the X-rays from a region of hot plasma are viewed by two detectors shielded by the two different foils, then the ratio of the observed power in each detector may be written very approximately as

\[
R = \frac{P_1}{P_2} = \left\{ \frac{\int_{E_1}^{\infty} \exp(-E/T_e) dE}{\int_{E_2}^{\infty} \exp(-E/T_e) dE} \right\} = \exp\left(-\frac{(E_2 - E_1)}{T_e}\right)
\]

(2.28)

The measurement of \( R \) and knowledge of \( E_1 \) and \( E_2 \) will then determine \( T_e \). The expression for \( R \) has been evaluated in an exact manner by Jahoda et al \([2.28]\) for a variety of different foil thickness and temperatures.

In ADITYA Soft X-rays are detected by silicon surface barrier diodes, which measure the integrated X-ray power above a threshold energy determined by Beryllium, placed in front of
the detectors. Central line averaged electron temperature was measured with the transmission ratios of soft X-ray flux through two Beryllium foils of 25 and 70 micron thickness [2.29].

2.4.3.4 Hard X-ray measurements

Hard X rays are photons with energy greater than hundreds of keV. These radiations arise when fast relativistic electrons (runaway electrons) strike the limiter or the first wall. In present day tokamaks, runaway electrons constitute a serious concern. The first and the most common goal of HXR measurement is simple evidence of the presence of runaway electrons in tokamak plasmas. In ADITYA NaI(Tl) scintillators coupled with photomultipliers are used to detect HXRs. The calibration of detector is done by the standard source of the Caesium-137 or Cobalt. Normally in ADITYA, it is kept at the 1V output signal corresponding to the 1MeV of radiations.

2.4.3.5 Spectroscopy

Because electron temperature edge value range from few eV to few tens of eV, atoms are neutral or in state of low ionization and visible radiation are emitted over a wide spectral range in tokamaks. Line emissions ($H_\alpha$, CIII, OI etc) are observed with visible light spectrometers or monochromators to monitor plasma discharge performance. Ruled grating are used as dispersing elements and photomultiplier tubes as detectors. The aim of much of this work is to determine impurity influxes. Impurity content is often determined from the intensity of the visible continuum radiation. Measurement of continuum radiation intensity in the visible spectral region is carried out to estimate the effective charge of plasma ions $Z_{\text{eff}}$. UV spectral region is used for particle transport studies. Grazing incidence and normal incidence monochromators are used to measure the intensity of resonance line radiation from impurity. A multi-track spectrometer is used to monitor time and space resolved visible spectral lines [2.30].

2.5 References

Chapter 2: Tokamaks and ADITYA Tokamak