Overview of Recent Experimental Results from Aditya Tokamak

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Abstract. Several experiments, related to controlled thermonuclear fusion research and highly relevant for large size tokamaks including ITER, have been carried out in ADITYA, an ohmically heated circular limiter tokamak. Repeatable plasma discharges of maximum plasma current of ~ 160 kA and discharge duration beyond ~ 250 ms with plasma current flattop duration of ~ 140 ms has been obtained for the first time in ADITYA. The discharge reproducibility has been improved considerably with Lithium wall conditioning and improved plasma discharges are obtained by precisely controlling the plasma position. In these discharges, chord-averaged electron density ~ 3.0 – 4.0 x 10^19 m^-3 using multiple hydrogen gas puffs, electron temperature of the order of ~ 500 - 700 eV have been achieved. Novel experiments related to disruption control are carried out and disruptions, induced by hydrogen gas puffing are successfully mitigated using biased electrode and ICR pulse techniques. Runaway electrons are successfully mitigated by applying a short local vertical field (LVF) pulse. A thorough disruption database has been generated by identifying the different categories of disruption. Detailed analysis of several hundred disrupted discharges showed that the current quench time is inversely proportional to q_edge. Apart from this, for volt-sec recovery during the plasma formation phase, low loop voltage start-up and current ramp-up experiments have been carried out using ECRH. Successful recovery of volt-sec leads to achievement of longer plasma discharge durations. In addition to that Neon gas puff assisted radiative improved confinement mode has also been achieved in ADITYA. In this paper, all the above mentioned experiments will be discussed.

1. Introduction

The ADITYA tokamak has been in operation for about more than two decades. The existing ADITYA with limiter configuration is planned to be upgraded into a state-of-art machine with divertor tokamak. Before upgrading the ADITYA, several experiments, related to controlled thermonuclear fusion research, which are very much relevant for large size tokamaks including ITER, have been carried out [1], [2], [6], [7], and [16]. The outcomes of these innovative experiments are very encouraging and will play a significant role in future tokamak operation.

In recent experimental schedule, repeatable plasma discharges of maximum plasma current of ~ 160 kA with duration a quarter of a second with plasma current flattop duration of ~ 140 ms has been obtained for the first time in the first Indian tokamak ADITYA. The discharge reproducibility has been improved considerably with Lithium wall conditioning [1] and precisely controlling the plasma position [2]. Improved discharges are attempted over a wider parameter range to carry out various confinement scaling experiments [3], [4] and [5]. In these discharges, significant improvement in electron density, central electron temperature and global energy confinement time by a factor of 3 has been observed.

Encouraging results from other novel experiments, such as disruptions control and runaway mitigations [6-7], radiative improved confinement mode experiment [8-9], Electron Cyclotron Resonance (ECR) assisted low loop voltage start-up have been obtained in ADITYA.

The injection of impurities (neon, silicon, or argon) leading to radiative edge cooling of a tokamak plasma is of considerable importance for a future fusion reactor, since it could solve the main difficulties related to the interaction of high temperature plasma with the material wall. A cold, highly radiative boundary distributes most of the power uniformly over the first wall. Structural damage due to high local heat deposition could thus be avoided [10-11]. Neon gas puff assisted radiative improved confinement mode has been observed in ADITYA. The density and temperature increased showing an improved confinement behaviour along with the increase in the radiated power. This indicates a better particle and energy confinements with Neon puff.

Low loop voltage tokamak startup and current ramp up is very essential for large size fusion devices, especially for the superconducting tokamak having limited induced electric field. Pre-ionization using Electron Cyclotron Resonance (ECR) has been reported in many tokamaks [12-14]. In these experiments, the breakdown voltage was successfully reduced typically up to 50% to 70% with RF compared to those without RF. Breakdown and current ramp up at a minimum electric field of 0.15 Vm^{-1} was achieved in DIII-D [15]. Similar experiments have been carried out in ADITYA. The substantial reduction in breakdown voltage with ECR, as low as ~ 7 V is achieved as compared to ~ 20 V, typically required without RF [16].

Furthermore, a large collection of ADITYA discharges have been analyzed for different types of spontaneous and deliberately-triggered disruptions using data mining programs [17]. A thorough disruption database has been generated by identifying the different categories of disruption. The most common cause of disruption in ADITYA is found to be due to the generation of MHD instabilities. The current quench time versus edge safety factor (q) study has been reported.

This article presents an overview of the recent experimental results from ADITYA. The experimental set up is described in Section 2, the main experimental results are discussed in Section 3 and the paper is summarized in section 4.

2. Experimental set-up

The experiments reported in this paper have been carried out in ADITYA, which is a medium size air-core tokamak with a Graphite circular limiter having major radius, R = 75 cm, and minor radius, a = 25 cm [18]. Typical plasma parameters for the discharges presented in this article are $B\varphi \sim 0.75-1.26$ T, peak loop voltage ~ 20 V, plasma current ∼100–160 kA, discharge duration ∼100 ms–250 ms, chord averaged central electron density $n_e \sim 1.5 - 3.8 \times 10^{19}$ m^{-3}, central electron temperature $T_e \sim 300 - 700$ eV, and edge safety factor q ∼3−4. The plasma is generated in a stainless steel vessel, normally evacuated to a base pressure of ~ 1.0 x 10^{-7} torr. The working plasma fueling gas is hydrogen is filling at a pressure of 8 x 10^{-5} torr – 1.0 x 10^{-4} torr. The Piezo-electric valve is installed at one of the bottom port to control the fuel gas pressure operated in a pulsed (Pre-fill) gas feed mode. In the pulsed gas feed, a square pulse is applied through pulse generator with the time-delay of 300 ms prior to the loop voltage is established [19]. The pre-ionization filament kept on with filament current 19 A and bias voltage 150 V during the plasma experiments. To achieve better wall conditioning in ADITYA, hydrogen DC glow discharge cleaning (GDC) assisted solid target lithiumization was carried out on a regular basis before operation. The GDC system consists of two UHV bellow driven movable electrodes, placed diametrically opposite at the center of the vessel [20], which have a positive potential act as anodes and vacuum vessel being kept as negative potential act as a cathode. The GDC was carried out at higher hydrogen gas pressure of ~ 8 x 10^{-4} to 1 x 10^{-3} torr and automated for maximum 12 h in absence of toroidal magnetic field. The solid target Lithiumization has been performed in ADITYA with two lithium rods of diameter 12 mm, toroidally 180° apart, were inserted at 20 mm inside the plasma volume. The partial pressure of various Mass spices like H_2, O_2, CO, N_2, CH_4 and H_2O were monitored on regular basis with quadruple mass analyzer (QMA). The multiple gas puffs was introduced in the vessel by using a Piezo-electric valve (500 SCCM at 100 V). A programmable pulse generator is used for multiple gas puff (for hydrogen and Neon both) to control the fuel gas.

In ADITYA, all the magnetic coils are powered by computer-controlled convertor pulsed power supplies and all data are stored in CAMAC/PXI based data acquisition system. The ohmic transformer coils were charged with negative converter power supply (-14 kA ohmic current) along with positive converter (+ 14 kA ohmic current) to generate the sufficient volt-sec for enhancing the plasma pulse length. The higher plasma current ramp rate (4 MA/sec) was achieved by operating IGBT based 1600 V Booster power supply assisted faster vertical magnetic field having faster response time (< 0.1 ms) for voltage rise. The equilibrium is achieved by two pairs of vertical magnetic field coils (BV1 & BV2) placed outside the vessel. To correct the radial plasma position shift on a faster time scale, a set of external coils named Black Correction Coil (BCC) is used in vertical field mode configuration to control the radial plasma position. A pre-programmed current pulse (150 A, 125 ms) generated with capacitor bank power supply is used to power the BCC for controlling the radial plasma position.

The main set of diagnostics used in these experiments includes external magnetic sensors to measure loop voltage (V_{loop}), plasma current (I_P) and plasma position. Eight-channel microwave interferometer is used for line average electron density (ne) measurements. Neutral (Hα) and impurity line radiation (O-I, C-III, visible continuum) are measured using visible spectrometer and photomultiplier tubes (PMTs) with wavelength filters. The central electron temperature is measured using Soft x-ray detectors. The ion temperature is measured with Charge Exchange Diagnostic (CXD) [21]. The total radiated power is measured using AXUV Bolometer [22] and plasma stored energy is measured using diamagnetic loop [23]. The MHD oscillations are measured by a garland of 16 Mirnov coils distributed at equal angular separations in the poloidal direction at a single toroidal location [24]. The hard x-ray flux measurement is carried out by using two NaI (Tl) scintillator detectors with diameters of 1.5 and 3 inch working in a current mode with PMT readout. Excellent images at high spatial and temporal resolution were obtained by fast visible imaging wide angle video camera. Electron Cyclotron Resonance Heating (ECRH) system is of 42 GHz frequency and 500 kW of RF power generated with gyratron in second harmonic X-mode.

3. Results and discussion

3.1. High Current Long Pulse Plasma Discharge Operation

In recent operational campaign, special efforts are made to enhance the plasma parameters to achieve ohmic discharges with improved confinement. Further, the discharges have been tailored for different experiments in order to obtain better results in those experiments. Before starting the high performance discharges, ADITYA vacuum vessel was successfully baked up to ~115 °C and achieved base vacuum of the order of ~ 3 x 10^{-8} torr on a regular basis. The control of impurities and hydrogen recycling is very much essential for high performance discharges. To achieve better wall conditioning, hydrogen DC glow discharge cleaning (GDC) assisted solid target lithiumization and limiter baking at ~ 100° C was carried out on a regular basis before everyday operation. The significant reduction in H_α, O-I, C-III and Vis-continuum impurity line radiation signals and partial pressure of various Mass spices like H_2, O_2, CO, N_2, CH_4 and H_2O were observed after extensive wall conditioning. The time evolution of typical high current long pulse ADITYA discharges is shown in Figure 1. The Figure shows the repeatable plasma discharges of maximum plasma current of ~ 150 kA and discharge duration beyond ~ 200 ms with plasma current flattop duration of ~ 140 ms. In addition to that very good discharge repeatability has been achieved in a wide range of toroidal magnetic field as shown in Figure 2. The minimum loop voltage of ~ 1.6 V is achieved in many shots during plasma current flattop. The ohmic converter provides the necessary loop voltage to manage the plasma in which ohmic current interruption by VCB inserting different resistors in steps to generated required loop voltages for plasma breakdown, startup and current ramp up. Then, current comes to zero and swing to negative by firing negative side rectifier. To achieve higher plasma current beyond 150 kA, the loop voltage was properly shaped. The Booster power supply assisted vertical magnetic field provided higher dI/dt to the BV coils, support the faster plasma current rise and the peak plasma current of ~160 kA has been reached in 40-50 ms time.

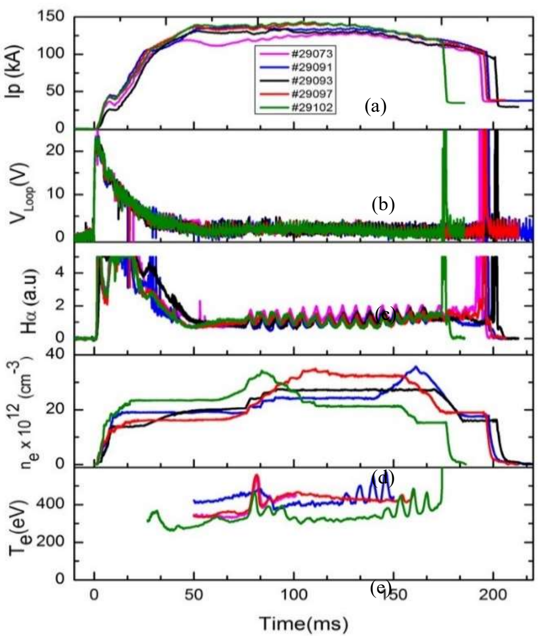

FIG. 1. The time evolution of typical ADITYA discharges (a) Plasma current (kA), (b) Loop Voltage (V), (c) H_α line emission, (d) line average electron density (n_e), (e) Central electron temperature (eV).

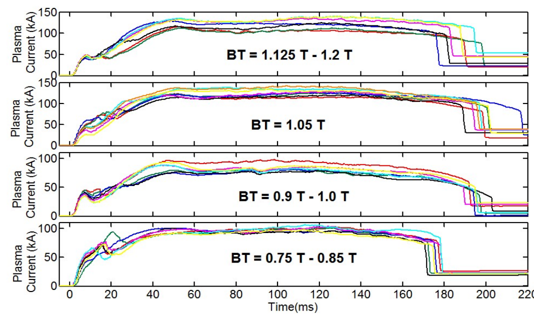

Fig. 2. Discharges repeatability at different toroidal fields.

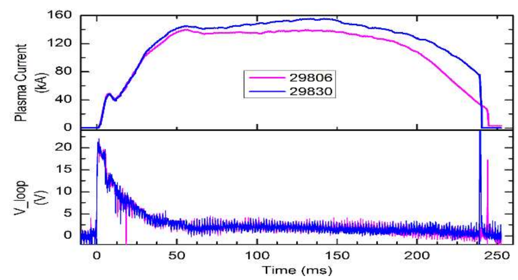

Fig. 3. Time evolution of typical high current long duration discharges (a) Plasma current (kA), (b) Loop Voltage (V).

To avoid q-limit disruption at higher I_P, the toroidal magnetic field was raised sufficiently higher up to ~ 1.26 T to maintain edge safety factor (q) ~ 3.

3.2. Radial Plasma Position Control

The ohmic transformer coils maintain sufficient volt-sec for stretching the plasma current pulse up to 250 ms as shown in Figure 3. In order to utilize the full volt-sec, plasma position control is very essential to have a repeatable longer duration discharges. The time evolution of typical ADITYA discharge with radial plasma position control with external pulse is shown in Figure 4. In recent –Ve converter power supply assisted high performance discharges (I_p ~ 100-160 kA, time (ms) ~ 200–250 ms), slight drop in loop voltage is observed during converter transition phase (Figure 4(a)) at around 80 ms, which leads to the drop in plasma current (Figure 4(b)). The actual equilibrium field could not drop so faster to response this sudden change in I_p

because of large inductance offered by BV coils (Figure 4(c) – blue curve). As a result, the horizontal plasma position moves inboard (Figure 4(d)) and if left uncontrolled, sometimes causes plasma disruption and perturb the repeatability of plasma discharges. In order to correct the horizontal plasma position shift on a faster time scale, a set of external coils named Black Correction Coil (BCC) is used in vertical field mode configuration to control the radial plasma position in ADITYA. A pre-programmed current pulse (150 A, 125 ms) (Figure 4(c) - red curve) generated with capacitor bank power supply is used to power the BCC for controlling the radial plasma position. Due to this external pulse, the plasma position shifts outboard (Figure 4(d) and obtained longer duration discharges beyond 200 ms.

Fig. 5(a) *Fig. 5(b)* *Fig. 5(c)* *Fig. 5(d)*

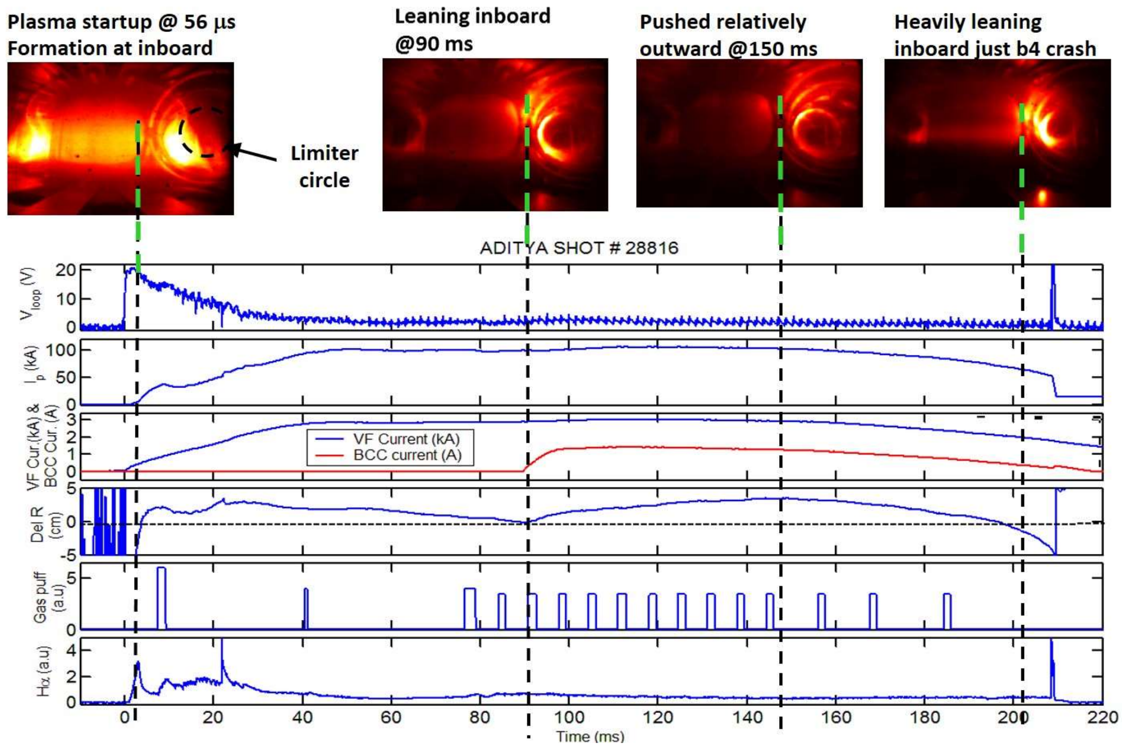

FIG. 4. The time evolution of typical ADITYA discharges shot # 18816 (a) Loop voltage (V) (b) Plasma current (kA), (c) Vertical field current (kA) - blue curve, external current pulse for position control – red curve, (d) Horizontal plasma position (e) Multiple Hydrogen gas puff (a.u) (f) H_α line emission (a.u).

The two dimensional tangential viewing fast visible imaging video camera installed on ADITYA captured wide angle panoramic view of tokamak from radial port. The entire poloidal cross section including limiter is within the field of view of camera. Data acquired at 14 kHz, consecutive frames are 71 microseconds apart. The excellent images of plasma evolution at high spatial and temporal resolution are obtained and shown in Figure 5(a), 5(b), 5(c) and 5(d). The plasma evolution takes place at inboard limiter (Figure 5(a)) confirmed with plasma position measurement. Again at 90 ms time, plasma position shifts towards inboard limiter (Figure 5(b)). The image taken at 150 ms (Figure 5(c)) is shown movement of plasma position at outboard limiter. At the end, just before the disruption, plasma again moved towards inboard limiter (Figure 5(d)). Thus, repeatable loner duration discharges are obtained with precisely controlling the radial plasma position.

3.3. Electron Density and Temperature enhancement

It is well known fact that puffing of the working gas has been a standard tool in tokamak experiments for increasing mean plasma density and improve confinement. During the past years of operation of ADITYA, many fuelling techniques (gas puffing, supersonic molecular beam injection (MBI)) have been studied to reach average densities beyond the Greenwald limit [25-26]. The multiple hydrogen gas puffing at the plasma edge as shown in Figure 4(e) is introduced in the vessel to increase the plasma density. The amount of injected gas is controlled in such a way that no significant change occur in the plasma current and its equilibrium position. The pulse widths timing and voltage level, time (T) for gas-puff to start, number of pulses and the time gap between the pulses are varied with pre-programmed gas puff according to the discharge requirements. A plot of chord average electron density versus shot label, estimated with microwave interferometer and central electron temperature versus shot label, estimated with Soft X-rays diagnostic for selected high confinement discharges are shown in Figure 6 (a) and 6(b) respectively. These plots showed that a significant number of discharges having higher density (2.7 -3.8 x 10^{19} m^{-3}) and higher electron temperature (500 eV -700 eV) have been achieved in ADITYA.

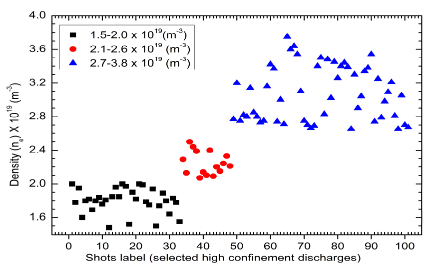

FIG. 6(a). The electron density distribution for selected shots versus shot lable.

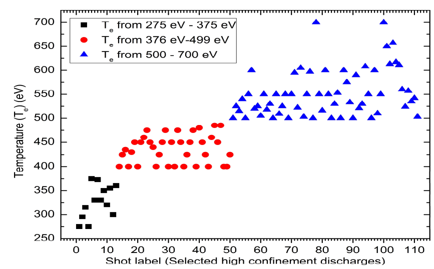

FIG. 6(b). The electron temperature distribution for selected shots versus shot label

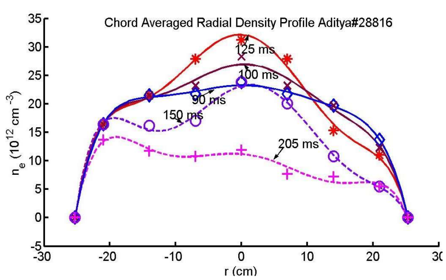

FIG. 7(a). The electron density profile with microwave as a function of radial distance.

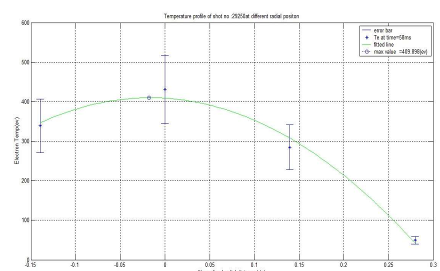

FIG. 7(b). The electron temperature with Soft X-rays profile as a function of radial distance.

The radial electron density profile and central temperature profile is shown in Figure 7(a) and 7(B) respectively. The density and temperature profile is estimated using the same diagnostics. For ADITYA parameters, the density profile index $n = n_0 (1- r^2 / a^2)^{\alpha}$ and temperature profile index $T_e = T_{e0} (1 - r^2 / a^2)^{\alpha}$, where $\alpha = 1$ and 1.75 has dependence for both density and temperature profile respectively. The maximum electron density $\overline{n}_e = 6.0 \times 10^{19}$ m^{-3} has been obtained and it is quite close to Greenwald density limit for ADITYA.

3.4. Energy Confinement time Enhancement

The estimation of global Energy confinement time (τ_E), is one of the most important characteristics parameter of a tokamak plasma, as all the tokamaks aim to achieve higher energy confinement time. The scaling of the energy confinement time with density constitutes one of the basic elements in the development of tokamak devices to a fusion reactor. The neo-ALCATOR scaling law for ohmically heated plasma discharges at not very high density was established from a large number of plasma discharges from different tokamaks in order to get a better insight into various operational limitations of tokamaks with varied parameters. The neo-ALCATOR scaling law is

$$\tau_E = 7 \text{ x } 10^{-2}\, n_e\, a\, R^2\, q_{eff} \qquad (1)$$

Where τ_E is measured in seconds, a- minor radius, R-major radius measured in metres, n_e is plasma density measured in 10^{20} m^{-3}, q_{eff} - edge safety factor. In ASDEX, improved ohmic confinement (IOC) regime was discovered in which linear scaling of $\tau_E \sim n_e$ was extended up to density limit [27]. Experimentally, the energy confinement time can be estimated as

$$\tau_E = \frac{3/2 < n_e T_e + n_i T_i > V}{I_p V_L - P_{rad}} \qquad (2)$$

Where Ip is the plasma current, V_L is the loop voltage, V is the plasma volume, and P_{rad} is the total radiated power. The transition to improved confinement in ADITYA is characterized by a spontaneous increase of plasma density by a factor of 2 and increase in the electron temperature by a factor of 1.5 to 2. Thus the total stored energy grows and the loop voltage and consequently the ohmic heating power drop. Both effects yield an improvement in β and the global energy confinement time. The energy confinement time analysis for large number of discharges has been carried out in ADITYA and experimentally estimated energy confinement time (τ_E) for ADITYA parameters is compared with confinement time estimated from neo-ALCATOR scaling is shown in Figure 8.

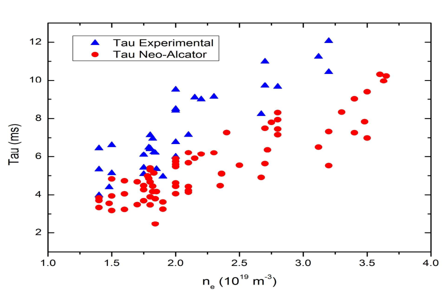

FIG. 8(a). τ_E experimental and neo-ALCATOR versus density (n_e) plot.

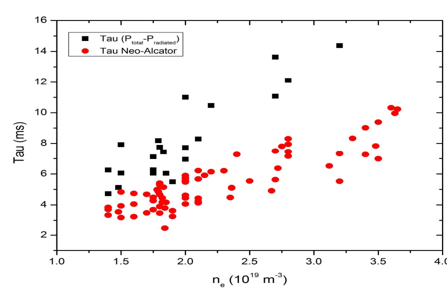

FIG. 8(b). τ_E expt. and neo-ALCATOR versus density (n_e) plot with $P_{total} - P_{rad}$.

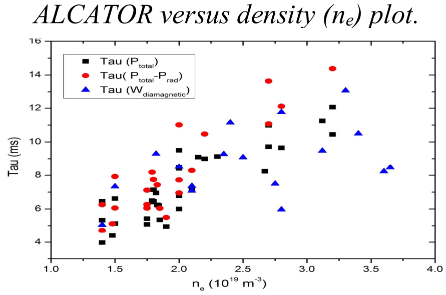

FIG. 8(c). τ_E expt. versus density (n_e) plot for P_{total}, $P_{total} - P_{rad}$ and W_{dia}.

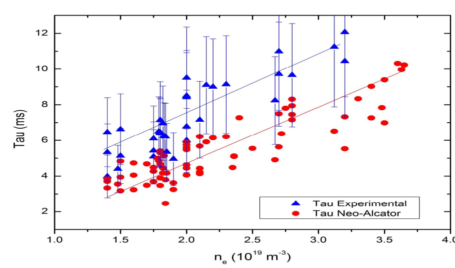

FIG. 8(d). τ_E expt. and neo-ALCATOR versus density (n_e) plot with error bars.

The results demonstrated in Figure 8 reveals that we have been able to achieve confinement times evidently higher than that predicted by neo- ALCATOR scaling almost >1.5 times higher. Also a good number of discharges lie in IOC regime.

3.5. ADITYA Operation Parameters Space

The operating space is restricted by several limitations among which the plasma performance has to be optimized. Plasma discharges in a tokamak can be realized only within a definite range of densities [28]. At a given plasma current there exist a lower and an upper density limit. The lower density limit leads to the generation of a runaway electrons. The upper density limit, for a given plasma current, there exist a maximum line average density i.e. Greenwald limit, which is defined as $n_G[10^{20}]$ = Ip [MA]/πa2 [m^2] for circular machines due to radiation loss and recycling at higher Z_{eff}. However, over the past years the limit has increased due to the application of advanced wall conditioning and better fueling techniques. The plasma current limit is due to the MHD instabilities, where strong MHD oscillations accompanied by shrinkage of the current channel, inferred from changes in plasma induction, were seen as the disruptive limit was approached. The 'Hugill' plot is a conventional means of presenting disruptive limit for high current (low q) operation as well as the density limit and became a standard method of displaying the operating space for tokamaks. The 'Hugill' diagram for improved ohmic confinement discharges of ADITYA is shown in Figure 9 is a plot of the inverse safety factor at the edge, $1/q_a$, versus the Murakami number, $n_e R/B_T$. In cylindrical geometry the edge safety factor is given by $q_{edge} = 5 a^2 B_T / I_p{*}R$. Upper line at 1/q = 0.5 is safety factor limit beyond this kink instability kicks in. The right lower limit is maximum density limit i.e. Greenwald limit.

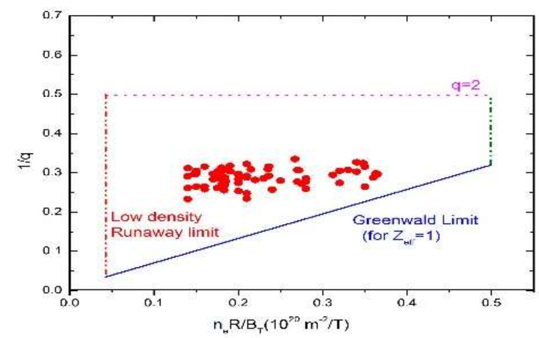

FIG.9. 'Hugill' plot describing operating space for ADITYA discharges

The above plot clears that, we have been able to attain density quite close to this Greenwald limit with improved lithium wall conditioning, radial plasma position control and optimized adjustment of gas puffing. The left most limit is the runaway density limits, below which we strike runaway discharges.

3.6. Radiative Improved Confinement Mode (RI mode) experiment

Experiment in the ADITYA tokamak was carried out to enhance the confinement through the puffing of neon gas into the plasma. This improved confinement region is known as Radiative Improved (RI) mode, which has been observed in many tokamaks [29-30]. It was discovered in the ISB-X [31] and thoroughly investigated in TEXTOR-94 [32]. It is believed that improved confinement in the RI mode is mostly based on the reduction of growth characteristics of the toroidal ion temperature gradient (ITG) mode due to the increase of Z_{eff} and also because of the suppression of turbulence due to increase of E×B shear rotation in the impurity injected plasma [33-34]. In ADITYA, neon gas was puffed during the current flat-top region using gas fuelling system discussed previously. The gas valve is located on the bottom port and 90^0 toroidally from the limiter. The pulse widths timing and voltage level, time for gas-puff to start, number of pulses and the time gap between the pulses was varied during the neon gas puff experiment. The Figure 10 shows the temporal evolution of the plasma current, loop voltage, chord average electron density (n_e), electron temperature (T_e), $H\alpha$ emission, radiated power (P_{rad}) and energy confinement time (τ_e), for discharges with and without neon gas puff. It can be seen clearly from Figure 10 that the n_e, T_e, and P_{rad} increases after the application of neon gas puff from 98 to 108 ms as depicted by shaded rectangle in Figure 10(d). Simultaneous decrease in $H\alpha$ signal and increase in n_e indicates better particle confinement after the neon gas puff. The τ_e was improved by a factor of 2 from 6.5 to 13 ms as shown in Figure 10(d) and the transition of energy confinement happens at 117 ms.

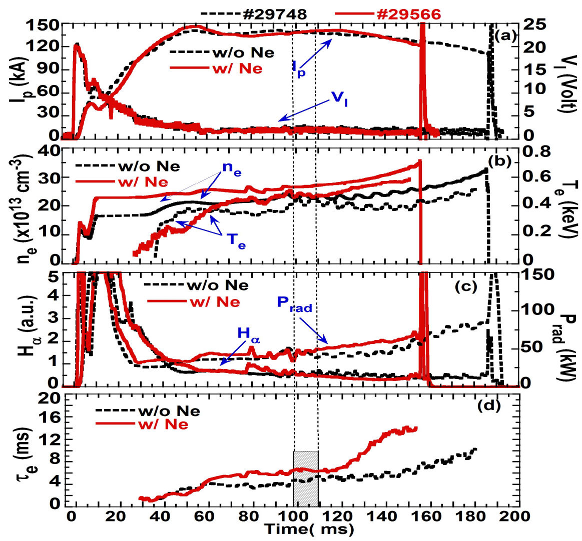

FIG. 10. The time evolution of typical ADITYA discharges shot # 29748 without Neon gas puff (black curve) and shot # 29566 with Neon gas puff (red curve) (a) Plasma current (kA), loop voltage (V) (b) line average electron density (n_e), central electron temperature (keV) (c) H_α line emission, total radiated power and (d) energy confinement time (Sec), Neon gas puff.

3.7. Disruption Characterization Studies in ADITYA

Disruptions are events in which large fraction of the plasma thermal energy is lost in very short time causing enormous damage to the plasma facing components in tokamaks. Dedicated experiments on disruption characterization and mitigation [7] have been carried out in ADITYA. A large collection (~ 17,000) of ADITYA discharges have been analyzed for different types of spontaneous and deliberately-triggered disruptions. To identify the disrupted discharges from the large collection of ADITYA archive, the electronic database code has been developed. Firstly, the statistical analysis of disruption has been carried out and the overall ADITYA disruption rate has been determined. The averaged over all discharges in collection from calendar year 2004-2015, the total disruptivity is found to be ~ 21% augmented with 5% deliberate disruptions for experiment and research purpose, which is quite similar to that observed in other tokamaks. Similar work has been carried out for JET disruptive discharges [35]. Furthermore, the disruptions in ADITYA discharges have been characterized into four major categories, disruption caused by MHD instabilities, density limit disruption, q limit disruption and equilibrium failure. Majority ~85% ADITYA disruptions are observed due to MHD growth. The quench caused by edge cooling which leads to interaction between magnetic islands with a dominant role of the m=2, n=1 island. MHD disruption are mostly observed with lock mode as shown in Figure 10. After identification and categorization of disruptive discharges of ADITYA, a database is generated as per ITPA suggested format and have been uploaded to ITPA disruption database working group [17].

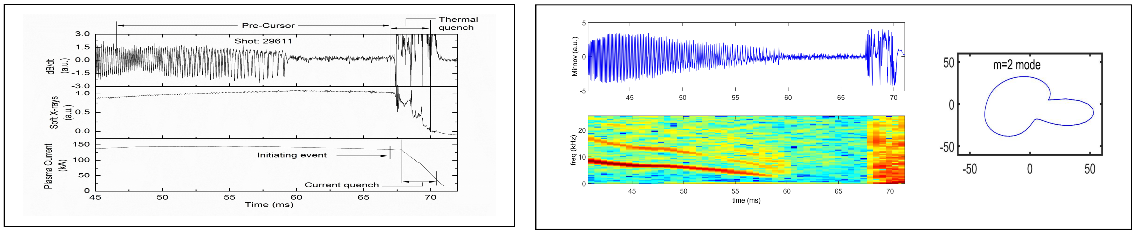

FIG.10. ADITYA shot # 20611, the typical m=2, n=1 lock mode disruption of ADITYA

The key parameters of the ITPA disruption database of plasma current quench timings such as TIME8 (time at 80% of the I_{pd}) and TIME2 (time at 20% of I_{pd}) have been consider for our present study. A detailed analysis of several hundred disrupted discharges showed that the current quench time is inversely proportional to edge safety factor (q_{edge}) as shown in Figure 11. The plasma current decays faster in discharges having higher edge safety factor (q_{edge}). This is due to higher growth of MHD islands in high qedge discharges.

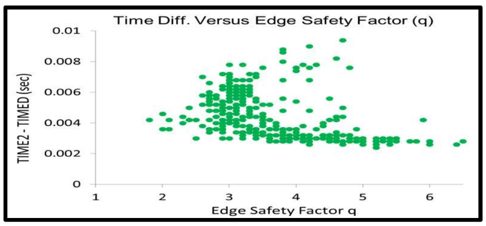

FIG. 11(a). Time difference (Time2-TimeD) (S) versus edge safety factor (q_{edge}).

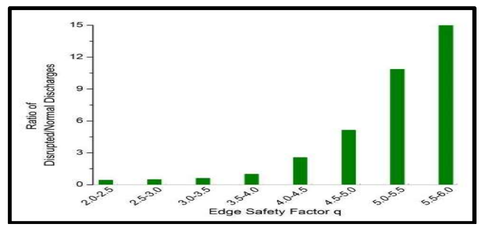

FIG. 11(b).Ratio of disruptive to normal discharges versus edge safety factor (q_{edge}).

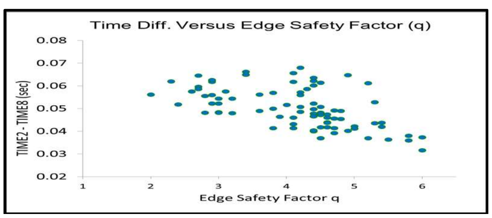

FIG. 11(c). Time difference (Time2-Time8) (S) versus edge safety factor (q_{edge}).

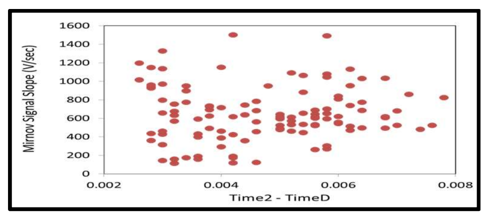

FIG. 11(d). MHD growth slope (V/S) versus time difference (Time2-TimeD)

The behaviour of island width (W) and Location (r_s) for Lock Mode Disruption has been studied. The Island width for both m = 2 and m = 3 increases with the edge safety factor. The island m = 2 and m = 3 are also moving towards the core with the increase in edge safety factor. The magnetic island width (W) and the location (r_s) do have a significance in the current quench procedure. The transport of the particle and the energy will be faster in the case where the edge safety factor is more with bigger islands are present near to core. This reveals, low disruption time at higher edge safety factor for ADITYA.

4. Summary

Several experiments, related to controlled thermonuclear fusion research and highly relevant for large size tokamaks including ITER, have been carried out in ADITYA. Repeatable plasma discharges of maximum plasma current of ~ 160 kA and discharge duration beyond ~ 250 ms with plasma current flattop duration of ~ 140 ms has been obtained for the first time in ADITYA. The peak electron density (ne) ~6 X 10^{19} m^{-3} and the maximum electron temperature (Te) ~700 eV have been achieved. The experimentally estimated energy confinement time (τ_E) for ADITYA parameters is compared with confinement time estimated from neo-ALCATOR scaling showed ~ 1.5 times higher confinement times than that predicted by neo- ALCATOR scaling. The Hugill plot for ADITYA operating parameter space showed that we have been able to attain density quite close to this Greenwald limit with improved lithium wall conditioning, radial plasma position control and optimized adjustment of multiple gas puffing. In addition to that Neon gas puff assisted radiative improved confinement mode has been observed in ADITYA. The energy confinement time improved by a factor of 2 in discharges with Neon gas puff. Detailed analysis of several disrupted discharges of ADITYA showed that the current quench time is inversely proportional to edge safety factor (q_{edge}).

References

[1] CHOWDHURI, M. B., et al., "Improvement of Plasma Performance with Lithium Wall Conditioning in Aditya tokamak', Plasma Science & Technology, Vol. 15, No.2, 2013.

[2] BALAKRISHNAN, V., et al., "Plasma Current and position feedback control in ADITYA Tokamak", Fusion Engineering & Design", Vol. 66-68, Sept. 2003, 809-813.

[3] MAJESKI, R., et al., "Enhanced Energy Confinement and Performance in a Low Recycling Tokamak", Physics Review Letters 97, 075002 (August 2006).

[4] SOLDNER, F. X., et al., "Improved Confinement in High-Density Ohmic Discharges in ASDEX", Physics Review Letters, Vol. 61, No. 9 (August-1988).

[5] ESPOSITO, B., et al., "Transport analysis of ohmic, L-mode and improved confinement discharges in FTU", Plasma physics and controlled Fusion, Vol. 46, No. 11, (Sept.-2004).

[6] DHYANI, P., et al., "A novel approach for mitigating disruptions using biased electrode in ADITYA tokamak", Nuclear Fusion, 54 (June-2014) 083023.

[7] TANNA, R.L., et al., "Novel approaches for mitigating runaway electrons and plasma disruptions in ADITYA tokamak", Nucl. Fusion 55 (May-2015) 063010 (5pp)

[8] UNTERBERG, B., et al., "Plasma wall interaction and plasma edge properties with radiation cooling and improved confinement in TEXTOR-94", Journal of Nuclear Materials 266-269 (1999), 75-83.

[9] WEYNANTS, R.R., et al., "Overview of radiative improved mode results on TEXTOR-94", Nuclear, Fusion, Vol. 39, No. 11Y, (1999), 1637-1648.

[10] MESSIAEN, A.M., et al., "Improved confinement with edge radiative cooling at high densities and high heating power in TEXTOR", Nucl. Fusion, Vo1.34, No.6 (1994), 825.

[11] MESSIAEN, A.M., et al., "Transport and improved confinement in high power edge radiation cooling experiments on TEXTOR", Nucl. Fusion, Vol. 36, No. 1 (1996), 39-53.

[12] TANAKA, S., et al., "Initiation of plasma current with the assistance of electron cyclotron waves in the WT-3 tokamak", Nuclear Fusion, Vol. 33, No.3 (1993), 505.

[13] GILGENBACH, R.M., et al., "Electron cyclotron/upper hybrid resonant pre-ionization in the ISX-B tokamak", Nuclear Fusion, Vol. 21, No.3 (1981), 319.

[14] KAJIWARA, K., et al., "Electron cyclotron heating assisted startup in JT-60U", Nuclear Fusion 45 (2005) 694–705.

[15] LLOYD, B., et al., "Low voltage Ohmic and electron cyclotron heating assisted startup in DIII-D", Nuclear Fusion, Vol. 31, No. 11, (1991), 2031-2053.

[16] SHUKLA, B. K., et al., "ECRH assisted plasma experiments on Tokamaks SST-1 and ADITYA", 26th IEEE Symposium on Fusion Engineering (SOFE), May 31-June 4, 2015, Austin, Texas USA, DOI:10.1109/SOFE.2015.7482270.

[17] EIDIETIS, N.W., et al., "The ITPA disruption database", Nucl. Fusion 55 (2015) 063030.

[18] BHATT, S.B., et al., "ADITYA: The First Indian Tokamak", Indian J. Pure Appl. Phys. 27, 710 (1989).

[19] TANNA, R.L., et al., "Influence of Wall Conditioning on ADITYA Plasma Discharges", Journal of Physics, conference series, 390 (2012) 012044.

[20] PATHAK, H.A., et al., "Glow discharge wall conditioning of tokamak ADITYA", Nuclear Materials, 220 -222 (1995) 708- 711.

[21] PANDYA, S. P., et al., "Core-ion temperature measurement of the ADITYA tokamak using passive charge exchange neutral particle energy analyzer", Rev. Sci. Instrum. Vol. 84, Issue 2, p023503 (February-2013)) 10.

[22] TAHILIANI, K., et al., "Radiation power measurement on the ADITYA tokamak", Plasma Physics Control. Fusion 51 (2009) 085004 (13pp).

[23] SAMEER KUMAR, et al. "Diamagnetic flux measurement in Aditya tokamak", Rev. Sci. Instrum., Vol. 81, 123505 (2010).

[24] Raju D. et al 2000 "Mirnov coil data analysis for tokamak ADITYA" Pramana-J. Phys. 55, 727, (6pp).

[25] BHATT, S.B., et al., "Gas puffing by molecular beam injection in Aditya tokamak", Fusion Engineering and Design, 75-79 (2005), 655-661.

[26] GREENWALD, M., et al., "Density limits in toroidal plasmas", Plasma Physics Control. Fusion 44 (2002) R27–R80.

[27] KLUBER, O., et al., "High-density tokamak discharges in the PULSATOR device with $\beta p > 1$", Nucl. Fusion 15, 1194 (1975).

[28] KOSLOWSKI, H. R., et al., "Operational limits and limiting instabilities in tokamak machines", Transactions of fusion science and technology, Vol. 49, 2006, pp. 147-154.

[29] VANDENPLAS, P. E., et al., "Review and present status of the TEXTOR radiative improved (RI) mode", J. Plasma Physics (1998), vol. 59, part 4, pp. 587–610.

[30] WU, Z. W., et al. "Radiative improved mode in the HT-7 tokamak", Journal of Nuclear Materials, 438 (2013) S576-S579.

[31] LAZARUS, E.A., et al., "Confinement in beam-heated plasmas: the effects of low-z impurities", Nuclear Fusion, Vol.25, No.2 (1985), 135-149.

[32] MANK, G., et al., "Quasistationary high confinement discharges with trans-Greenwald density on TEXTOR-94", Physics Review Letters, Vol. 85, No. 11 (2000), 2312-2315.

[33] TOKAR, M.Z., et al., "Model for transition to radiatively improved mode in a tokamak", Physics Review Letters, Vol. 84, No. 5 (2000), 895-898.

[34] TOKAR, M.Z., et al., "Confinement mechanisms in the radiatively improved mode", Plasma Physics Control. Fusion 41 (1999) B317–B32.

[35] DE VRIES, P.C., et al., "Statistical analysis of disruptions in JET", Nuclear Fusion 49 (2009), 055011, (12pp).