ADITYA UP-GRADATION EQUILIBRIUM STUDY

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Abstract

The ADITYA tokamak device is used to produce circular plasma for few hundreds of milli-seconds. The edge physics study in this device is led to significant contributions. The up-gradation of this device (ADITYA-U) is focused to address issues relevant to heat removal capability at the plasma edge. This requires to construct plasma equilibrium with divertor configuration. In this regard, additional pair of coils at the inboard and outboard are introduced to construct plasma equilibrium. The inboard pair mainly creates the divertor configuration while the outboard pair provides flexibility in increasing the size of the plasma. This study has shown that plasma equilibrium with double and single null configurations can be produced for plasma current up to 100 kA and with plasma poloidal beta of 0.3. The operational space in these plasma parameters is limited by the requirement on gap of 3 cm between null point and the vacuum vessel as well as limited due to maximum allowable current of 150 kA in divertor coil other formats will be accepted. This paper describes the required layout for the manuscripts and how they should be submitted.

1. INTRODUCTION

The design of ADITYA [1] tokamak was to produce plasma with ohmic (TR) coils and sustain it for few hundreds of milli-seconds. A two pair of copper coils (BV1 & BV2) provided the equilibrium of circular shaped plasma while a pair of Fast Feed (FF) back coils provided the radial control. In the past few decades, the plasma operation in ADITYA approached the design values in terms of plasma current, shot duration, edge q-value, plasma density and temperature. The additional features added to this device is heating and current drive systems namely ECRH of 500Mw at 42 GHz, ICRH of 200 kW at and LHCD up to 100kw at 3.4 GHz. These systems are routinely used to produce plasma scenario of our interest. The recent experiments in ADITYA addressed issues like dynamics of runaways [2], 2nd harmonics ICRH minority heating [3], effect of electrode biasing and flow related issues [4] With these achievements, it is decided to upgrade the device so that this device could address the latest physics issues related to up coming devices like ITER. The one critical area of tokamak is heat handling capability and plasma-wall interaction and this could limit the scope of tokamak as a fusion reactor. This issue can be addressed either by developing material to handle high heat flux or construct divertor configuration to reduce the heat load on the divertor plate. The presently available material for the divertor target is tungsten and can with stand up to 10 MW/m² Hence, this device should be designed to have flexibility in divertor configuration and divertor target. As this device is pulsed one, material for divertor targets can be characterized in this device. The divertor configuration like single -null and double null are of greater importance and the design of this device should have these features.

2. MHD EQUILIBRIUM USING IPREQ

In tokamak, the toroidal plasma column expands radially outward due to hoop force. The basic equilibrium of plasma against the hoop force is provided by the external coil currents [5]. These external coils are also used to shape the plasma boundary which has advantages over stabilizing ideal MHD modes and improve the plasma confinement [6]. In the equilibrium construction, the plasma boundary is defined either by limiter or by divertor plate. The limiters can not be used for removing high heat load as well as for maintaining long pulse operation. The divertor configuration can overcome this issue and hence, it is important to have plasma limited by magnetic null at one location in the poloidal cross-section of tokamak (x-point) [7]. To construct such equilibria, one has to solve the Grad-Shafranov [5] equation along with the external coil currents. Tokamak is a toroidal device with symmetric along the toroidal direction. Hence, it is enough to construct equilibrium in the poloidal cross-section. For a static equilibrium, using ideal MHD equation, the momentum equation along with Maxwell’s equations together leads to Grad-Shafranov (GS) equation (Eq. 1) which describes the tokamak equilibrium. IPREQ [10] is anaxisymmetric equilibrium code solving GS equation in the cylindrical co-ordinate system (R,Φ,Z).
\[ R \frac{\partial}{\partial R} \left( \frac{1}{R} \frac{\partial}{\partial R} \right) + \frac{\partial^2}{\partial Z^2} \psi = -R \mu_0 J_\theta \]  \hfill (1)

\[ J_\theta = R \frac{dp}{d\psi} + \frac{F}{R \mu_0} \frac{dF}{d\psi} \]  \hfill (2)

IPREQ code solves the GS equation which is an elliptic partial differential equation for the poloidal flux function \( \Psi (R, Z) \). This equation is highly non-linear and it is necessary to employ iterative procedures for solving this equation. The equilibrium can be easily constructed if one could specify the pressure profile and toroidal flux function as defined in Eq.2 and 3. This definition is known as ORNL profile [11] and this code has the capability of using other profile definitions also. For the present study, we have used ORNL profile definition and is defined by Eq. 4.

\[ \frac{dp}{dx} = P_0 \left| e^{-\alpha x} - e^{-\alpha} \right| / \left| e^{-\alpha} - 1 \right| \]  \hfill (3)

\[ \frac{dF^2}{dx} = 2 \mu R_0^2 P_0 \left| 1/\beta_j - 1 \right| \left| e^{-\gamma x} - e^{-\gamma} \right| / \left| e^{-\gamma} - 1 \right| \]  \hfill (4)

We consider the rectangular computational domain in R and Z with external poloidal field (PF) coils. The computation domain includes the plasma and vacuum region, the toroidal current density \( J_\Phi \) is non-zero in the plasma and zero in the vacuum region which surrounds the plasma. The self-consistent determination of this plasma-vacuum interface, \( \Psi = \Psi_x \) (plasma is limited by a x-point), is an important part of the solution. The inner and outer limiter is also considered at the mid plane for this calculation so that they will act as safety limiters in the device. The equilibrium construction of ADITYA-U using IPREQ is described in the next section. The operational flexibility in terms of plasma parameters and coil currents are also studied.

3. EQUILIBRIUM OF ADITYA CIRCULAR PLASMA

The existing ADITYA device has a square vacuum vessel with poloidal limiter to limit the plasma at a radius of about 25 cm. The ohmic system is capable of initiating and sustaining plasma current up to 250 kA. The present ADITYA device produces circular equilibrium supported by two pairs of copper coils. These coils are connected in series and fed with single power supply. The coil BV1 has 22 turns and BV2 has 60 turns and capable of carrying conductor current up to 5 kA/turns. A typical ADITYA equilibrium with circular shape for plasma current of 150 kA is shown in Fig. 2 and the coil parameters and plasma parameters are shown in Table-1.

Table 1. Vertical coil parameters and Plasma parameters

<table>
<thead>
<tr>
<th>Coil</th>
<th>R(m)</th>
<th>Z(m)</th>
<th>AR(m)</th>
<th>AZ(m)</th>
<th>NI(MA)</th>
</tr>
</thead>
<tbody>
<tr>
<td>BV1</td>
<td>0.366</td>
<td>1.05</td>
<td>0.164</td>
<td>0.115</td>
<td>-0.288</td>
</tr>
<tr>
<td>BV2</td>
<td>1.634</td>
<td>1.19</td>
<td>0.18</td>
<td>0.038</td>
<td>-0.105</td>
</tr>
</tbody>
</table>

Ip(kA)  R_s(cm)  a(cm)  K  Bp  li
150     73.4     23.0   0.98 0.02 1.01
For this, the central density of ADITYA plasma is $3 \times 10^{19} \text{m}^{-3}$ and central temperature of 400 eV. The plasma will lean to inner or outer limiter depending upon the plasma equilibrium. For this equilibrium, the conductor current in BV coils is about 4.8kA. Fig.1 shows the schematic of square vacuum vessel with circula plasma. In this equilibrium, the shift of magnetic axis and the plasma last closed flux surface has a rigid shift of 2.0cm from vessel center of 75cm. The internal inductance is about ~1. The plasma minor radius is 23 cm and is circular in shape. The pressure profile (P), Toroidal current (J) profile and safety profile (q) are shown in Fig 3 & Fig 4 respectively. Such plasma equilibrium is sustained for 160 ms with ohmic

4. MODIFICATION REQUIRED FOR ADITYA UPGRADATION

In the ADITYA-upgradation, there are few technical constraints/implications: The redesign/modification of Toroidal Field (TF) coil will need longer shutdown period and hence it is decided that TF will remain as such. Aditya square vacuum vessel is replaced by circular vessel to accommodate the divertor coils. This has introduced a constraint that circular vessel will have diameter equivalent to the width & height of the square vessel (~60 cm) The available space for copper divertor coils will be around 25 sq. cm and this can allow maximum of 150 kAt in each coil Plasma equilibrium with x-point should be constructed within the circular vessel and the x-point should be kept as far as possible from the vessel. For operational flexibility, it is decided to have independent power supplies for vertical field coils (BV1 & BV2) The coil system should be able to produce single and double-null plasma divertor configuration over a reasonable range of plasma parameters and should occupy the entire vacuum vessel.

Table 2. Proposed coil parameter
Table 3: Plasma Parameters for the Aditya up-gradation with low $\beta_p$

<table>
<thead>
<tr>
<th>Cases</th>
<th>BV1</th>
<th>BV2</th>
<th>D1</th>
<th>D2</th>
<th>$\beta_p$</th>
<th>$R_0$(cm)</th>
<th>$a$(cm)</th>
<th>Vol</th>
<th>$li$</th>
<th>$K$</th>
<th>$Rxd$(cm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>-136kAt</td>
<td>-50kAt</td>
<td>150kAt</td>
<td>0kAt</td>
<td>0.02</td>
<td>70.2</td>
<td>14.5</td>
<td>0.39</td>
<td>0.99</td>
<td>1.37</td>
<td>4.65</td>
</tr>
<tr>
<td>2</td>
<td>-136kAt</td>
<td>-50kAt</td>
<td>150kAt</td>
<td>8kAt</td>
<td>0.02</td>
<td>77.6</td>
<td>18.0</td>
<td>0.61</td>
<td>1.07</td>
<td>1.18</td>
<td>4.50</td>
</tr>
<tr>
<td>3</td>
<td>-136kAt</td>
<td>-50kAt</td>
<td>100kAt</td>
<td>0kAt</td>
<td>0.02</td>
<td>72.1</td>
<td>17.3</td>
<td>0.56</td>
<td>1.01</td>
<td>1.37</td>
<td>2.4</td>
</tr>
</tbody>
</table>

Table 4: Plasma Parameters for the Aditya up-gradation with high $\beta_p$

<table>
<thead>
<tr>
<th>Cases</th>
<th>BV1</th>
<th>BV2</th>
<th>D1</th>
<th>D2</th>
<th>$\beta_p$</th>
<th>$R_0$(cm)</th>
<th>$a$(cm)</th>
<th>Vol</th>
<th>$li$</th>
<th>$K$</th>
<th>$Rxd$(cm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>-136kAt</td>
<td>-50kAt</td>
<td>150kAt</td>
<td>0kAt</td>
<td>0.29</td>
<td>72.5</td>
<td>15.8</td>
<td>0.46</td>
<td>1.04</td>
<td>1.26</td>
<td>3.61</td>
</tr>
<tr>
<td>2</td>
<td>-136kAt</td>
<td>-50kAt</td>
<td>150kAt</td>
<td>4kAt</td>
<td>0.29</td>
<td>76.8</td>
<td>18.5</td>
<td>0.62</td>
<td>1.13</td>
<td>1.15</td>
<td>3.72</td>
</tr>
<tr>
<td>3</td>
<td>-136kAt</td>
<td>-50kAt</td>
<td>100kAt</td>
<td>0kAt</td>
<td>0.29</td>
<td>75.1</td>
<td>19.9</td>
<td>0.70</td>
<td>1.01</td>
<td>1.20</td>
<td>2.0</td>
</tr>
</tbody>
</table>

5. DIVERTOR CONFIGURATION

The plasma equilibrium with divertor configuration is constructed with IPREQ [8] and one such equilibrium is validated with CEA code [9]. The plasma beta, which is the measure of stored plasma pressure to magnetic pressure, is varied to show the capability of divertor coils. The Table-3 shows the coil current requirements and plasma parameters for low poloidal beta (~0.02). The Table-4 shows the corresponding cases for higher poloidal beta so that in the future if heating systems/input power is enhanced, these coils should be able to support these plasma equilibria. The coil currents are in ampere-turns. The plasma current for these cases is about 100 kA. The coil currents are in their limit, further increase in plasma current is not possible. The coils currents simply scale linearly with plasma current.
5.1 Double null configuration

In the case-1, the outer pair of divertor coil (D2) is not carrying current. The inner pair of divertor coil (D1) is carrying 150 kAt (up and bottom coils). Fig. 6 shows the flux surface plot for this configuration. The green vertical lines represent the in-board and out-board limiters at the mid-plane.

This equilibrium plasma is formed close to inboard side and the plasma radius is also smaller. The plasma is limited by the separatrix, which is shown in the figure. The null points are well within the vacuum vessel. In case-2, the outer pair of divertor coils is carrying about 8 kAt. This makes the plasma radius larger and hence the plasma occupies the entire volume of vacuum vessel as shown in Fig.7. This coil can be used to form different size of plasma, which may be useful to vary physics parameter like .

6. OPERATIONAL FLEXIBILITY

6.1 Low $\beta_p$

Low $\beta_p$ case is shown in Fig.8. $\beta_p$ is the ratio of plasma to poloidal magnetic field pressure, lower value of $\beta_p$ is 0.02. For this, the central density is $2 \times 10^{19}/m^3$ and central temperature of 300 eV. In low $\beta_p$ with increase in diverter coil current the shift of x-point more towards the vessel center with significant value of Rxd is 4.65 cm. The estimated energy confinement times are obtained from the formula: $\tau_E = 3.8 \times 10^{-21} n_e a^2$ which is referred as Alcator scaling law [12] where $n_e$ is plasma density and $a$ is minor radius. Power loss on divertor plate is 0.195 MW/m$^2$
6.2 High $\beta_p$

For these case high value of $\beta_p$ is 0.29 and corresponding result is shown in Fig.9 shows that at high value the plasma occupy more volume than the lower value $\beta_p$ case. With further increase in $\beta_p$, the outward shift of flux surface becomes significant. Power loss on divertor plate is 0.41MW/m$^2$ corresponding to density $2 \times 10^{19}$/m$^3$ and temperature of 600 eV.

6.3 Equilibrium study with additional divertor coil

With above mentioned results for high $\beta_p$ case maximum x-point distance from vessel boundary (RXD defined in fig.10) is 3.72cm as shown in Table-5. To increase the Rxd value one has to increase either the divertor coil current or decrease the I$_p$ value. But D1 coil is its maximum limit, (in Available space) So the next idea is that one can add extra coil just above the D1 coil to put more force on the x-point, to move more inside the vessel. Detail case study for high $\beta_p$ cases with effect of extra D3 coil is defined in Table.5.

<table>
<thead>
<tr>
<th>Cases</th>
<th>Ip(kA)</th>
<th>BV1(kAt)</th>
<th>BV2(kAt)</th>
<th>D1(kAt)</th>
<th>D2(kAt)</th>
<th>D3(kAt)</th>
<th>RXD(cm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>100</td>
<td>-136</td>
<td>-50</td>
<td>150</td>
<td>0</td>
<td>50</td>
<td>5.10</td>
</tr>
<tr>
<td>2</td>
<td>100</td>
<td>-136</td>
<td>-50</td>
<td>150</td>
<td>5.0</td>
<td>50</td>
<td>5.47</td>
</tr>
<tr>
<td>3</td>
<td>100</td>
<td>-136</td>
<td>-50</td>
<td>150</td>
<td>5.0</td>
<td>60</td>
<td>5.48</td>
</tr>
<tr>
<td>4</td>
<td>120</td>
<td>-136</td>
<td>-50</td>
<td>150</td>
<td>5.0</td>
<td>50</td>
<td>3.12</td>
</tr>
<tr>
<td>5</td>
<td>100</td>
<td>-136</td>
<td>-50</td>
<td>180</td>
<td>0</td>
<td>60</td>
<td>5.8</td>
</tr>
<tr>
<td>6</td>
<td>100</td>
<td>-136</td>
<td>-50</td>
<td>180</td>
<td>8.0</td>
<td>60</td>
<td>6.17</td>
</tr>
</tbody>
</table>

![Figure.10. Flux with presence of D3 coil](image10)

7. CONCLUSIONS

The divertor coil system for ADITYA upgrade has the capability to produce single and double-null configurations. The maximum current carrying capability of these coils should be 150 kAt so that one can form divertor configuration with x-point well inside the vacuum vessel. The outer divertor coils are needed so that the plasma can occupy the entire volume of vacuum vessel depending upon current in this coil. For producing
single-null, the coils have to carry up-down asymmetric currents. In all these configurations, it is assumed that the BV1 and BV2 will be able to carry different currents and hence independent power supplies are needed.

REFERENCES


