Recent ICRH-Wall Conditioning, Second Harmonic Heating and Disruption Mitigation Experiments using ICRH System in Tokamak ADITYA


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Abstract: Here we report the recent experiments carried out on tokamak ADITYA using the developed ICRH system of 1 MW at 24.8 MHz frequency corresponding to the second harmonic heating at 0.75 T. The experiments are carried out to have plasma heating at second harmonic, disruption mitigation and also wall conditioning (ICWC) in presence of toroidal magnetic field. The wall conditioning experiments are carried out in presence of toroidal magnetic field under resonant (0.75T), non-resonant (0.45 T) conditions as well as with 20% He gas in a hydrogen plasma (0.45T). All three sets are found more effective in releasing wall impurities like water & methane as half an order (~ 5) of initial vacuum condition. As per data, the resonant ICWC is more effective to reduce carbon impurity and non-resonant ICWC is more effective to reduce oxygen impurity from vessel. The heating experiments at second harmonic are carried out using RF pulses of different magnitudes (5 ms-100 ms) at different RF powers (40 kW-200 kW) in plasma duration of 100 ms. The soft X-ray data shows an electron temperature rise from 250 eV to maximum of 500 eV and NPA data as well data from Doppler broadening shows the ion temperature rise up to 350 eV. In order to carry out mitigation of disruptions induced by hydrogen gas puff, ICRH system was used in both fixed and real time feedback mode. In an attempt to control the disruptions in real time the gas-puff induced H_α intensity increase is used as a precursor for the disruption and mitigation is successfully carried out by introducing RF power to extend the plasma current.

1. Introduction

Introduction of RF waves in plasma in ICRH range has many applications starting from pre-ionization, current ramp up, fast wave and ion Bernstein wave heating, stabilization of some of the modes like ELMs in the plasma, vessel wall conditioning, with and without the presence of resonance layer in presence of toroidal magnetic field, current drive via momentum transfer from wave to the plasma etc. With the help of ICRH waves one can do either electron heating, ion heating or both [1] to increase the temperature and pressure of the plasma. Since Ohmic system has limitations to increase plasma above 1 keV, ion cyclotron heating system is an integral part of the fusion tokamaks to reach the fusion temperature and cyclotron heating at second harmonic seems to be most effective as well as bit easy technique due to availability of high power RF technology at a relative lower cost. Plasma heating by fast magneto-sonic waves in the ion cyclotron resonance frequency (ICRF) range at second
harmonic is an established method of radio frequency heating in tokamaks and other devices [2,3] and is found to be quite successful. The fast wave (FW) is a compressional wave that propagates primarily perpendicular to the static toroidal magnetic field [4]. In most cases, after tunnelling a thin evanescent layer from the low magnetic field side, it propagates to the cyclotron resonance layer as defined by the launched RF frequency and static toroidal magnetic field. The wave energy is absorbed either directly by ions through cyclotron absorption, or indirectly via mode conversion in a single or multi-species plasma. FWs have also been seen to transfer their wave energy to electrons broadly by two mechanisms.

The first one is indirect electron heating, where resonating majority or minority ions pick up energy from waves and transfer their energy to electrons via collisional equi-partition. The second mechanism is a direct electron heating where FW is damped on electrons via electron Landau damping (ELD) and transit time magnetic pumping (TTMP) [5-7]. Both direct and indirect electron heating is observed in JET [8] and DIII-D [9] tokamaks with various minority heating scenarios, namely H in D, 3He in 4He and 3He in D plasmas. In DIII-D experiments, the effectiveness of electron heating is found to increase with target electron temperature, which is in the range 0.7–1.3 keV. In TFTR, direct electron heating by FW is observed in the 3He minority regime with D plasma heated with neutral beam injection (NBI), but without any 3He present in the plasma [9]. In their experiments, the only ion resonance is the D fundamental at the high-field side. In all these experiments, the target plasma is either of two species or preheated with NBI or EC to enable the high-energy tail electrons to interact with FWs. To the best of our knowledge, direct electron heating in a low-temperature (<500 eV) single species plasma (H2) in a medium-sized tokamak is not observed elsewhere.

ICRF plasma production and its assisted low voltage ohmic start up have been demonstrated in TEXTOR-94[7]. Interestingly without ICRF assistance no start up is achieved on application of low loop voltage. The plasma production through ICRF is believed to be mainly because of absorption of RF wave energy by electrons in presence of toroidal magnetic field. The RF electric field parallel to magnetic field may be responsible for this neutral gas breakdown and initial plasma build up. The parallel electric field may be generated between antenna central strap and sidewall of the antenna. Recently, we have carried out the experiments on pre-ionization and current build-up at different loop voltages, pressures, magnetic fields and RF powers using the developed ICRH system of 200 kW on tokamak ADITYA [10].

In a superconducting tokamak the normal glow discharge cleaning becomes very difficult and one needs to find alternatives which will work in presence of toroidal magnetic field. Wall conditioning using ion cyclotron waves is one of the prominent candidates for wall conditioning in presence of toroidal magnetic field and seems to be much more efficient than the conventional discharge cleaning techniques like glow discharge, pulse microwave discharge cleaning etc. in absence of toroidal magnetic field. ICRF-DC has been developed on the circular tokamaks TEXTOR, TORE SUPRA and HT-7 using the present generation ICRF antennas without any modifications in hardware [11].

We also carried out the experiments of disruption mitigation using novel technique of using ion cyclotron waves in tokamak ADITYA. In this experiment we produced density disruption using gas puff and then we could get original plasma current using feedback from H alpha signal to start RF power during disruption. This is a novel technique and one can explore it further with detailed experimentation.
In short, if one has ion cyclotron system installed on tokamak, one can do pre-ionization, current ramp up, heating, disruption mitigation as well as wall conditioning experiments which are the basic needs of fusion reactors.

2. ADITYA Tokamak

ADITYA is a medium size tokamak with major radius 0.75 m and minor radius of 0.25 m, with toroidal magnetic field up to 1.5 T and has circular plasma with graphite limiter and most of the time the tokamak is operated with hydrogen gas. The diagnostics used in ICRH experiments are Langmuir probes, visible camera, spectroscopy, soft X-ray and hard X-ray detection techniques, diamagnetic loop, heterodyne on-line density measurements, Thompson scattering, neutral particle analyzer, limiter thermography, Residual Gas Analyzer (RGA) along with normal machine diagnostics.

3. Experimental Details

A system of plasma heating by poloidal type fast wave antenna is currently being employed on Aditya tokamak [10,12-14]. The ICRF system on ADITYA is of 20-40 MHz frequency range and 1 MW of RF power. A simplified block diagram of complete ICRF system as well as that of vacuum interface on ADITYA is shown in Figure 1. A 100-meter long 9” co-axial copper transmission line carries RF power from generator to tokamak ADITYA. A series of SPDT switches enables to divert the RF power either towards ADITYA or towards 50 Ohm soda-water dummy load system housed in generator hall for testing purpose. A matching network consisting of two stubs and two phase-shifters is placed between the antenna and generator for matching antenna-plasma impedance to that of the 50 Ohm impedance of the generator and transmission line. A Vacuum Transmission Line (VTL) section, which has separate vacuum system, isolates the ADITYA vacuum to atmospheric pressure transmission line. The VTL section also facilitates the radial movement of antenna up to 3 cm in the scrape of layer (SOL) of ADITYA plasma.

![Figure 1. ICRF-Aditya system schematic. 1-a, 1-b: dual directional couplers, 2: SPDT switch, 3: mechanical stub, 4: liquid phase shifter, 5: liquid stub, 6: mechanical phase shifter, 7: VSWR probe section, 8: Fast wave antenna. Details of VTL section are shown in the right side of schematic.](image-url)

Here we report the recent experiments carried out on tokamak ADITYA using the developed ICRH system. The experiments are carried out to have plasma heating at second harmonic using 1 MW system at 24.8 MHz at 0.75 T and 0.82 T, disruption mitigation using RF power from poloidal fast wave antenna and also wall conditioning in presence of toroidal magnetic field. Before carrying out experiments the transmission line, antenna and vacuum interface are
conditioned with multiple short RF pulses to increase the power handling capacity and is done after tuning the complete system for a particular frequency.

**Wall Conditioning using ICRH (ICWC):** The wall conditioning experiments are carried out by giving many RF pulses from 50 ms up to 1.2 seconds and the data is recorded using RGA and spectroscopy diagnostics. The experiments are carried out with resonance layer at 0.75 T, without resonance layer at 0.45 T and with the addition of 20% He in hydrogen plasma at 0.45 T. The toroidal magnetic field has a flat top for 1 second and RF pulses of 50-60 kW at 24.8 MHz are introduced during flat top and Gas-feed pressure (continuous mode) is maintained in the range of $1 \times 10^{-4}$ to $3 \times 10^{-4}$ Torr. The relative levels of oxygen and carbon contain impurities have been measured using RGA.

To compare the effect of all three sets in wall conditioning, RGA data was analyzed before ICWC & after ICWC. We found significant change in $M_{15}$, $M_{16}$, $M_{17}$, $M_{18}$ after ICWC in all 3 sets. $M_{15}$, $M_{16}$ are carbon contain impurities, $M_{17}$, $M_{18}$ are oxygen contain impurities. Maximum $M_{18}$ (water vapor H$_2$O) was released under non-resonant condition in set – II in factor of 4.45 (Before: $3.4 \times 10^{-8}$ Torr, After: $1.5 \times 10^{-7}$ Torr) compared to sets I (factor: 4.34) & III. In set-III, the factor found was 2.54, which is very less (~ 40 %) compared to other sets I & II. Maximum $M_{16}$ (Methane CH4) was released in set-I under resonant condition in factor 5.94 (Before: $1.9 \times 10^{-9}$ Torr, after: $1.2 \times 10^{-8}$ Torr) compared to sets II (factor: 4.63) & III. In set-III, the factor found was 2.54, which is very less (~ 50 %) compared to other sets I & II. The wall conditioning experiments are carried out using RF pulses of 20 ms up to 1 seconds and it is observed that long duration pulses of seconds are more effective in wall conditioning. Figure 2 shows the typical waveforms of short and long RF pulses along with the H$_{\alpha}$ signal and vacuum level.

All three sets are found more effective in releasing wall impurities like water & methane as half an order (~ 5) of initial vacuum condition. As per data, the resonant ICWC is more effective to reduce carbon impurity and non-resonant ICWC is more effective to reduce oxygen impurity from vessel.

**Figure 2:** Typical data recorded during RF heating: (a) shows a long RF pulse of 1 second duration and Fig. 1(b) shows many RF pulses along with H alpha signal recorded and the pre-fill pressure (multiplication factor $4 \times 10^{-5}$ torr)

**Second Harmonic Heating experiment:** The heating experiments at second harmonic [15] are carried out using RF pulses of different magnitudes (5 ms-100 ms) at different RF powers (40 kW-200 kW) in plasma duration of 100 ms. The soft X-ray data shows an electron temperature rise from 250 eV to maximum of 500 eV and NPA data as well data from
Doppler broadening shows the ion temperature rise up to 350 eV. Figure 3 shows the typical data recorded of RF heating.

D-Band (140 GHz) quadrature phase Interferometer is used to measure the line averaged density of the Aditya tokamak plasma during ICRF heating. The probing RF frequency and receiver IQ detector is phase locked by the 100 MHz crystal oscillator which produces two I and Q signals are of 100 KHz each which are digitized. Since the 100 KHz Signals from IQ detector and trigger of ADC converters are locked by one reference, we can control the pass zero points of the intermediate frequency (IF) signals without any additional reference signal, i.e. relation between period of IF signal and number of digitized points inside this period is constant anytime in the absence of plasma.

By this method, we get absolute rate Pi / (digitized points) φo and can control phase shift of signal during discharge of plasma. As we have two 100 kHz signals from IQ detector (sine and cosine), the total pass zero points is twice more and finally time phase analysed is 2.5 microseconds. The software calculates time different between close pass zero points and compare it with absolute rate.

Figure 3(b) shows the on-line density measurements in presence of RF pulse and one can see an increase in density with introduction of RF pulse and is in the range of 1x10^{13}/cm^{3}.

![Figure 3: The data recorded during rf heating experiments: (a) RF power: 125 kW, RF duration: 50-85 ms (b) RF power: 96 kW, RF duration: 15 ms-100 ms](image)

Figure 4 shows the time evolution of Hα profile during ICRH heating experiments. The diagnostic consists of 512 elements linear array camera which is position aligned to view the complete poloidal cross section of plasma. The system gives time resolution of 20 msec so Hα signal is acquired to get the profile. After the ICRH power is launched, there is reduction in Hα intensity indicating the rise in plasma temperature and also shows that the position of plasma remains same.

In tokamak plasma, the main power loss channels are radiation, charge exchange neutrals and transport losses. The measurement of power losses through convection and conduction can be carried out by the thermal imaging of the Plasma Facing Components (PFCs) heated due to the direct contact by the plasma and plasma surface interaction (PSI). Thermal imaging diagnostic of these PFCs provides real time monitoring of the surface temperatures remotely with wide field of view (FOV). The ADITYA tokamak has a poloidal limiter having 14
graphite tiles as plasma facing material at one toroidal location. The thermal imaging system is deployed on the ADITYA tokamak at radial port#3 having an IR-Camera (spectral response ~3µm to 5µm range, 320x256 pixel array, temperature sensitivity ~0.025°C). The IR-camera works in the snapshot mode with frame rates 143 frames per second provides sampling time of 7 ms. The IR-camera which measures temperature up to 1200°C is located out side the vacuum vessel and a CaF$_2$ vacuum view port has been used for the IR signal transmission (>90% in 3µm to 5µm range). The system has wide FOV~22° (horizontally) x 17° (vertically) which covers inboard limiter in direct view and outboard limiter in reflected view using a stainless steel mirror (reflectivity>90%) mounted on the high field side. The system provides spatial resolution of ~1 x 1 mm$^2$ on the inboard limiter tiles and ~3 x 3 mm$^2$ on outboard limiter tiles.

Attempts have been made to trace signatures of RF heating during the discharge in which ICRH pulse is launched and observations are compared with normal plasma discharge without ICRH pulse. For this purpose tile surface-temperature behaviour of ion flow side is compared with electron flow side.

A representative RF heating plasma discharge# 27629 is analysed where ICRH pulse length is ~32 ms and launch window is starts from ~50 ms to ~83 ms. Corresponding out-board limiter surface temperature profile of tile-1 is shown in the Fig.5 and compared with normal discharge condition (a discharge without ICRH-pulse).

![Figure 4](image)

**Figure 4:** Time evolution of plasma profile during RF heating (a) RF power: 125kW, RF duration:30 ms (b) RF power : 60 kW, RF duration: 10 ms

In case of normal plasma discharge (without ICRH pulse) it has been observed that the ratio of surface temperature at ion heating-side to electron heating-side of the limiter remains <1 at any point of time during the discharge It has been observed that for the set of plasma discharges (~14 discharges) the ratio is >1 just after the introduction of RF pulse. This can be attributed to the absorption of ICRH energy by ions and increase in the perpendicular ion velocity component which can increase cross-field diffusion and this may results in deposition of higher heat flux at ion flow-side of the limiter surface compared to limiter electron flow-side. In few shots, it has been found that the electron side surface temperature also increases after application of RF-pulse.
Figure 5: Shows temporal evolution of outboard Tile-1 temperature for ion side and electron side: (a) Plasma discharge#27629 With RF-pulse (100 kW, ~50 ms to 83 ms) and (b) Plasma discharge#27696 without RF-pulse (a normal plasma discharge).

It is observed in our experiment that sometimes the reflected power is high and when it crosses the 40% value then the hard wired interlock operates and RF power does not go through fully. Hence instead of introducing RF power for longer duration in few shots we introduced many RF pulses of 5 ms on time and 5 ms off time and we observed that for plasma it looks like continuous RF but RF generator does not trip, the MHD activity does not extend further and also we noticed that the interference in other diagnostics is less (Fig. 6(b)).

Figure 6: (a) shows multiple RF pulses (100 kW) of short duration also give similar RF heating of plasma (b) shows ion temperature evolution with time for different RF powers.

Y axis: Ti in eV, X axis: time in ms

Core-ion temperature measurements have been carried out by the energy analysis of passive Charge Exchange (CX) neutrals escaping out of the ADITYA tokamak plasma using a 45° parallel plate electrostatic energy analyzer. The Neutral Particle Analyzer (NPA) uses a gas cell configuration for re-ionizing the CX-neutrals and Channel Electron Multipliers (CEMs) as detectors. Energy calibration of the NPA has been carried out using ion-source and $\Delta E/E$ of high-energy channel has been found to be ~10%.

Density mitigation: Here we report the mitigation of disruptions induced by hydrogen gas puff, using ICRH system in both fixed and real time feedback mode. After obtaining normal repetitive discharges of 100 ms durations, hydrogen gas is puffed in sufficient amount during
the plasma current plateau phase (from 40 to 45 ms) to obtain disruptive discharges. These disruptions are successfully mitigated by application of ICRH pulse of power of ~ 50 to 70 kW through a fast wave poloidal type antenna launched ~ 5 ms prior to the gas-puff. In an attempt to control the disruptions in real time the gas-puff induced Hα intensity increase is used as a precursor for the disruption. A special comparator based trigger circuit is designed to generate a trigger for the onset of ICRH pulse when the Hα intensity crosses a pre-set threshold. Figure 7 shows the successful mitigation of the disruption with RF pulse. It is expected that these results will be useful for future steady state tokamaks.

![Figure 7: Disruption mitigation results with Hα signal for feedback.](image)

Conclusions

The wall conditioning in presence of resonance layer is more effective to reduce carbon impurity and non-resonant ICWC is more effective to reduce oxygen impurity from vessel. The wall conditioning with longer pulses of the order of second is more effective as compared to short pulses. Second harmonic heating experiments show electron temperature rise up to 500 eV and in temperature rise up to 350 eV. Gas puff disruption and mitigation with ICRH power with feedback control from Hα signal is a novel technique and shows quite encouraging results for future fusion grade reactors.

References