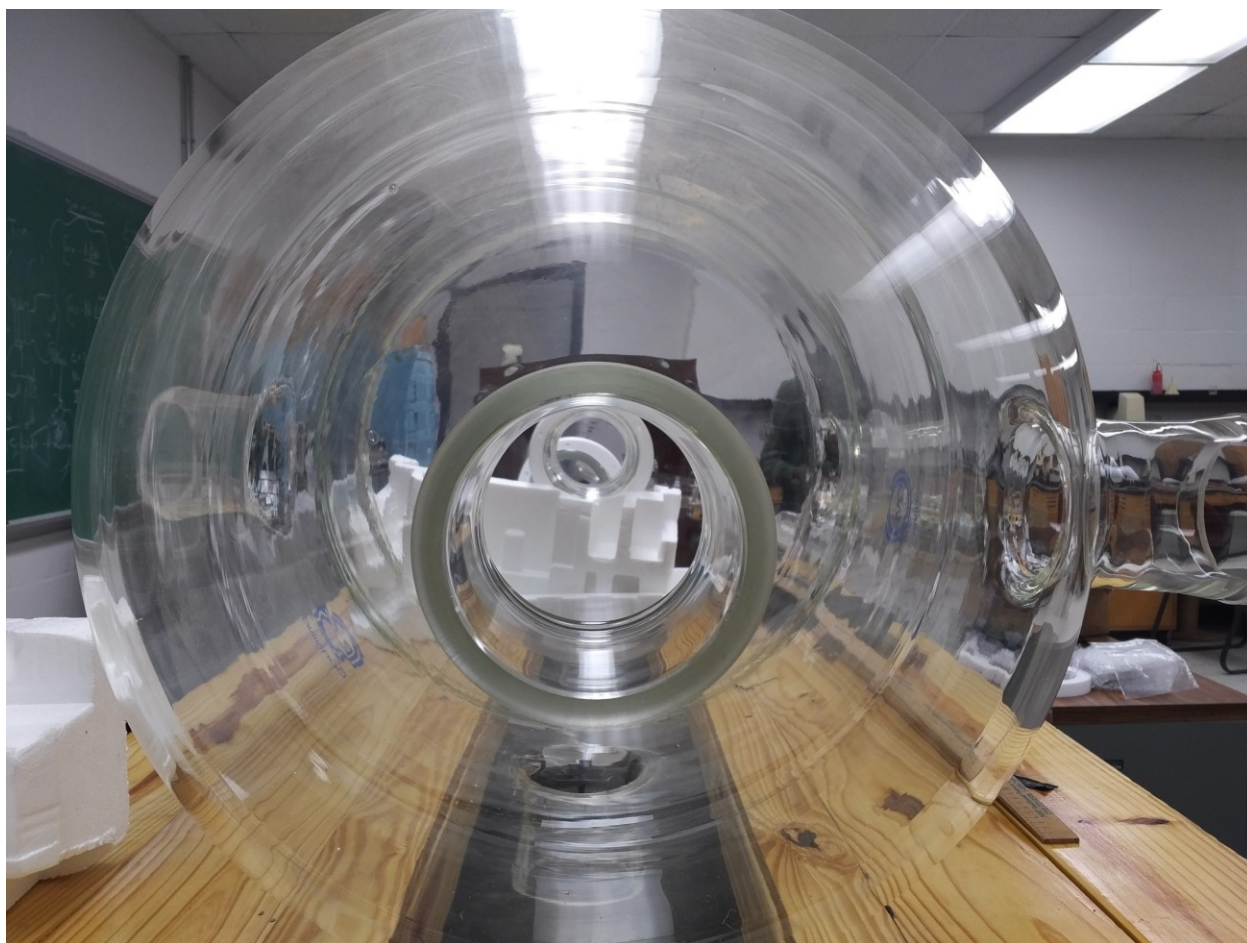


Final Technical Report

Department of Energy: DE-FG02-04ER54777

Plasma Research through DOE Support (2004-2015) - Prairie View A&M University

DR TIAN-SEN HUANG (PI)*



17 February 2017

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SUMMARY

Department of Energy: DE-FG02-04ER54777

Plasma Program through DOE Support (2004-2015) at Prairie View A&M University

DR TIAN-SEN HUANG (PI)*

During recent years (2004-2015), with DOE support, the PVAMU plasma research group accomplished new instrumentation development, conducted several new plasma experiments, and is currently poised to advance with standing-wave microwave plasma propulsion research.

On the instrumentation development, the research group completed: (i) building a new plasma chamber with metal *CF* flanges, (ii) setting up of a $6kW/2450MHz$ microwave input system as an additional plasma heating source at our rotamak plasma facility, (iii) installation of one programmatic Kepco ATE 6-100DMG fast *DC* current supply system used in rotamak plasma shape control experiment, built a new microwave, standing-wave experiment chamber and (iv) established a new plasma lab with field reversal configuration capability utilizing $1MHz/200kW$ RF (radio frequency) wave generator.

Some of the new experiments conducted in this period also include: (i) assessment of improved magnetic reconnection at field-reversed configuration (FRC) plasma, (ii) introduction of microwave heating experiments, and (iii) suppression of $n = 1$ tilt instability by one coil with a smaller current added inside the rotamak's central pipe. These experiments led to publications in *Physical Review Letters*, *Reviews of Scientific Instruments*, *Division of Plasma Physics (DPP) of American Physical Society (APS) Reports*, *Physics of Plasmas Controlled Fusion*, and *Physics of Plasmas* (between 2004 and 2015).

With these new improvements and advancements, we also initiated and accomplished design and fabrication of a plasma propulsion system. Currently, we are assembling a plasma propulsion experimental system that includes a $5kW$ helicon plasma source, a 25 cm diameter plasma heating chamber with $1MHz/200kW$ RF power rotating magnetic field, and a 60 cm diameter plasma exhaust chamber, and expect to achieve a plasma mass flow of $0.1g/s$ with $60km/s$ ejection. We anticipate several propulsion applications in near future as we advance our capabilities.

Apart from scientific staff members, several students (more than ten undergraduate students and two graduate students from several engineering and science disciplines) were supported and worked on the equipment and experiments during the award period. We also anticipate that these opportunities with current expansions may result in a graduate program in plasma science and propulsion engineering disciplines.

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1. Research Accomplishments for the Years 2012 thru 2015

A. Ion Doppler Spectroscopy (IDS) measurements for ion temperature

With the recently installed high resolution ion Doppler spectroscopy, the ion temperature and plasma flow at Prairie View rotamak plasma have been measured. The IDS measurements are observed for various impurity emission lines of which carbon lines exhibit stronger intensities. Furthermore, the diagnostics are examined in an experiment where plasma experiences sudden disruption and quick recovery. The motorized gear of the monochromator allows spectral resolution of 0.01 nm . In this case, the IDS measurements show $\sim 130\%$ increase in ion temperature. Flow measurements are shown to be consistent with plasma rotation (these results are published as part of *Houshmandyar et al.*, 2012).

B. Active control of plasma shape and poloidal boundary flux

Active control of plasma shape and poloidal boundary flux has been widely studied with the Prairie View rotamak device. With the introduction of plasma shaping coils, the elongation of plasma shape and the increase in plasma current have been systematically studied. These experimental results were very successful (these results were published as part of *Yang et al.*, 2012).

C. Study of MHD instability in rotamak-ST with an ultra-low A

Magnetohydrodynamics (MHD) instability as aspect ratio A (R_o/a) approaching 1 has been studied with the Prairie View rotamak device. Two regimes of rotamak-ST, (routine rotamak operation and spherical tokamak operation), have been systematically studied. The study includes two operation regimes: high toroidal magnetic field B_t and low toroidal field B_t (these results appeared as part of *Yang et al.*, 2012).

D. Improved magnetic reconnection experiment in FRC plasma

More recently, with experimental facility's modification the magnetic reconnection experiment in FRC plasma has been further studied at Prairie View rotamak device [*Xu et al.*, 2014]. By adding one toroidal current in rotamak's central axis area, the cutting of one FRC into two FRCs in the experiment process becomes more obvious.

Newly added toroidal coil is located inside the stainless-steel pipe; the coil has a radius of $R=1.8$ cm with 20 turns. As a comparison, the magnetic configurations without and with the inner coil are shown in Figure 1.

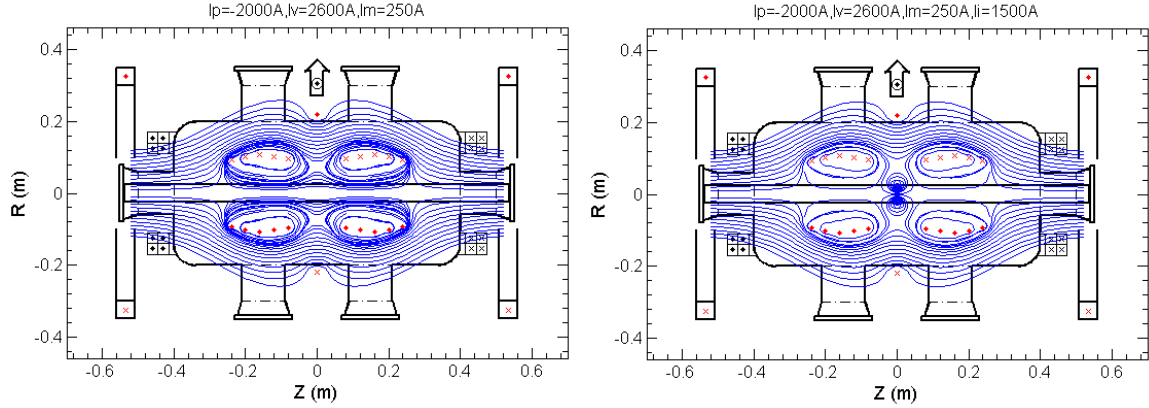


Figure 1. Comparison of the magnetic reconnections in the cases without the inner coil (left) and with the inner coil (right)

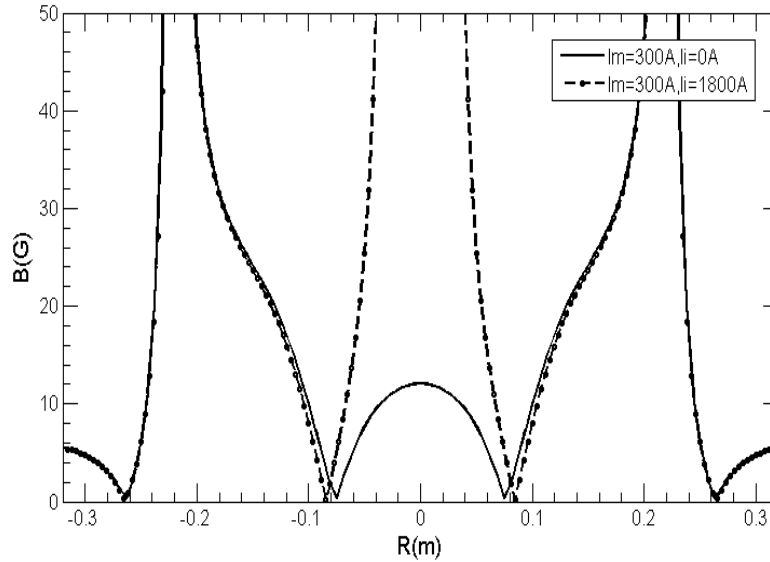


Figure 2. Variation of the magnetic field strength along the radial

The comparison of the magnetic field changes in the cases with the inner coil (solid lines) and without the inner coil (dash lines) are shown in Figure 2. The configuration with the inner coil has greatly increased the gradient of magnetic field in the center part of the chamber, and makes the field-reversed configuration (FRC) reconnection occurrence in a larger area.

E. Rotamak plasma heat with microwave

Recently, a $6kW$ $2.45GHz$ microwave system has been added on Prairie View rotamak. This is the first time that electromagnetic wave is injected into a rotamak plasma. Electron Bernstein Waves (EBW) are expected to be excited in the plasma through mode conversion from electromagnetic waves. A series of microwave experiments have been performed in both O-mode and X-mode configuration for both FRC and ST modes on Prairie View rotamak. Our experiments confirmed that the plasma in O-mode configuration doesn't response to this $2.45GHz$ microwave. But an effective pre-ionization of plasma has been observed when the filling pressure of the hydrogen gas is under $p_f=1mTorr$.

Though the microwave power is much lower in comparison with the RMF's several hundreds kW in the rotamak microwave experiments, clear oscillations of plasma current I_p and magnetic field B_R were excited when microwave was injected into plasma. From plasma current signal, oscillations with wide frequency range, $0 - 5kHz$, can be observed when microwave was injected into plasma. The magnetic oscillations were identified as $n = 1$ radial shift mode (the phase shift between two sets of Mirnov coils is π). According to theoretical study, the radial shift mode is expected to become unstable at sufficiently small elongation $E < 0.7$ ($E = Z_S / R_S$, with Z_S and R_S the separatrix half-length in the axial direction and separatrix radius, respectively). The microwave may have decreased the elongation of the plasma. Usually excitation of radial shift mode results in plasma disruption within $1ms$. But in the microwave experiments, excitation of radial shift mode seemed to favor the building of the plasma current.

2. Research Accomplishments for the Years 2009 thru 2011

A. Study of MHD instability in the RMF-driven plasmas

The PV Rotamak with cylindrical chamber is uniquely positioned for such study. The 40 ms long duration of plasma discharge allows observing global MHD instability modes. Meanwhile a set of magnetic shaping coils permits to form plasmas with $E < 1$ (oblate FRC), $E > 1$ (prolate FRC), and doublet-shape configurations with two magnetic axes. In addition, two sets of Mirnov coil array and a group of flux loop are built to detect the magnetic perturbations and plasma shape respectively.

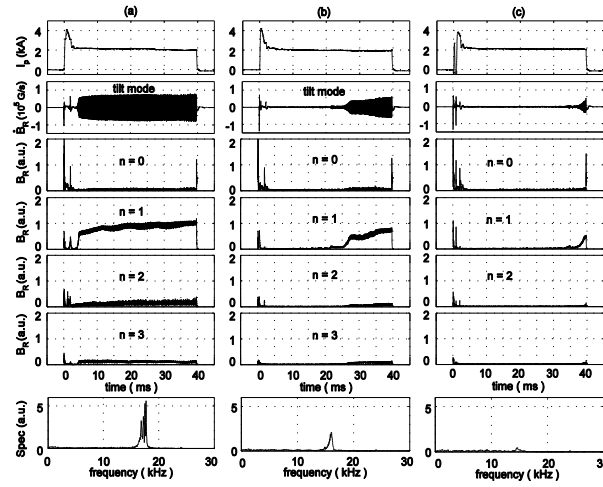


Figure 1. Time evolution of plasma current, MHD oscillation signal, magnitude of $n = 0-3$ perturbations to B_R and spectra for (a) $B_v = 25$ G; (b) $B_v = 28$ G; (c) $B_v = 30$ G.

Measurements from Mirnov magnet array provide the first evidence for $n = 1$ tilt and radial shift instabilities in rotamak-FRC plasma. The $n = 1$ tilt mode is shown in Fig. 1 This mode usually grows within a period of 1-2 ms, and then becomes saturated during the shot. Depending on vertical field and filling gas pressure, the oscillation frequency of the mode is between 15 and 19 kHz; the growth time can be up to 10 ms. This tilt mode is non-destructive, but it usually results in plasma current drop by 10%. As indicated in Fig.1, when B_v increases from 25 G to 30 G, the tilt mode starts later, and its frequency and amplitude gradually becomes smaller with higher B_v .

The tilt mode stability is strongly affected by plasma shape. We examined how the plasma shaping could affect stability of tilt mode in the range of $E \sim 0.7$ -1.5. The most striking result is seen when only the middle shaping coil (at $Z = 0$) is energized, producing doublet-shape FRCs with an internal figure-of-eight separatrix (with two O -points and an internal X -point on the midplane). It is seen in Fig. 2(a) that after the middle coil is energized, the $n = 1$ tilt mode is completely suppressed. In this shot, the middle coil is energized with $I_m = 400$ A current during $t = 20$ -40 ms. The change of elongation from $E \approx 0.92$ to $E \approx 1.37$ after the middle coil is energized cannot explain by itself the suppression of the tilt mode.

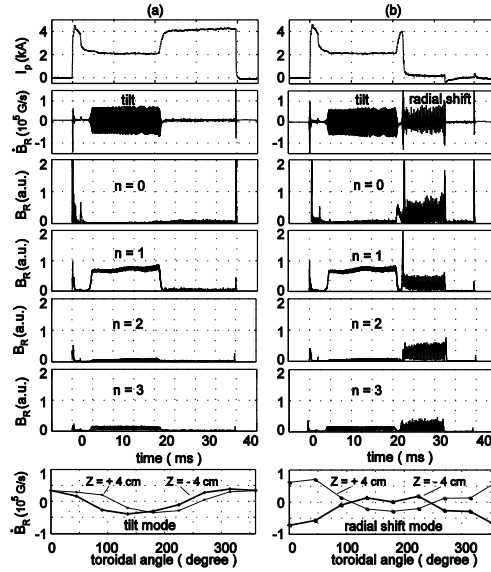


Figure 2. Time evolution of plasma current, MHD oscillation signal, magnitude of $n = 0$ -3 perturbations to B_R and signals from two sets of Mirnov coils. I_m current is fed to middle coil during 20-40 ms (a) $I_m = 400$ A; (b) $I_m = 550$ A.

Repeatable discharges demonstrate that the tilt mode can be completely suppressed by energizing the middle coil with current in the range of $I_m \sim 250$ -500 A. When I_m current is higher, 500-600 A, the tilt mode is suppressed, but $n = 1$ radial shift mode is excited and plasma is disrupted within 2-3 ms after energizing the coil, as shown in Fig. 2(b). For $I_m > 600$ A, plasma is disrupted immediately as the coil is energized.

Over a few decades of rotamak research, this is the first time to observe the tilt/radial shift mode in rotamak plasma. The detailed description for the experiments has been published in Physical Review Letters [Yang *et al*, 2009].

B. Experiments on active control of poloidal magnetic flux

Poloidal magnetic flux conservation is an important issue for RMF-driven plasmas. To address this issue, an alternative approach has been studied for the first time at PV Rotamak with a cylindrical chamber. A hybrid active/passive equilibrium control system was installed. The passive part consists of four copper rings that serve as a partial flux conserver. The active part consists of three coils wound over the chamber surface and connected to a programmable current source. The main purpose of the active coils is to keep the plasma separatrix off the chamber wall, and to control the plasma shape.

Without active coils and copper rings, the temporal change of the boundary poloidal flux is shown in Fig. 3(a). The boundary flux drops sharply by 60–80% during first 0.1–0.2 *ms* of discharge, then continues to decline during next 1–2 *ms*. For the rest of 38 *ms* the boundary flux remains at approximately zero level for $|z| < 7$ *cm*, indicating that the separatrix is resting on chamber wall.

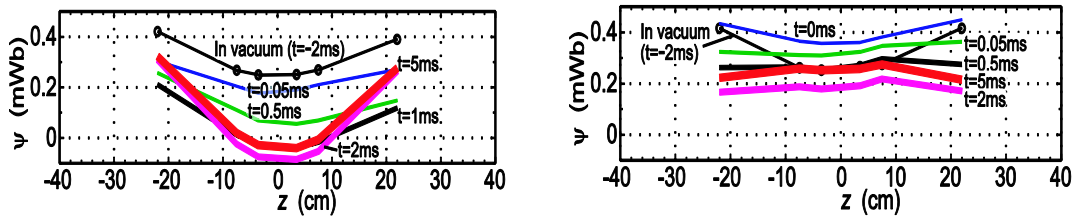


Figure 3. Change of the boundary poloidal flux in rotamak-ST discharge (a) without and (b) with active coils and copper rings. The position of flux loops is indicated by circles. The flux in vacuum is produced by the vertical magnetic field only.

Based on the measured boundary flux profile and total plasma current, the plasma shape is reconstructed as displayed in Fig. 4(a). Previously, a direct scan of magnetic field with internal magnetic probes has shown a similar, nearly spherical shape of flux surfaces.

Further experiments have been conducted with adding four copper rings. The presence of the rings helps to reduce the initial sharp drop of the flux to only 5–10%, which is then replenished by the active coils. Our experimental results indicate that the passive copper rings can help eliminating the fast variations of the boundary magnetic flux but only during plasma forming stage (within initial 2 ms).

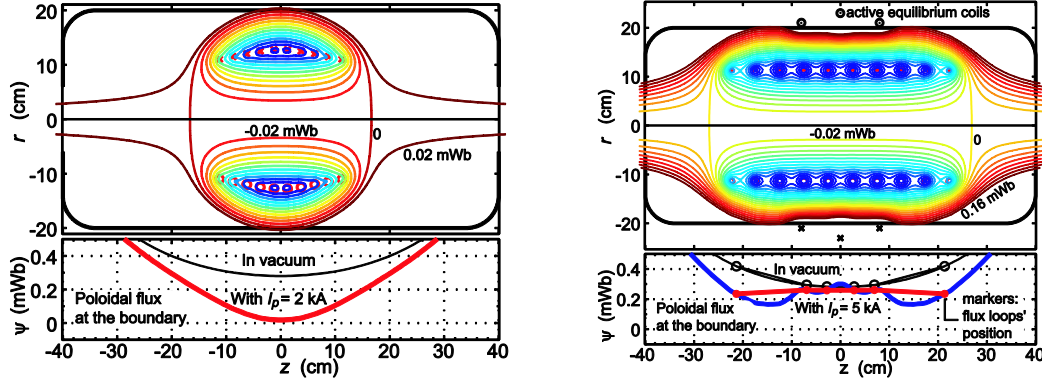


Figure 4. Change of the boundary poloidal flux in rotamak-ST discharge (a) without and (b) with active coils.

One paper giving the detailed description for the experiments and a theoretical model has been published in Physics of Plasmas [Petrov, *et al*, 2010].

C. The effect of magnetic field shape on plasma current drive efficiency

It was noted in our experiments that when the shaping coil is energized in the FRC regime, the RMF appears to have a shallower penetration (2 cm near the chamber wall), similar to the property of the higher performance mode observed in TCS when the flux conserving copper straps are wrapped around the quartz tube. However in our case the plasma current is substantially boosted while in TCS there is no change. This phenomenon seems to be contrary to the mechanism of RMF current drive, which requires a good penetration to enhance current drive. Thus, how the change of magnetic field shape affects plasma current drive efficiency in both FRC and ST regimes needs to be clarified.

Experiments have been carried out in both regimes with different combinations of shaping coils being applied at $t = 20 \text{ ms}$ until end of the discharges. To simplify the comparison, we discuss only the three cases which include the middle coil: (a) one (middle) coil; (b) three coils (one middle and two side coils at $z = \pm 8 \text{ cm}$); (c) five coils (one middle and four side coils).

Without energizing magnetic shaping coils, the rotamak-ST and FRC plasmas have a similar shape as shown in Fig. 5(a), the minor difference is the plasma elongation $E = 1.0$ in the ST regime while $E = 0.9$ in the FRC regime. As seen from Fig. 5(a), the closed magnetic surfaces are quite short along the z -axis, in spite of the fact that the RMF antenna extends to the end of the chamber.

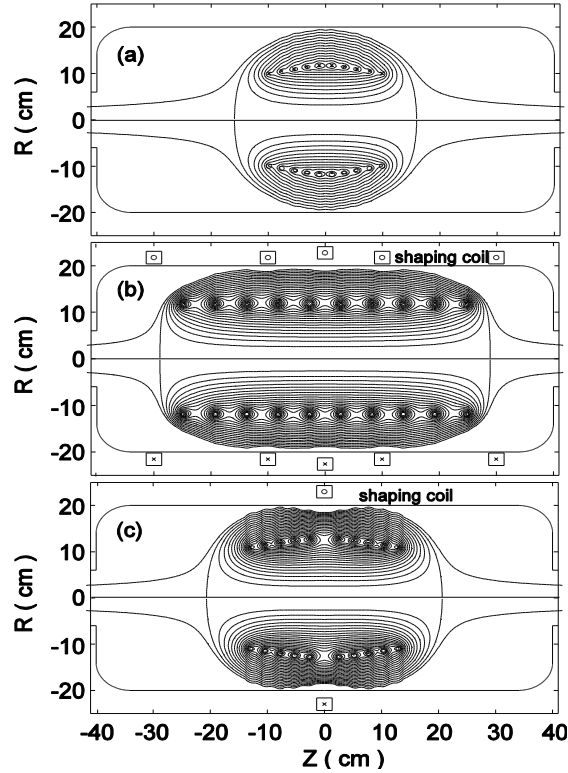


Figure 5. Reconstructed plasma shape in the discharge of (a) the ST regime, (b) the ST regime with one middle coil and four side coils, (c) the FRC regime with one middle coil.

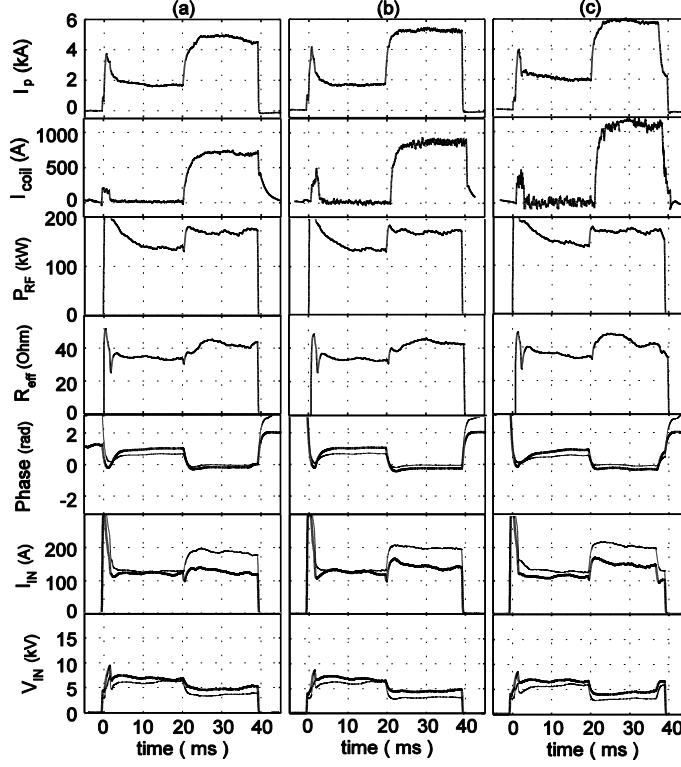


Figure 6. Time evolution of rotamak-ST plasma current, total coil current, antenna input power, the effective impedance of the plasma, antenna phase difference, input current and voltage before and after applying (a) one middle coil, (b) one middle and two side coils (at $z = \pm 8$ cm), and (c) one middle and four side coils.

For a given rf generator power of 180 kW , in the ST regime, the plasma current can be boosted by 200% from 2.0 to 6.0 kA when five shaping coils in series connection are energized with a total current of 1.2 kA , as shown in Fig. 6(c), the reconstructed plasma shape is seen in Fig. 5(b). The improvement of current drive efficiency is mainly attributed to the radial compression and the substantially axial extension of the plasma column; this in turn substantially improves impedance matching and thus increases antenna input power. In the FRC regime, plasma current can be boosted by 100% when one middle coil is employed with a total current of 0.55 kA , as shown in Fig. 7(a), the reconstructed plasma shape is seen in Fig. 5(c). It is observed that more side coils are used less coil current can be applied. Appearance of the radial shift mode limits the achievable plasma current in FRC regime.

Detailed explanation of how magnetic field shape affects plasma current and its radial profile in the regimes with and without toroidal published in *Physics of Plasmas* [Yang, *et al.*, 2011].

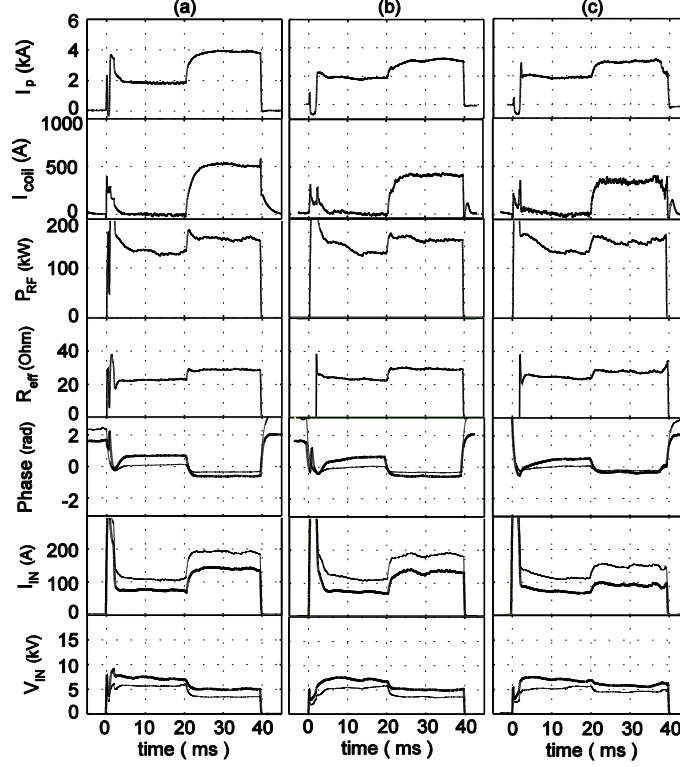


Figure 7. Time evolution of rotamak-FRC plasma current, total coil current, antenna input power, the effective impedance of the plasma, antenna phase difference, input current and voltage before and after applying (a) one middle coil, (b) one middle and two side coils (at $z = \pm 8$ cm), and (c) one middle and four side coils.

D. The effect of toroidal magnetic field on $n = 1$ mode stability in rotamak plasmas

One of the advantageous features of the rotamak-ST is its improved stability. However, over a few decades of rotamak research, there was no report about the effects of toroidal magnetic field on plasma stability. Furthermore, the MHD stability of the rotamak-ST with an ultralow aspect ratio is expected to be significantly different from those of conventional tokamaks with large aspect ratios, but investigation on this topic is lacking as well. These issues are extremely

crucial in order to achieve high toroidal beta and high toroidal utilization factor (I_p/I_z) in rotamak-ST discharges.

Experiments were typically conducted in 40 ms rotamak-FRC discharge with $I_p \approx 2$ kA, with the axial current I_z ramping linearly from 20 ms to 40 ms during a shot. There were two series of ramping experiments: one series with small I_z from 0 to 1.4 kA as shown in Fig. 8, and another series with large I_z from 0 to 6 kA as shown in Fig. 9. The strong stabilizing effect of I_z on the tilt mode is seen from Fig. 8. The amplitude of the tilt mode is gradually reduced with the increase of I_z , and the tilt mode is completely suppressed when I_z reaches 0.4 kA at $t = 24$ ms. This value is about 20% of the plasma current. The result agrees well with the finding on TS-3.

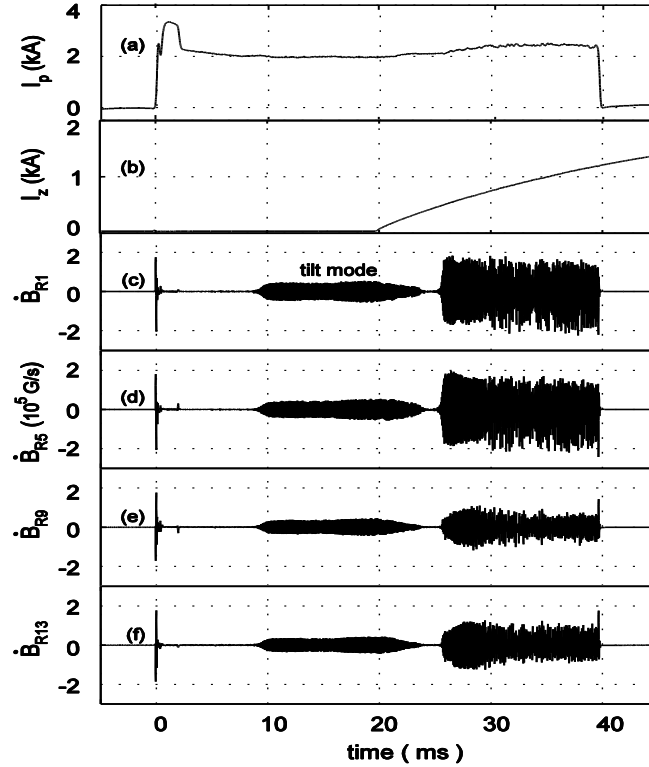


Figure 8. Time evolution of plasma current, axial current, MHD oscillation signals of a left set coil #1 and #5, and a right set coil #9 and #13. Two coils on same set are at 180° toroidal angle interval. I_z ramps almost linearly to 1.4 kA, $p_f = 1.1$ mTorr.

The most interesting result seen in Fig. 8 is that when I_z ramps above 0.5 kA ($I_z/I_p = 25\%$), a new rotating mode appears soon after the suppression of the tilt mode. This mode grows very

fast at $t = 24.5 \text{ ms}$ and saturates within 1 ms . Its amplitude remains maximum when I_z is in the range of $0.6\text{-}1.2 \text{ kA}$ ($B_t = 5\text{-}10 \text{ G}$). In contrast to the tilt mode which usually leads to plasma current drop by 10%, this new mode boosts plasma current by 25% during its saturation period. Moreover, the magnetic perturbations from the left side Mirnov array ($Z = -4 \text{ cm}$) (Fig. 8(c), (d)) increase faster than that of the right side array ($Z = +4 \text{ cm}$) (Fig. 8(e), (f)); when this mode stays saturation stage, the perturbation amplitudes of the left Mirnov array become as twice as that of the right side array. It is the first time to observe the asymmetric MHD mode in rotamak plasma.

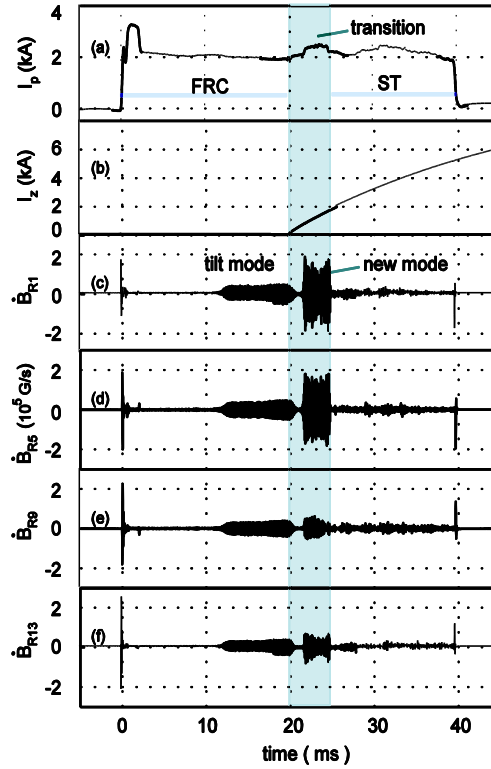


Figure 9. Time evolution of plasma current, axial current, MHD oscillation signals of a left set coil #1 and #5, and a right set coil #9 and #13. Two coils on same set are at 180° toroidal angle interval. I_z ramps to 6.0 kA , $p_f = 1.1 \text{ mTorr}$.

Another series of experiments was conducted with large I_z ramping and the tilt mode existing in the discharge at $t = 10 \text{ ms}$ before I_z ramps at $t = 20 \text{ ms}$. As can be seen in Fig. 9, in the region where the value of I_z is less than 1.5 kA , the effects of TF on the tilt mode (suppression) and the new mode (trigger) are basically same as demonstrated in Fig. 8. Since I_z ramps fast in this type of experiments, the tilt mode is suppressed quickly and the new mode saturates immediately after

being triggered in the discharge. One interesting result shown in *Fig. 9* is that when I_z approaches 1.6 kA , the new mode suddenly disappears; the corresponding ratio of I_z/I_p is around 70%. The discharge is free from any instability mode when I_z is larger than 1.6 kA . Repeated shots confirm that $I_z \approx 1.6\text{ kA}$ is the threshold value to suppress the new mode, presumably due to the increase of magnetic shear by continuously increasing TF. Since the rotamak-ST regime typically operates under the condition of $I_z > 3.0\text{ kA}$ and the ratio of $I_z/I_p > 150\%$, this result explains why no MHD mode was observed in previous rotamak-ST experiments.

Our paper with the detailed description of the experiments has been accepted for publication in the *Physics of Plasmas* [Yang, *et al.*, 2011].

Lab equipment and instrument construction

A. High speed CCD camera system

The new high speed CCD camera system, which consists of a Phantom V710 CCD camera from Vision Research Inc., a wound fiber optic image bundle (80k channels), and a few lenses, has been set up on PV rotamak, as shown in *Fig. 10*. The Phantom V710 CCD camera can get over 7,500 frames-per-second (*fps*) at full wide-screen megapixel resolution; at lower resolutions, it can get even higher frame rates up to 1,400,000 *fps*.

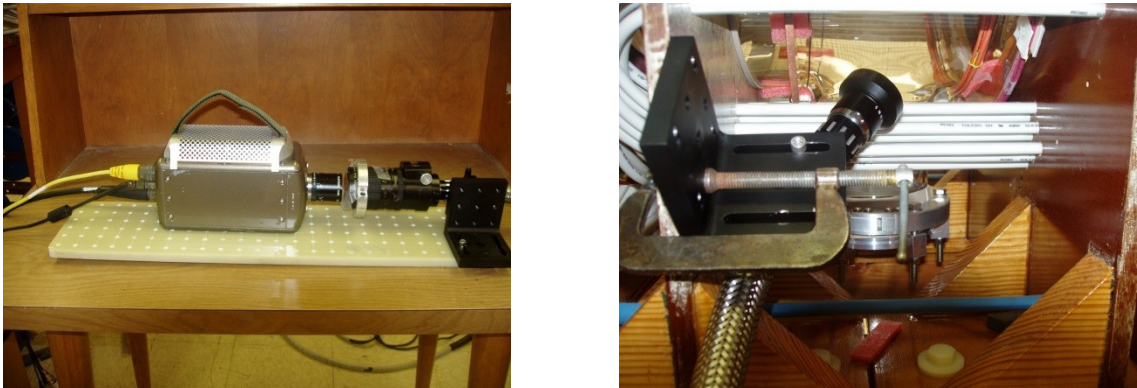


Figure 10. (a) The Phantom V710 high-speed CCD camera, (b) fiber optical bundle with C-mount wide-angle lens.

The new high speed CCD camera system not only gave us clear and direct understanding about plasma formation, MHD instability modes' triggering, development, saturation, and suppression, but also proved our experimental measurements from Mirnov coil array. This new system significantly increases our diagnostic ability in studying MHD mode instabilities in rotamak plasmas, as well as the physics process occurring in the magnetic reconnection experiment. Fig. 11 gives some typical images.

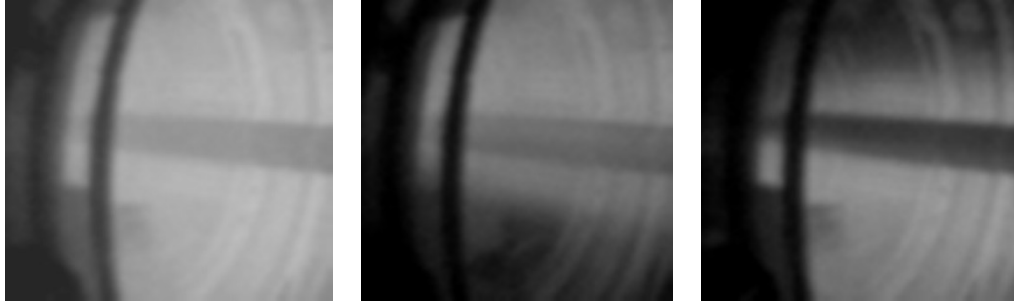


Figure 11. (a) Image of stable discharge, (b) and (c) images of disrupted discharge with the appearance of $n = 1$ radial shift mode.

B. Spectroscopy diagnostic tools

One commercial spectrometer, *i*-trometer from BWTek inc., has been set up to identify the impurities in rotamak plasmas. The *i*-trometer is designed to have a spectral range of 200-1100 *nm*, with ~ 1.5 *nm* wavelength resolution, and minimum integration time of 7 *ms*. Fig. 12 shows an example of the recorded spectrum by *i*-trometer. The major impurities in rotamak plasma are silicon, oxygen. Fig. 12 also indicates the lack of emission lines below ~ 350 *nm*, due to the Pyrex absorbance. The spectrum of plasma impurity is regarded as a reference for the diagnostics of ion temperature measurement.

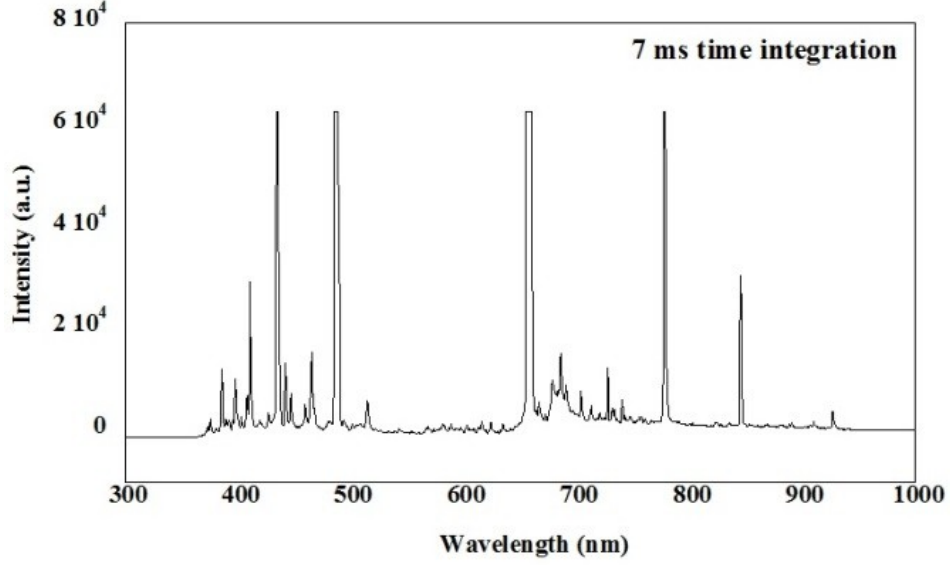


Figure 12. Plasma spectrum recorded with *i*-trometer during 40 *ms* discharge.

In order to measure ion temperature during the process of magnetic reconnection, a fast ($1\ \mu\text{s}$ time integration) ion Doppler spectroscopy (IDS) is currently under construction. The IDS system includes a Jarrell-Ash (JA) 5 monochromator, the PMT, a transimpedance amplifier, a fiber optics with 0.22 NA, and the collection optics which consist of two 3 *cm* long lens tubes, two plano-convex lenses (with 185-1100 *nm* spectral response) and an SMA coupler. The monochromator employs an 1180 Groove per millimeter ruled grating, which allows 0.01 *nm* wavelength resolution; this monochromator is calibrated by an Ocean Optics™ HG-1 mercury-argon calibration lamp. A PCI-5110 board from National Instrument™ (NI) is used to record the PMT signal.

C. A group of Mirnov coil array with 32 coils

To characterize MHD instability, a non-intrusive diagnostic tool for magnetic field perturbation, has been built. A group of Mirnov coil arrays locate respectively at $z = \pm 4\ \text{cm}$ and $z = \pm 30\ \text{cm}$ along chamber axis. At each *Z* position, eight B_R -oriented coils are mounted around chamber surface at equal 45° intervals along the toroidal angle.

Measurements from the array of Mirnov coils provided the first evidence for $n = 1$ tilt and radial shift instabilities in a 40 *ms* rotamak-FRC discharges, demonstrated the stabilization effect of magnetic shaping coils and the toroidal magnetic field.

D. A flux loop array

To study magnetic flux conservation and plasma shape control, a flux loop array which consists of six flux loops wound over the chamber surface at $z = \pm 3.5$, ± 7.5 , and ± 22 cm, has been built to measure the boundary poloidal flux.

Based on the measured boundary flux profile and total plasma current, the plasma shape can be reconstructed, as displayed in Fig. 5.

E. A new data acquisition system with 150 channels

With the rapid development of diagnostic tools on rotamak, including the measurements of 8-channel real time RF power, 6-channel flux loop, 32-channel Mirnov magnet array, and 90-channel magnetic probe array which was originally from RPPL of University of Washington, a new data acquisition system is an urgent need to our laboratory. National Instruments' data acquisition system with PXIe-6124, 16-bit, 4MS/s Simultaneous Sampling Multifunction DAQ has been recently set up.

F. A new cylindrical chamber with conflate flanges

Large amount impurities present in pyrex glass chamber with rubber *O*-rings flanges, prevent RMF driving plasmas from achieving a high temperature. In order to obtain high performance plasma discharges, a new cylindrical chamber with conflate flanges has recently been built. This modification can allow chamber being baked up to 400⁰ C, as already demonstrated by RPPL of University of Washington. The construction of other subsystems, such as supporting base and frame, vacuum, RMF antennae, matching and tuning, will be finished soon.

3. Research Accomplishments for the Years 2006 thru 2008

A. Building of a cylindrical chamber device

In order to compare the rotamak plasma discharges in the devices with differently shaped chambers and to conduct the magnetic reconnection experiment with two merging RMF-FRCs, a cylindrical chamber device was built in 2006. The plasma chamber is a 100 liter (*i.d.* = 40 cm, *L* = 80 cm) Pyrex vessel. Its background vacuum is about 5×10^{-7} Torr. One pair of the equilibrium magnetic field coils set at two ends of the chamber provide a field up to 230 G in the chamber's center. There is also a toroidal magnetic field provided by a set of six wires fed through the central axial stainless steel pipe with *o.d.* = 4 cm, which is covered with a quartz tube. The total current of six wires can be up to 20 kA. Two pairs of coils around the chamber form the *rf* antennas, powered by the same *rf* generators used in the spherical chamber device, to produce RMF; the latter drives plasma current. The new device has an additional magnetic coil in the mid-plane of the chamber wound over the surface of the chamber ("middle shaping coil"). The coil is designed to modify the equilibrium magnetic field during plasma discharge. A picture of the new device is shown in Figure 1.



Figure 1. Newly built cylindrical chamber device

B. Constructing of a poloidal magnetic flux control system

The magnetic flux control system is constructed on the cylindrical chamber. The system consists of a passive part and an active part. The passive part includes four 0.6 cm thick, 2.75 cm wide copper collars. The active part includes the “middle shaping coil” at $Z = 0$ and two “side shaping coils at $Z = \pm 8$ cm that are wound on chamber surface; the coils can be energized by a high precision fast response programmable current source (Kepco ATE6-100ML).

C. Other major instrument improvement

In the current grant period, PVAMU group has also completed a gas puffing system and a two-component pickup probe for magnetic field scanning. The gas puffing is made with a piezoelectric valve MV-112 manufactured by MaxTek company. It was installed in the spherical rotamak device. The valve is triggered by controller E1885 manufactured by PPPL, which allows to open the valve with shorter than 1 ms reproducibility. In experiments the valve was actuated for duration of 3 to 4 ms. The opening of the valve was adjusted to add the neutrals' inventory from 10% to 100%, depending on particular experiment.

The magnetic pickup probe is positioned inside a sealed L-shaped glass tube that can move in radial direction so it can reach almost any point in poloidal cross section. The signals from the pickup probe are going to be filtered and numerically integrated.

Achievements in scientific research

A. Comparison of RMF-FRC plasma properties in cylindrical and spherical devices

The information about the effects of chamber shape, rf power density, and external magnetic field on the characteristics of RMF-FRCs is of crucial importance in the direction of scaling studies for future large devices. For this reason, we conducted a series of RMF-FRC experiments at the newly built cylindrical chamber device and compared those with the experiments previously performed at our spherical chamber device.

Figure 2 shows the schematic of the spherical and cylindrical chambers. In the cylindrical chamber device, the middle shaping coil placed in the mid-plane can be briefly energized during plasma shot, so the plasma can be reshaped and pushed away from the chamber's wall.

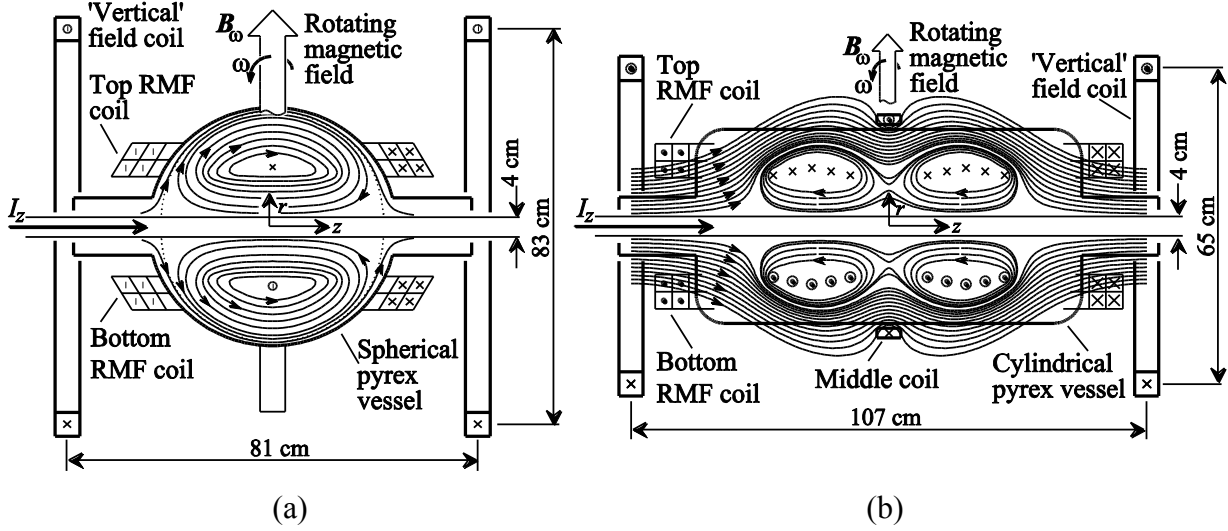


Figure 2. Schematic of the spherical chamber and the cylindrical chamber

The experiments have been conducted in both FRC and ST regimes. The results further confirm that within certain range of the equilibrium magnetic field B_v , the plasma current driven by RMF grows linearly with B_v in both FRC and ST discharges. Relatively, the ST regime is more stable and achieves higher plasma current I_p .

In comparison with the experiments conducted in the spherical chamber device, for the same *rf* power the plasma current, electron density and temperature in the cylindrical device are lower. It can be explained as the consequence of lower power density in the cylindrical chamber. Figure 3 shows the dependences of the plasma current on the vertical magnetic field in the cylindrical and spherical chamber devices for the total input *rf* power 200 kW.

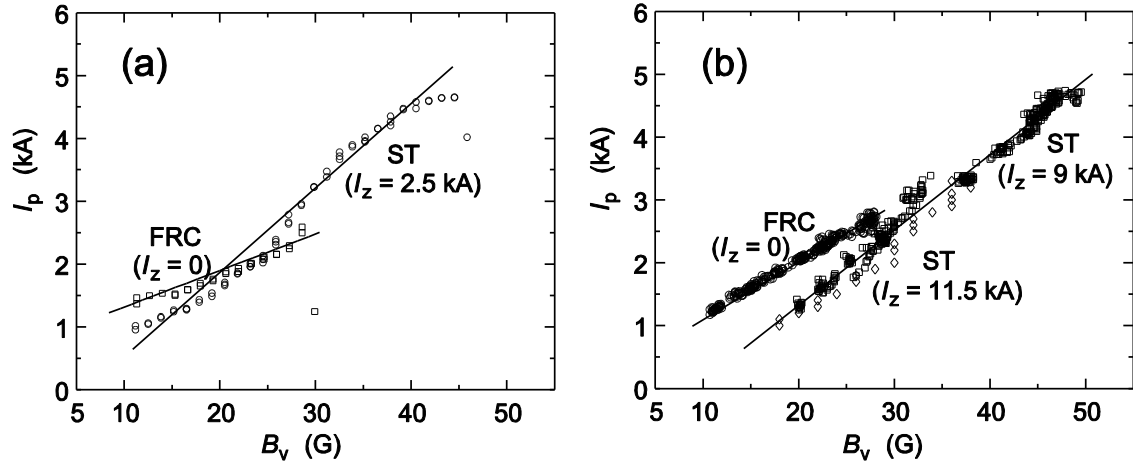


Figure 3. Dependences of plasma current on vertical magnetic field in the cylindrical chamber device (a), and the spherical chamber device (b)

The additional middle magnetic coil is made of 8 wires (2 layers by 4 turns) wound close to the chamber side surface. For both FRC and ST experiments discussed in this section, the vertical field was 25 G at chamber center, and the *rf* power was about 200 kW; in ST case, the axial current that produces toroidal field was 2.5 kA.

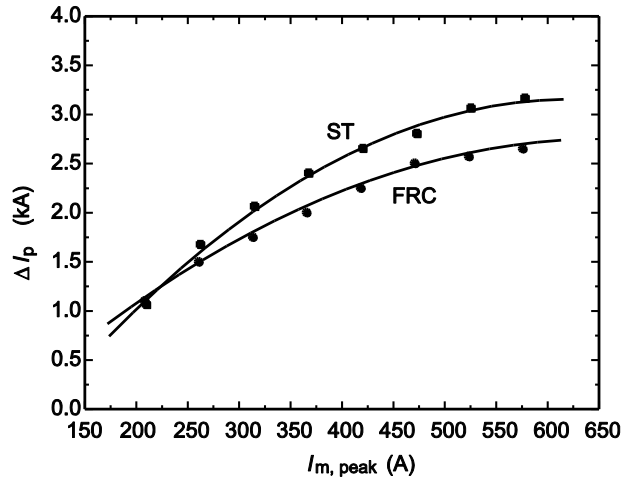


Figure 4. Increase of plasma current as a function of the applied peak current in the middle coil.

When the current in the middle coil is below the “disruptive” level, the response of plasma current to the peak value of applied current in the middle coil is summarized in Figure 4. For

both FRC and ST regimes $\Delta I_p (= I_{p,\text{peak}} - I_p (15 \text{ ms}))$ grows up monotonously with the increase of peak current in the middle coil; the increase of plasma current is usually higher in ST regime than in FRC regime. Depending on the initial conditions, plasma current can be enhanced by up to 3.1 kA from 2.1 kA to 5.2 kA; the peak plasma current is about 250% of the original plasma current. When the applied peak current in the middle coil reaches the value of 600 A, the plasma current starts to be disrupted, but in ST regime it always re-appears at a higher value after the current in the middle coil is reduced. The detailed measurements of magnetic field in plasma show that the field of the middle coil can be effectively used to reshape the plasma and to push it away from chamber walls, especially in ST regime. This result suggests that an active feedback system can be used to control equilibrium in RMF-driven plasmas.

Our paper with the detailed description of these experiments appeared in Plasma Physics and Controlled Fusion [Yang *et al.*, 2008].

B. Study of magnetic field structure evolution in RMF-FRC plasmas

Though extensive studies for the mechanism of RMF current drive have been done, there is no experimental or theoretical study on how the RMF-driven plasma would respond to a change of external equilibrium field shape yet. The main problem is: What kind and what range of changes in the equilibrium magnetic field can keep the plasma away from the chamber wall without inducing instabilities or ceasing the RMF-driven plasma current? The complication into the problem is added by the fact that as the outermost magnetic surface of the plasma is pushed or deformed, the coupling between RMF-antenna and plasma is changed, and so is the driving torque on electrons, which in turn will change the magnitude of plasma current and its distribution. The issue is related to the feasibility of an active equilibrium feedback system for poloidal magnetic flux conservation in RMF-FRC plasmas. In the presently operated short-pulse-RMF-driven FRCs, the basic equilibrium is provided by a stationary equilibrium ‘vertical’ field and flux conserving rings. For the long-pulse devices, however, an active equilibrium feedback system, similar to the systems used in modern tokamaks, is needed.

As the first step in studying this problem, we conducted a set of experiments at our newly built cylindrical chamber rotamak. In the experiments, the middle shape coil was energized for a period of 10 *ms* during 40 *ms* plasma discharges; magnetic field components inside plasma were measured directly by magnetic pickup coils. The experiments were performed for two regimes: Absence and presence of an external toroidal field produced by axial I_z current.

This study showed that the field of the magnetic coil placed at the mid-plane of plasma chamber can be used to control effectively the profile of plasma current and the structure of magnetic configuration in the plasma. The response of the plasma to a brief current in the middle coil depends strongly on the initial conditions, in particular, on the presence of the toroidal magnetic field.

If the toroidal field is absent (rotamak-FRC regime), the plasma appears to be very sensitive to the field of middle coil. If the current in the middle coil reaches a certain threshold level, the plasma current will be disrupted immediately and drop to near zero, as shown in Figure 5a. The cease of FRC regime is preceded by development of a negative current density in the region $R = 6 - 13$ *cm*, just before the disruptive level is reached, as shown in Figure 5b.

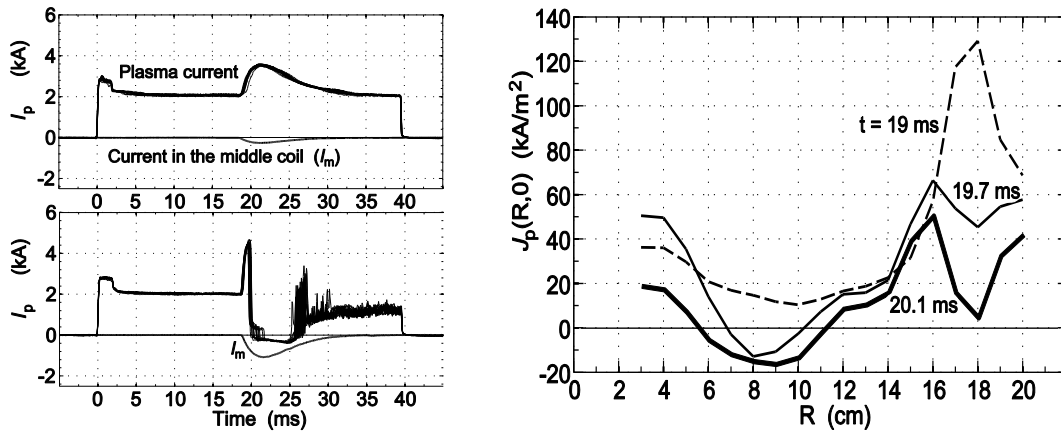


Figure 5a and 5b. Response of rotamak-FRC plasma current to small (top plot) and large (bottom plot) current in the middle coil. (5b). Profiles of the plasma current density along $Z = 0$ chord, when the current in the middle coil is just started ($t = 19$ *ms*), reaches disruptive level of 600 *A* ($t = 19.7$ *ms*) and then further grows to 850 *A* ($t = 20$ *ms*), when the total plasma current is zero.

On the other hand, if the toroidal field is present (rotamak-ST regime), the plasma responds to the middle coil field in a more controllable manner. Even when the middle coil field is slightly above the disruptive level, the plasma current does not drop to zero, but only reduces to a smaller value as shown in Figure 6a. There is no trend of generating a negative current density in the core region of the plasma, as shown in Figure 6b. Instead, the plasma current splits into two distinctive current rings ('doublet' shape). After the current in the middle coil is relaxed to a magnitude below the disruptive level, the plasma current always returns to its original magnitude and structure. The experimental results imply that an efficient control of the shape and position of RMF-driven plasma is feasible in ST regime.

The experimental results that show the changes in magnetic field structure and current density in the plasma as the middle coil is energized during the 40 ms discharge have been published [Petrov *et al.*, 2008].

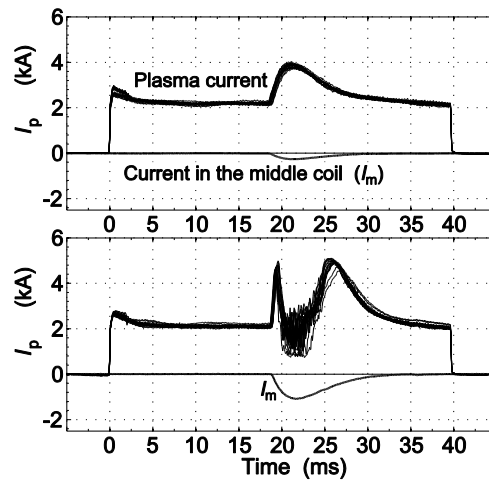


Figure 6a. Response of rotamak-ST plasma current to small (top plot) and large (bottom plot) current in the middle coil.

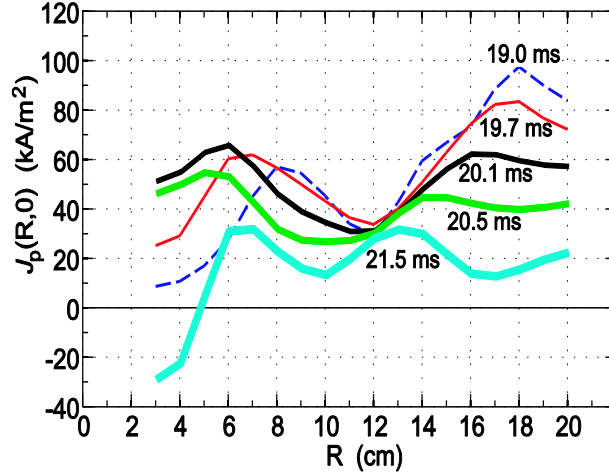


Figure 6b. Profiles of the plasma current density along $Z = 0$ chord, when the current in the middle coil is just started ($t = 19 \text{ ms}$), reaches disruptive level of 600 A ($t = 19.7 \text{ ms}$) and then further grows to 1080 A ($t = 20 \text{ ms}$), when the total plasma current is reduced.

C. Two FRCs magnetic reconnection experiment

Magnetic reconnection experiments in RMF-FRC plasma with the toroidal field, namely the ST regime, have been successfully performed in the new $80\text{cm}/40\text{cm}$ diameter cylindrical chamber device. In the experiments, RMF produced by two 0.5 MHz *rf* generators drives a plasma current about $2 - 3 \text{ kA}$ in the presence of a vertical magnetic field 25 G and the toroidal field resulted from a axial current of 2.5 kA . During a 40 ms , 200 kW power discharge, a 10 ms pulse-current 1.1 kA is applied to the middle coil to sever the plasma current into two rings and form two FRCs. When the current in the middle coil drops to zero, the two FRCs merge into one FRC again. In the experiments, variations in magnetic field profile and $H_{\square}$ signal were measured. Both the magnetic field evolution and the variation of $H_{\square}$ signal show the magnetic reconnection process.

The magnetic field in cross-section of plasma was scanned with pickup probes for the whole process from one FRC being split into two FRCs to the two being merged as one. Figure 7 shows the measured magnetic field and the plasma current density derived from the magnetic field at time 19.7 ms , 27.0 ms and 35.1 ms .

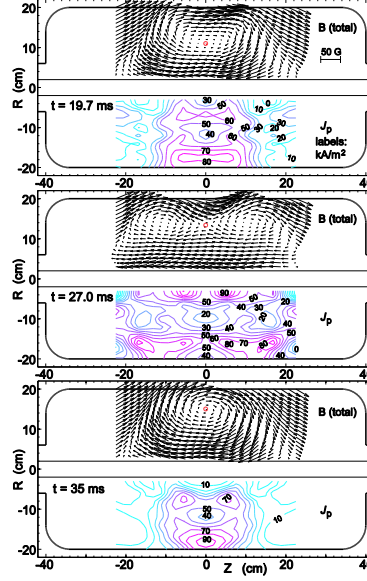


Figure 7. Evolution of magnetic field and plasma current density in the magnetic reconnection process (the current in the middle coil is added from 19 *ms* to 30 *ms*)

Figure 8 shows the cross-chamber- H_{α} signal taken for z between 4 *cm* and 34 *cm*. (The signal has a gap near $z = 0$ corresponding to the “blind” spot blocked by the middle coil.)

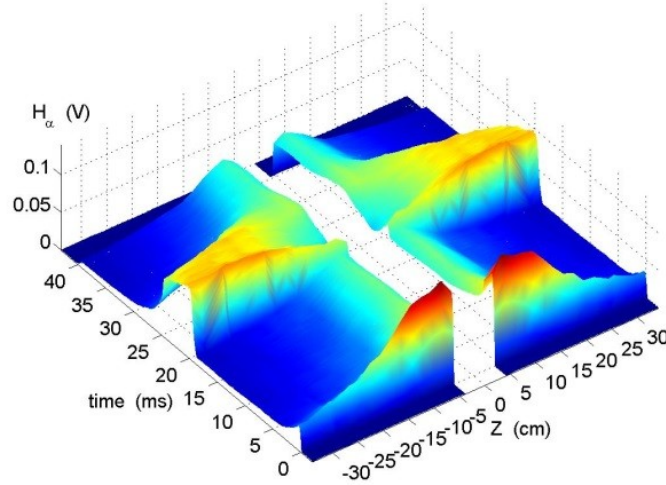


Figure 8. Signal variation in the process of magnetic reconnection, scanned at different z positions

Details on this experiment have been reported at the 2007 APS/DPP Meeting [Huang *et al.*, 2007].

D. Effects of gas puffing in RMF-FRC experiments

To study the particle-power balance in RMF-FRC plasmas, we conducted a series of experiments with hydrogen gas puffing in the spherical rotamak device [Petrov *et al.*, 2006]. Plasma discharges with puffing undergo few subsequent stages before repeatable shots are obtained; the number of stages (5-20 shots) depends on puffing parameters and chamber walls' conditions that become saturated with hydrogen.

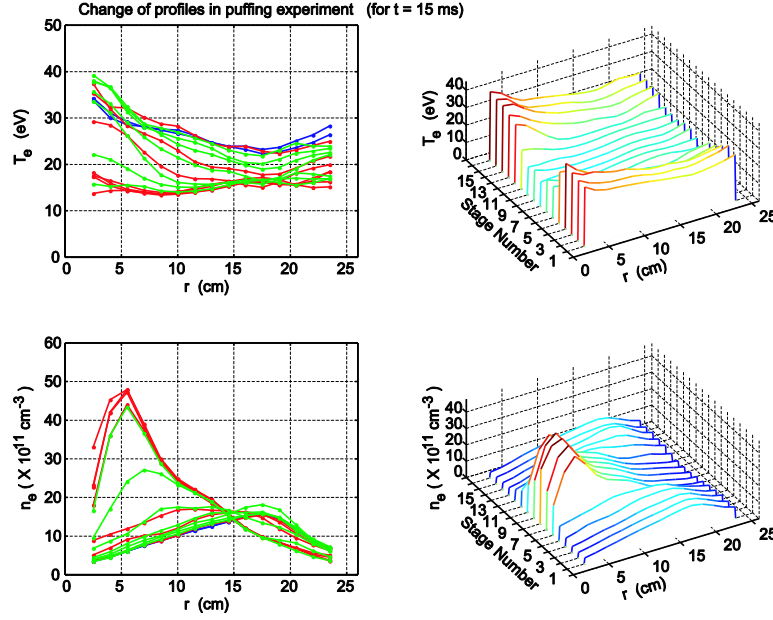


Figure 9. Profiles of plasma temperature and density, puffing is made in shots 3 ~ 9

In the discharges with puffing, the electron density increases from $(1.0-1.3) \times 10^{12} \text{ cm}^{-3}$ up to $(4-5) \times 10^{12} \text{ cm}^{-3}$, but the temperature drops from 30 to 15 eV. The profile of density in the shots with puffing becomes peaked near the center of chamber as shown in Figure 9. For the temperature, the trend is opposite. The data is consistent with global power balance but a transport model should be applied to explain the change of profiles.

An unusual result is a strong response of plasma current to the puffing. By proper choice of puffing conditions, the value of plasma current driven by RMF can be doubled, as shown in Figure 10.

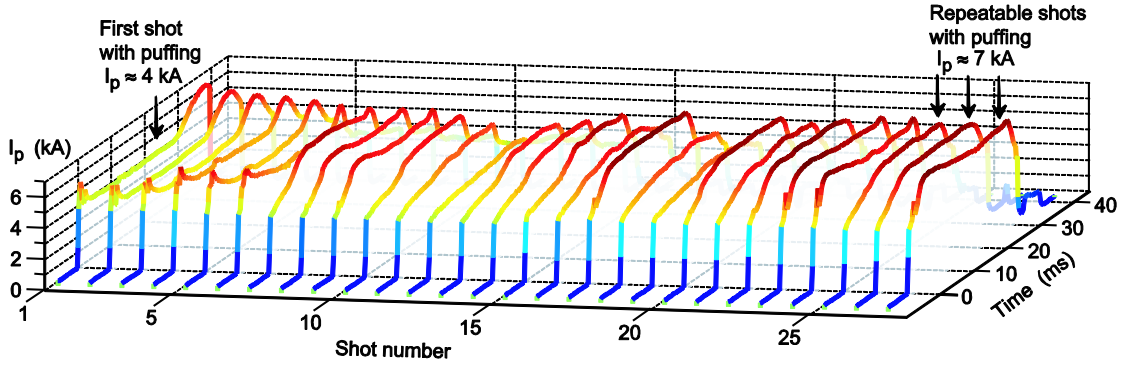


Figure 10. A sequence of shots with puffing. The puffing was started at $t = 20$ ms for a period of 3 ms in every shot

A. Experiments on poloidal magnetic flux control and study of MHD instabilities in RMF-FRC plasmas

Poloidal magnetic flux conservation is an important issue for RMF-FRC devices. As the plasma current is produced in a non-conductive chamber, the formed FRC is free to expand and contact chamber walls. When the measured value of poloidal flux at the chamber wall drops to zero, the separatrix magnetic surface contacts the glass wall.

In the short-pulse RMF-FRC devices, the problem could be solved with using flux-conserving rings made of copper or aluminum and placed outside or inside the chamber. The magnetic field, however, still diffuses through the rings in a few *ms*. Even if the rings are made of superconductive material, the requirement for a spacing gap between the rings for the RMF penetration would result in a gradual drain of the flux between the rings.

To solve this issue for long pulse RMF-FRC devices, an alternative approach has been studied at PVAMU Rotamak group with assistance from PPPL's Offsite University Support

Program. A hybrid active/passive flux control system is installed in the cylindrical Rotamak. The passive part consists of four copper rings, 27.5 mm wide/6 mm thick, placed at $Z = \pm 5.5$ cm and $Z = \pm 23.5$ cm just outside the chamber surface. The active part consists of three coils wound over the chamber surface and connected to a programmable current source Kepco ATE6-100ML with current rise time ~ 1 -2 ms. One coil is located at $Z = 0$, two other coils at $Z = \pm 8$ cm. The coils can be briefly energized during the 40 ms plasma shot. Depending on the currents in the coils, their magnetic fields can be used to change plasma shape and/or push plasma away from the walls thus preventing the loss of magnetic flux.

Without the flux control system, the poloidal flux at the chamber wall (~ 0.28 mWb for $B_v(0,0) = 25$ G) drops sharply by 60-80% during the first 0.1-0.2 ms of the shot, then continuously reduces during the following 1-2 ms, as seen in Figure 11, left column. For the rest of 38 ms in the discharge, the poloidal flux at the chamber wall remains at approximately zero level for $|Z| < 10$ cm.

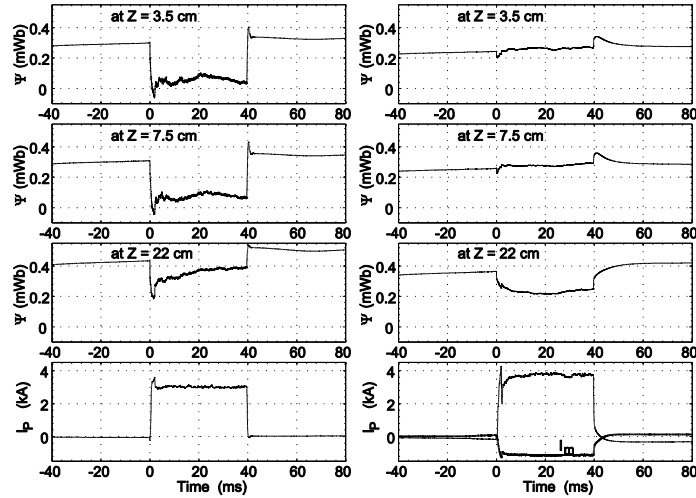


Figure 11. Change of poloidal magnetic flux at the chamber wall and plasma current in Rotamak-ST regime.

In contrast, if the central and side coils on the chamber surface are energized from the start of discharge, the flux is maintained at 0.3 mWb level during most time of the discharge, as seen in

Figure 11, right column. The left column shows the case before the copper rings and active coils are installed; the right column shows the case when four copper rings and three active coils are installed. The profile of poloidal flux along the Z axis becomes almost uniform after 2-3 ms of discharge. The copper rings, on the other side, help to reduce the initial sharp drop of the flux to only 5-10%, which is then replenished by the active coils.

The active/passive system works very well for the rotamak-ST regime, when the external toroidal field is present. However, for the rotamak-FRC regime, it is found that the presence of copper rings results in initial drop of plasma current. Also, the use of active coils is far less efficient than in the ST case. The rotamak-FRC plasma becomes unstable if the magnitude of current in the active coils exceeds certain level.

As we conducted the study of magnetic flux control in rotamak plasmas, we also studied MHD instabilities on the same device. MHD instabilities remain a major concern for RMF-FRC plasmas because of the negligible toroidal magnetic field. Traditional prolate FRCs are predicted to be unstable to $n = 1$ internal tilt mode and $n \geq 2$ modes; oblate FRCs are also unstable to $n = 1$ radial shift mode and $n \geq 2$ modes.

To explore this issue, a series of experiments were conducted in our cylindrical chamber rotamak with new diagnostics — two sets of Mirnov coils, located at $z = \pm 4$ cm. At each z position, 8 coils are mounted around chamber at equal 45° interval in toroidal direction. The elongation of plasma is controlled by three active coils described above. When the central coil is energized only, the elongation of plasma $E = Zs/Rs$ grows from 1.0 up to 1.5; when only the side coils are energized, the elongation drops below 1.0.

For rotamak-FRC regime, when $E \approx 1$ (the shaping coils are not energized), the $n = 1$ mode appears during discharge; by phase shift between signals from two sets of Mirnov coils (roughly $\pi/4$), we identified it as mostly a tilt mode. It grows in 2 - 4 ms and then becomes saturated.

The instability is fully suppressed when the central shaping coil is energized. However, if the current in the central coil is too large (resulting in $E \sim 1.5$), another mode appears which we

identify as mostly $n = 1$ radial shift mode (the phase shift between two sets of Mirnov coils is π). Usually excitation of this mode results in plasma disruption within few ms . On the other side, if the side coils are energized (resulting in $E < 1$) the tilt mode could not be suppressed.

For rotamak-ST regime, when the external toroidal field is added, the MHD instabilities are usually not observed in the same conditions as for rotamak-FRC regime. They can be sometimes observed for a low gas flow conditions. They are suppressed by central coil field, similar to rotamak-FRC case.

B. FRC plasma heating with magnetic reconnection

Magnetic reconnection is one fundamental physics process in plasma, important in astrophysics, space science and fusion plasma physics [Yamada *et al.*, 2007]. In performing this task, we will concentrate on achieving magnetic reconnection heating, as well as enhancing the understanding on magnetic reconnection physics. The experiments will be performed by using equilibrium control/shaping coils. For measuring electron temperature, we will employ newly improved Langmuir probe. For measuring ion temperature we plan to build a sensitive scanning spectrometer system.

G. Flux control and study of MHD instabilities in the process of plasma discharge

We will continue the experiments in rotamak-FRC regime using the equilibrium flux control system. The main issue is to find why this regime is so sensitive to the field of active shaping coils resulting in reversal of plasma current in the plasma core, and why the presence of copper rings leads to an initial drop in plasma current. For rotamak-ST regime, we will focus on studying the observed plasma bouncing along z direction when the active coils are energized and on general MHD stability questions.

An important issue for the equilibrium control system is to make magnetic field profile reconstruction from signals of flux loops and pick-up coils on the chamber surface, which would

provide better control of plasma shape and position. The final aim is to build a sophisticated flux control system including active and passive components that would allow sustaining the poloidal magnetic flux and reducing instabilities in long pulse.

H. Dependence of RMF-current drive efficiency on rf frequency

The RMF current drive is the result of the interaction of the screening current with rf field, which makes a non-zero net current in the toroidal direction. The rf wave frequency determines the depth of RMF penetration into the plasma and the magnitude of the net electric field.

As expected from the theory, in the rotamak-FRC plasmas the RMF penetration is better at lower frequency. However, in rotamak-ST plasmas the penetration of RMF is significantly improved for the same frequency [Zhong *et al.*, 2004], presumably due to whistler excitation. This fact indicates the possibility of working at higher frequency for rotamak-ST regime. We plan to modify our present 0.5 MHz/800 kW rf wave generator system able to work at the frequency 1 MHz, and conduct the current drive experiments with it. One rf power expert from PPPL has checked our generators, and concluded that the modification of the generators is practicable, and the cost is not very high.

I. Dependence of RMF-current drive efficiency on external magnetic field shape

The experiments for both rotamak-FRC regime and rotamak-ST regime show the magnitude of RMF-driven plasma current is proportional to the externally applied vertical magnetic field, with different slopes of the dependence in FRC and ST discharges (see e.g. Figure 3). However, only very few studies (mostly by our group) had been conducted to show how the plasma current, including its magnitude and profile, is affected by the shape of the magnetic field.

The Flinders rotamak-ST experiments show that the plasma current density profile has two peaks in the r direction and the z direction. The University of Washington RMF drive FRC experiments at its TCS also indicate the appearance of two peaks in the current density profile in radial direction. Although the mechanism for the inner current peak generation in FRC regime is

not yet understood, it can be seen [Petrov *et. al.*, 2008] that the magnitude of the inner peak strongly depends on the shape of equilibrium magnetic field. Some of the UW's experiments show an improvement in the plasma stability by shortening the antenna length.

We plan to conduct the experiments in our cylindrical chamber device to find the effects of changing the external magnetic field shape on the current drive efficiency. In the experiments, plasma density and temperature, and plasma current density distributions will be measured. For more effective measurement of the current density distribution, an improved magnetic probe array system will be applied. The results from this task will increase the understanding on the mechanism of RMF current drive in FRC plasma, and will be a reference in designing the new device.

J. Non-invasive formation of the toroidal magnetic field with Proto-sphera technique

It has been confirmed, by experiments conducted in Flinders University and PVAMU [Huang *et al.*, 2005], that both the plasma current drive and *rf* penetration in the case with toroidal magnetic field, named rotamak-ST regime, is much better than those in the case without toroidal magnetic field, named rotamak-FRC regime. However, the toroidal magnetic field requests a central column, through that the wires are set, to be added in the chamber. Adding the central column not only destroys the plasma chamber's simple hollow structure, but also eliminates the nature diverters in the device. In order to produce the toroidal magnetic field without the central column, we propose to use the Proto-sphera technique to build one non-invasive toroidal magnetic field producer in our spherical chamber rotamak device, and conduct somehow different rotamak-ST experiments in this device.

Proto-sphera is one proposed compact torus device from European fusion community. It is composed of a spherical torus and a force-free screw pinch. In the proposed device, the metal centre-post current of spherical tokamaks is replaced with the screw pinch plasma electrode current. The central current is provided by specially designed electrodes: Water cooled anode ring and directly heated cathodes ring, shown in Figure 12. The experiments conducted in a pilot

device, Proto-pinch, have demonstrated the maximum current density at the interface between the plasma and electrodes in this design can be 100 A/cm^2 .

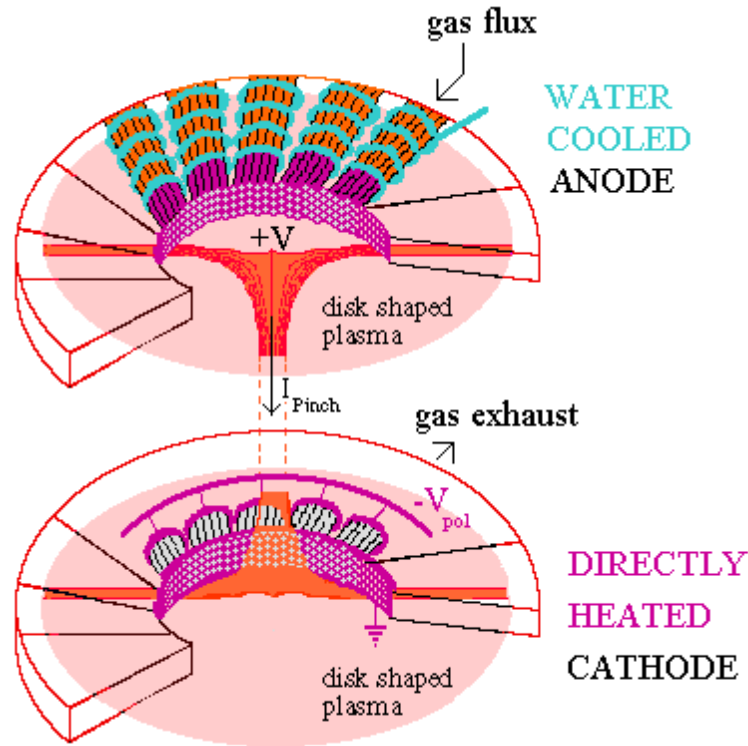


Figure 12. Scheme of annular anode and cathode system in Proto-sphere

Our plan is to remove the central stainless steel column from our spherical rotamak device, then install a small Proto-sphere electrode set on the two end flanges. The electrodes will produce 1-2 kA central current during plasma discharges. On the design and manufacture of electrodes, Proto-sphere group will provide support.

Once the non-invasive formation of the toroidal magnetic field is completed, we will conduct the rotamak-ST plasma experiments in the modified device. We are interested in the comparison of RMF-Proto-sphere plasma with the rotamak-ST plasma we have studied. The new experiments may have new plasma configuration because the plasma is expected to self-reorganize until it achieves a stable state. Once the RMF-Proto-sphere experiment gets success, we will adopt Proto-sphere technique in the new device.

4. Research Accomplishments for the Years 2002 thru 2005

A. Establishment of the PVAMU rotamak lab

In September 2002, operation of the rebuilt rotamak device began. The lab consists of three rooms: rotamak room, *rf* generator room, and control room. (There is one shielding cage in each of the rotamak room and the *rf* generator room.) In building the rotamak, a new 60 liter pyrex plasma chamber and new vertical and toroidal magnetic coils were manufactured. In addition, with the assistance of PPPL (Princeton Plasma Physics Laboratory) and Oak Ridge National Lab (ORNL) improvements on the equilibrium magnetic field and *rf* power output were made. The maximum vertical and toroidal magnetic fields increased by 50% from the levels used previously in Australia. The *rf* generator output power for rotating magnetic field increased from 400 kW to 600 kW. Figure 1 sketches the structure of the rotamak chamber and a FRC driven by RMF inside it.

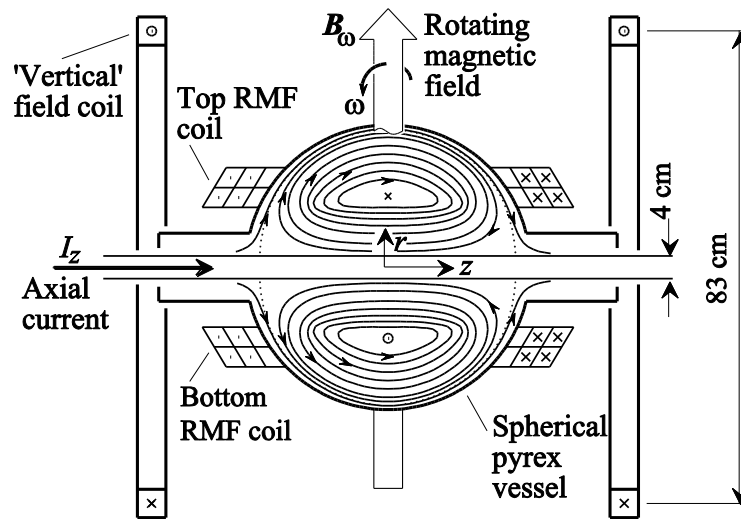


Figure 1. The schematic of rotamak.

The device has been continually improved. In particular, we have either rebuilt or newly built various diagnostic instruments and data acquisition system. At the present time, the device has the diagnostic tools: (a) Hall probes; (b) Magnetic probes; (c) Rogowski coils; (d) Langmuir probes; (e) Mach probes; (f) Electric field probes; (g) *rf* current sensors; and (h) *rf* voltage probes.

Also, the data acquisition systems: (a) A PC data acquisition board with two fast channels for maximum sample rate of 100 MHz; (b) Two digital oscilloscopes, one model Agilent 54622D and one model Tektronix TDS2014, and (c) A CAMAC system from PPPL having 15 channels with maximum sample rate 500 *kHz*, and 32 channels with maximum sample rate 40 *kHz* .

B. Performance of PVAMU rotamak

After the rotamak was rebuilt, the PVAMU group first worked on the experiments of measuring general properties of rotamak plasmas in different conditions. The experiments have been conducted in the cases of zero toroidal field and non-zero toroidal field. The former corresponds to field reversed configuration (FRC) and the latter corresponds to spherical tokamak (ST). Steady hydrogen-plasmas with 3 *kA* current in FRC and 10 *kA* current in ST, both maintained for 40 *ms*, have been achieved. A list of the device parameters and plasma parameters is given in Table 1.

The PVAMU has performed rotamak experiments on three subjects: (i) high harmonic *rf* components in rotamak plasmas, (ii) self-generation of toroidal magnetic field in rotamak-FRC plasmas, and (iii) comparison of rotamak-FRC plasmas with rotamak-ST plasmas. These experiments have increased the understanding of the mechanism of RMF current drive and the confinement of rotamak plasmas. A brief summary of the experiments follows:

			With TF	Without TF
GENERAL PARAMETERS				
Radius of chamber	a	cm	25	25
RF generator frequency	f	MHz	0.5	0.5
RMF magnitude in plasma (max)	B_o	Gauss	16	24
‘Vertical’ field ($z=0$, $r=0$, vacuum)	B_{v0}	Gauss	50	25

Toroidal field ($r=12.5\text{ cm}$)	B_ϕ	Gauss	140	0
Total field ($z=0, r=a, \text{plasma}$) $B=(B_\omega^2+B_v^2+B_\phi^2)^{1/2}$		Gauss	97	42
PLASMA PARAMETERS				
Electron density	n_e	cm^{-3}	2.0×10^{12}	2.0×10^{12}
Electron temperature	T_e	eV	30	20
Ion temperature (estimate)	T_i	eV	5	5
Electron gyro radius in total field B	ρ_e	cm	0.2	0.3
Ion gyro radius in total field B	ρ_i	cm	3.4	7.7
β for the total field B	$8\pi n_e k T_e / B^2$		0.22	0.77

Table 1

High-harmonic rf components in rotamak plasmas

In most theoretical and experimental studies, attention has been focused on penetration of the fundamental harmonic of the rotating field into the plasma. We extended the study to high harmonics. We measured and analyzed for the first time high harmonic oscillations of the RMF in discharges with and without a toroidal field. In the case without a toroidal field, B_r and B_ϕ have large odd order harmonics, as expected by theory. The presence of toroidal field greatly enhances the even harmonics. The ratio of the second harmonic to the fundamental can be as high as 45% in this case.

In the experiments, frequency spectra were determined at different radii. The observed signals and radial profiles of the harmonics of B_r and B_ϕ components are shown in Figures 2 and 3. Figure 2 gives the probe signals and frequency spectra of rotating magnetic field in plasma. Figure 3 compares the penetrations of the harmonics into the plasma. The applied toroidal field dramatically improves the penetration of the RMF into the plasma.

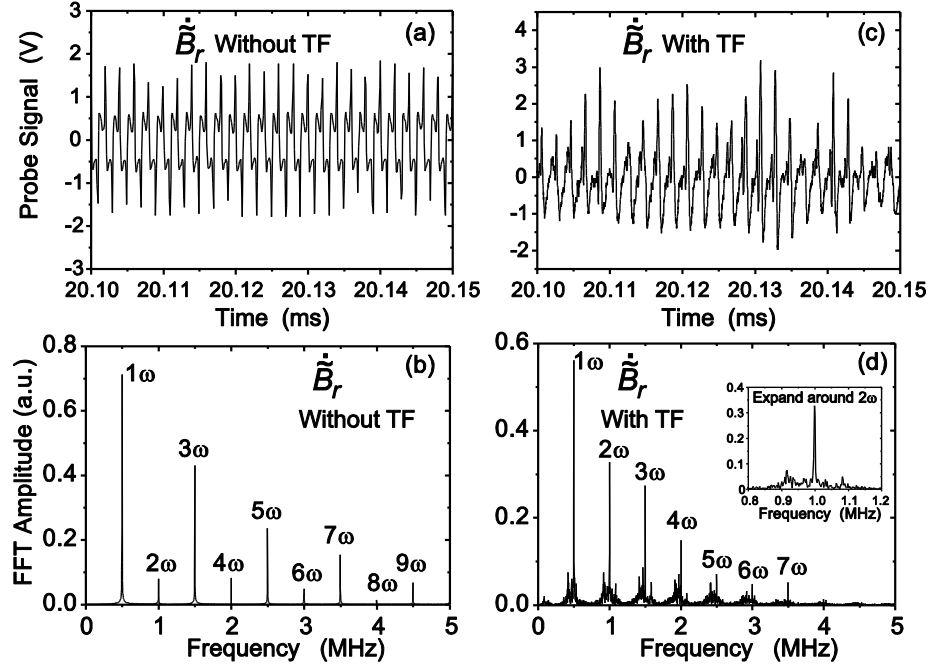


Figure 2. Probe signals and frequency spectra of rotating field in plasma.

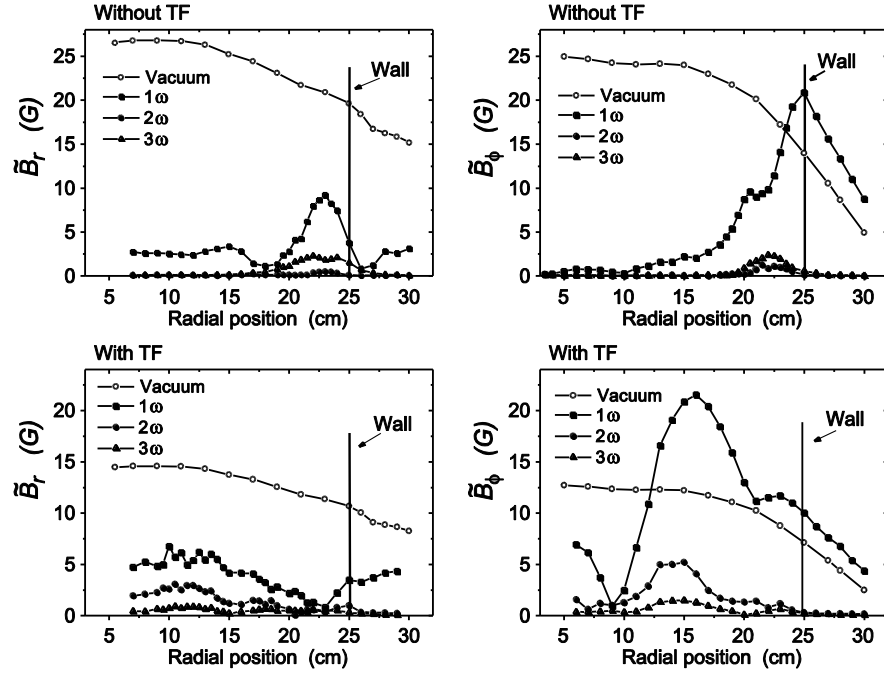


Figure 3. Radial profiles of harmonics of RMF radial and toroidal components.

Figure 4 (a) and (b) show the radial profiles of the self-generated toroidal field in the first and second rounds of the experiments, respectively. Figure 5 shows the loops of the model toroidal magnetic field for the two rounds of experiments designated in Figure 4.

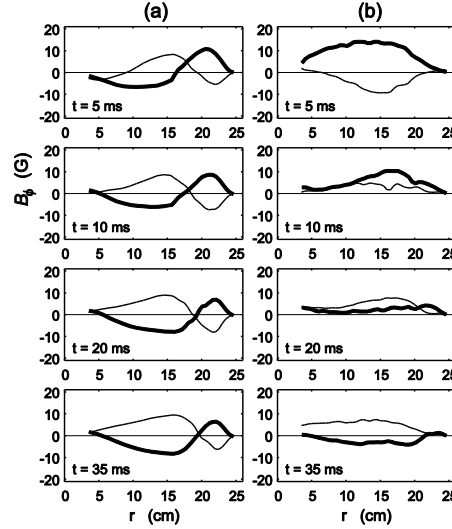


Figure 4. Radial profiles of the self-generated toroidal magnetic field, (a) in the first round of experiments, and (b) in the second round. The thin lines correspond to the scans at $z = -5$ cm, and the bold lines to the scans at $z = +5$ cm.

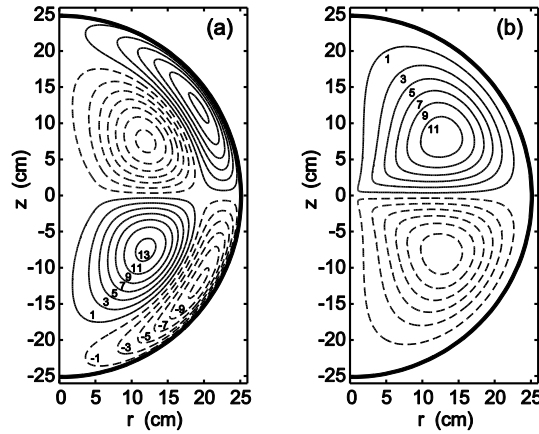


Figure 5. The loops of the model toroidal magnetic field, (a) in the first round of experiments, and (b) in the second round of experiments. Solid lines correspond to the positive B_ϕ (opposite to the plasma current), and dashed – to the negative B_ϕ . The labels designate the magnitude of B_ϕ in Gauss.

A paper “Observation of the self-generated toroidal magnetic field in rotamak” by Huang, Petrov and Zhong for the aforementioned work has been submitted to the journal *Physics of Plasmas*.

Comparison of rotamak-FRC and rotamak-ST plasmas

The RMF drive current, and the density and temperature of the plasma had been investigated for both FRC and ST operations. Most of the experiments previously completed at Flinders University were reproduced at the rebuilt device. We obtained the radial profiles of vertical magnetic field, current density, and plasma density and temperature for both the FRC and ST regimes. Results are similar to those obtained by the Flinders group. New results from our experiments include the linear relationships of the optimized plasma current with the applied vertical magnetic field for the two regimes, the relationship of the plasma current with the added toroidal magnetic field, and correlation of the improvement in RMF penetration with excitation of Alfvén waves. Figure 6 shows the linear relationships of the optimized plasma current with the applied vertical magnetic field for the two regimes, resulted from our experiments. Figure 7 shows the dependence of plasma current on the toroidal field that is indicated by the axial current under the condition of fixed vertical magnetic field.

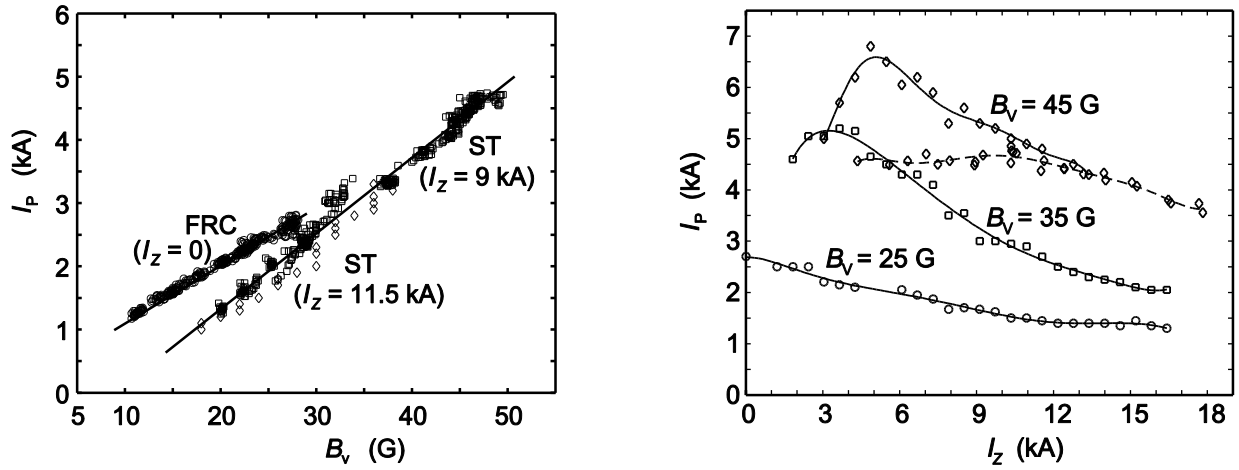


Figure 6. The dependence of plasma current on the ‘vertical’ magnetic field for the FRC (circles) and ST regimes. Figure 7. The dependence of plasma current on the axial current for a fixed magnitude of the ‘vertical’ field.

A paper titled “Comparison of rotamak plasmas in FRC and ST configurations” by Petrov, Zhong, and Huang, that summarizes the aforementioned work, has been submitted to the journal *Plasma Physics and Controlled Fusion*.

5. Relevant Publications (2004-2015) with Support from DE-FG02-04ER54777

A. Microwave experiments on Prairie View Rotamak

R. Zhou, M. Xu, and T.S. Huang
Physics of Plasmas 22, 05450, 2015

B. Improved Magnetic Reconnection Experiment at FRC Device

M. Xu, R. Zhou, D. Vasquez, and T.S. Huang
Abstract Submitted for the DPP14 Meeting of the American Physical Society DPP14-2014-001073, 2014

C. High sensitivity, low noise Mirnov coil array on Prairie View Rotamak

X. Yang, S. Houshmandyar, O. Dada, E. Reddic, and T. S. Huang
Review of Scientific Instruments, 83, 10E304, 2012

D. High resolution ion Doppler spectroscopy at Prairie View Rotamak

S. Houshmandyar, X. Yang, and R. Magee
Review of Scientific Instruments 83, 10D506, 2012

E. Euler potentials' discontinuity in the presence of field line-aligned currents

T.S. Huang and D. Wu
SPECIAL ISSUE: Basic Plasma Processes in Solar-Terrestrial Activities April 2012
Vol.57 No.12: 1436 – 1437, 2012

F. Boost of plasma current with active magnetic field shaping coils in Rotamak discharges

X. Yang, J. Goss, D. Kalaria, and T.S. Huang
Physics of Plasmas, 18, 080703, 2011

G. Effect of toroidal magnetic field on $n = 1$ mode stability in Rotamak plasmas

X. Yang, J. Goss, D. Kalaria, and T.S. Huang
Physics of Plasmas, 18, 080708, 2011

H. Experiments on Rotamak plasma equilibrium and shape control

T.S. Huang, X. Yang, Y. Wang, Y. Petrov
Phys of Plasmas, 17, 012506, 2010

I. Suppression of $n = 1$ tilt instability by magnetic shaping coils in Rotamak plasmas

T.S. Huang X. Yang, Y. Petrov
Physical Review Letters, 2009, Lett 102, 255004

- J. Magnetic Field structure evolution in rotating magnetic field plasmas**
T.S. Huang X. Yang, Y. Petrov
Physics of Plasmas 15, 072509, 2008
- K. Comparison of Rotamak plasma discharges in spherical and cylindrical devices**
T.S. Huang X. Yang, Y. Petrov
Plasmas Physics and Controlled Fusion, 50, 085020, 2008
- L. The effect of toroidal field on the rotating magnetic field current drive in Rotamak plasmas**
Z Fang-Chuan, T.S. Huang, Y. Petrov
Chinese Physics, Vol 16, 2007
- M. Magnetic reconnection of two FRCs driven by rotating magnetic field**
Y. Petrov, X. Yang, T.S. Huang
Abstract Submitted for the DPP06 Meeting of the American Physical Society, 2006
- N. Comparison of Rotamak plasmas in FRC and ST configurations**
T.S. Huang, Y. Petrov, F.C. Zhong
Plasma Physics and Controlled Fusion, 47, 1517, 2005
- O. High harmonic fields in a Rotamak plasma**
T.S. Huang, Y. Petrov, F. Zhong
Physics of Plasmas 11, L1, 2004

NOTE: For all re-prints of this listing, please see Appendix-C (included in this document)

6. Education and Training

During the three year funding period, PVAMU Plasma Research Program has maintained one or two graduate students and two or three undergraduate students in each semester, from engineering and science majors, to participate in the project. We trained them with basic knowledge about plasma physics, how to work with diagnostic tools, and how to use LabView and Matlab software tools to acquire and process experimental data. The students also worked on designing and building various electric circuits and diagnostic tools (Langmuir probes, Mirnov coils, Rogowski coils, and magnetic probe array) and operate plasma device and diagnostic tools for proposed experiments. Through their participation in the project, the students not only learned the varying practical skills that they do not receive from their classes and teaching labs, but also acquired the knowledge of fusion plasma physics. The following is the listing of the students who participated in the rotamak project during the past three years.

Undergraduate Students:

- Ricky Shaw
- Milan Barnett
- Jermain Goss
- Ezra Sidney
- Ashfaqur Razzaq
- Erick Reddic
- Oluwatomisin Macaulay
- Daniel Isokpunwu
- Samiria Percival
- Amadou Tounkara
- Brandon Woodson
- Elegbede Adetunji.

Graduate Students:

- Samuel Aboagye
- Dhara Klaria

7. Instrument Development

A. High Speed CCD Camera System

The new high speed charge-coupled device (CCD) camera system, which consists of a Phantom V710 CCD camera from Vision Research Inc., a wound fiber optic image bundle (80k channels), and several lenses, has been set up for the Prairie View rotamak. The Phantom V710 CCD camera can get over 7,500 frames-per-second (*fps*) at full wide-screen megapixel resolution; at lower resolutions, it is expected to get even higher frame rates up to 1,400,000 *fps*.

B. Spectroscopy Diagnostic Tools

One commercial spectrometer, *i*-trometer from BWTek inc., has been set up to identify the impurities in rotamak plasmas. The *i*-trometer is designed to have a spectral range of 200-1100 *nm*, with ~ 1.5 *nm* wavelength resolution, and minimum integration time of 7 *ms*. The major impurities in rotamak plasma are silicon, oxygen and we noted that the lack of emission lines is below ~ 350 *nm*, due to the Pyrex absorbance. The spectrum of plasma impurity is regarded as a reference for the diagnostics of ion temperature measurement.

In order to measure ion temperature during the process of magnetic reconnection, a fast (1 μ s time integration) ion Doppler spectroscopy (IDS) was added. The IDS system also includes a Jarrell-Ash (JA) 5 monochromator, the PMT, a transimpedance amplifier, a fiber optics with 0.22 NA, and the collection optics which consist of two 3 *cm* long lens tubes, two plano-convex lenses (with 185-1100 *nm* spectral response) and an SMA coupler. The monochromator employs an 1180 Groove per millimeter ruled grating, which allows 0.01 *nm* wavelength resolution; this monochromator is calibrated by an Ocean Optics™ HG-1 mercury-argon calibration lamp. A PCI-5110 board from National Instrument™ (NI) is used to record the PMT signal.

C. A group of Mirnov coil array with 32 coils

To characterize magnetohydrodynamic (MHD) instability, a non-intrusive diagnostic tool for magnetic field perturbation, has been built. A group of Mirnov coil arrays are located at $z = \pm 4$ *cm* and $z = \pm 30$ *cm* along chamber axis. At each *Z* position, eight B_R -oriented coils are mounted around chamber surface at equal 45° intervals along the toroidal angle.

Measurements from the array of Mirnov coils provided the first evidence for $n = 1$ tilt and radial shift instabilities in a 40 ms rotamak-FRC discharges, demonstrated the stabilization effect of magnetic shaping coils and the toroidal magnetic field.

D. A flux loop array

To study magnetic flux conservation and plasma shape control, a flux loop array which consists of six flux loops wound over the chamber surface at $z = \pm 3.5, \pm 7.5$, and ± 22 cm, has been built to measure the boundary poloidal flux. Based on the measured boundary flux profile and total plasma current, the plasma shape are reconstructed.

E. A new data acquisition system with 150 channels

With the rapid development of diagnostic for the rotamak, including the measurements of 8-channel real time RF power, 6-channel flux loop, 32-channel Mirnov magnet array, and 90-channel magnetic probe array which was originally from RPPL of University of Washington, a new data acquisition system is an urgent need to our laboratory. National Instruments' data acquisition system with PXIe-6124, 16-bit, 4MS/s Simultaneous Sampling Multifunction DAQ has been recently set up.

F. A new cylindrical chamber with conflate flanges

Large amount impurities present in the pyrex glass chamber with rubber O-rings flanges, prevent RMF driving plasmas from achieving a high temperature. In order to obtain high performance plasma discharges, a new cylindrical chamber with conflate flanges has been recently built. This modification can allow chamber being baked up to 400⁰ C, as already demonstrated by Redmond Plasma Physics Laboratory (RPPL) of University of Washington.

8. Other Equipment and Resources

A. Lab Rooms

Rotamak Device Room: $30' \times 31' = 930 \text{ sf.}$; with shielding cage

Control Room: $38' \times 27' = 1026 \text{ sf}$

RF Generator Room: $30' \times 25' = 750 \text{ sf.}$; with shielding cage, chilled water circulation tank

Instrumentation Room: $30' \times 36' = 1080 \text{ sf}$

B. Rotamak Devices

Spherical chamber rotamak: 60 l ($\text{o.d.} = 50 \text{ cm}$) and 10^{-6} Torr vacuum system;

Cylindrical chamber rotamak: 100 l ($L = 80 \text{ cm}$, $\text{o.d.} = 40 \text{ cm}$) with $5 \times 10^{-7} \text{ Torr}$ vacuum;

RMF coils with matching and tuning capacitors.

C. Control and data acquisition network

Programmable logic controller (PLC)

Programmable time-delay/pulse generator 8-channel for trigger signals

Interface units

Integrators and filters for measurement

PC data acquisition board NI-PCI-5112 with two fast channels 100 MS/s

PC data acquisition board NI-PCI-6254 with 32 channels 1.25 MS/s

A CAMAC system having 15 channels at 500 kHz , and 32 channels at 40 kHz

D. Diagnostics tools

Langmuir probes

H_{α} detector

Hall probes

Magnetic probe array

RF voltage and current probes

Rogowski coils

Flux loops (6 loops at $Z = \pm 3 \text{ cm}$, $\pm 7 \text{ cm}$, and $\pm 22 \text{ cm}$)

Mirnov coil array (four sets by 8 coils around chamber)

Mach Probes

Electric field probes

High resolution and fast response Ion Doppler Spectroscopy (IDS)

Phantom 710 CCD Camera from Vision Research Inc

E. Power sources

Two 400 kW, 500 kHz RF generator

20 kV/196 kJ capacitor bank and its charging system

DC power for vertical field: 20 kW/250 A EMHP-80-250 power supply

Capacitor bank and charging system for toroidal field: 146 kJ/1.44 F

Programmable current source Kepco ATE6-100ML (100 A/6 V)

Programmable power supply/amplifier Kepco BOP100-1M (1 A/100 V)

F. Chilled Water Supply: Used for cooling RF generator's circulation water

G. Compressed air supply: Used for capacitor bank operation

9. Current Facilities and Future Plans

At Prairie View A&M University, the plasma laboratory system is set up in four major laboratory rooms: (1) rotamak plasma device system room (930 *sqft*), (2) operation-control room (1026 *sqft*), (3) radio frequency power generator room with 2x400*kW* RF power generator room (800 *sqft*), and (4) capacitor bank units' cage (400 *sqft*). The main equipment and instruments include: two plasma experiment devices with high-vacuum pump systems, electric-field and magnetic-field controlling devices, electrical power supply systems with different frequencies and different voltages, various measurement devices and diagnostic tools, and a control and data acquisition station.



Rotamak Laboratory



Control Room



800*kW* RF Power Generators

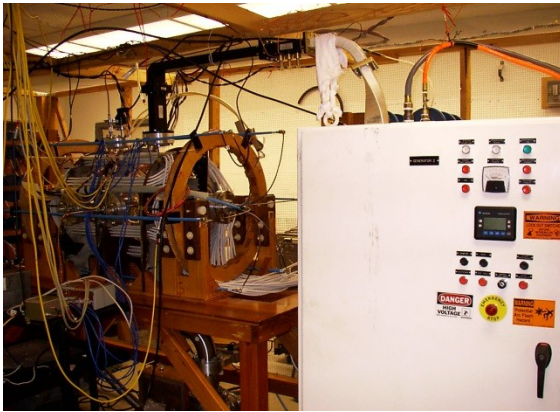


Capacitor Bank Units

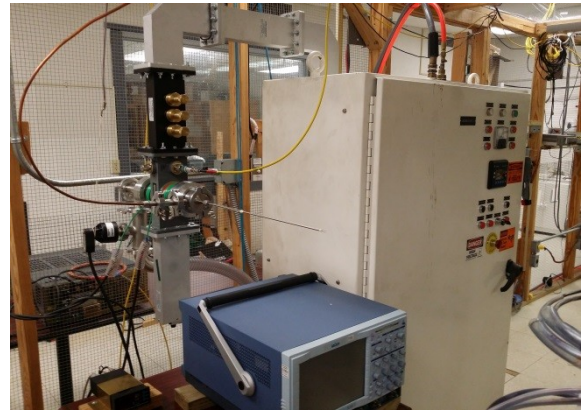
Microwave technology implementation:

A microwave science and technology facility, with emphasis on large power microwave standing-wave research, has been set up as a stage for the development of experiments that have been successfully accomplished recently through the support of DOE.

In 2014, PVSO's fusion plasma experiment group conducted a microwave experiment by adding $6kW/2.45GHz$ microwave through a microwave guide system into the plasma chamber as an additional energy source. Though the power of the microwave is much smaller relative to the existing $300kW$ radio frequency wave, there are several interesting microwave effects observed in these experiments. The experimental results have been published in the journal, *Physics of Plasmas* (Zhou et. al., 2015).



6kW microwave used as additional power source at PVSO' rotamak experiment



Newly built microwave standing-wave fusion plasma experiment system

In 2015, after completing the fusion plasma experiment, we took the $6kW/2.45GHz$ microwave power system off from the fusion plasma experiment facility and used it to build a microwave standing-wave experiment system as the first major research device of our new high-power microwave laboratory. The microwave standing wave experiment device is shown in the above pictures (right).

In 2016, we used the new microwave standing-wave experiment system with a specially designed mini-fusion plasma chamber system with various electric and magnetic diagnostic tools and a computer control system to conduct a new-type of nuclear fusion plasma experiment. The main feature of the experiment is the device with a small volume but having very high energy density and very high electromagnetic wave frequency ($3.45GHz$) energy input. The pilot experiment was successful and very promising for a new direction for fusion plasma research.

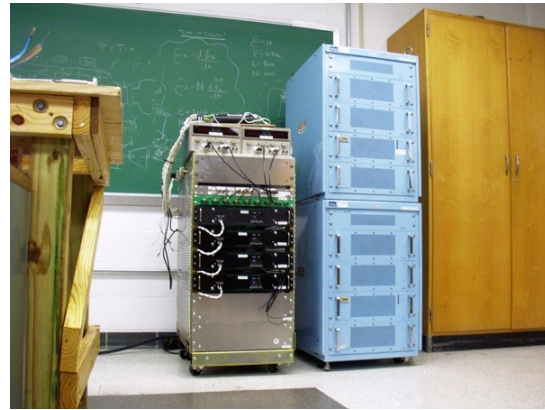
Our high power microwave lab will not be limited only to fusion plasma research. We will work on the studies of microwave's electric field and magnetic field analysis, and their applications in wide science and technology areas including, atomic physics, molecular physics, chemical engineering, and advanced (high frequency) magnetic resonance image studies. We also plan to add a microwave experimental system with a $915MHz/100kW$ microwave generator and associate instruments.

Plasma propulsion experimental study:

Currently, we are building a large-power plasma propulsion experimental system that includes a $5kW$ helicon plasma source, a 25 cm diameter plasma heating chamber with $1MHz/200kW$ RF power rotating magnetic field, and a 60 cm diameter plasma exhaust chamber, as show in the pictures below. In the new system, a helicon plasma is produced with density $n \sim 10^{19}m^{-3}$ and temperature $T_e \sim 2eV$ ($20,000^{\circ}C$). The helicon plasma then enters into the rotating-magnetic field heating chamber where the plasma is expected to be heated to $40eV$ ($400,000^{\circ}C$). The heated plasma then ejects into an exhaust chamber. In the exhaust chamber the ejected plasma's impulse will be measured. We expect to have a plasma mass flow of $0.1g/s$ with $60km/s$ ejection and continue to investigate potential propulsion applications.



Propulsion Experiment Table



Two RF Power Generators

10. Acknowledgements

This report is prepared with explicit recognition of unequivocal support from Department of Energy (DOE) through DE-FG02-04ER54777 from 2004-2015 to Prairie View A&M University (PVAMU) – we acknowledge the support of Dr Barish, Program Manager, DOE and Ms. Tanisha Smith, Grants Management Specialist.

This document is dedicated to Dr Huang, founding director of the Prairie View Solar Observatory (PVSO), who served as a PI on this and all other DOE grants for plasma research until his retirement in June 2016. His dedication and commitment to initiate and expand plasma research at PVAMU is unique and unparalleled to any such establishment around the world. We admire his ingenuity and dedication to build such a facility from which his legacy in plasma research will continue and expand for decades to come at PVAMU.

We also thank all the scientists who worked during this grant tenure for all their contributions in particular – Dr Petrov, Dr Yang, Dr Le Sager, and Dr Zhong.

We recognize dedicated support in preparation of this report by Dr Erickson, Sr Scientist of PVSO and Visiting Professor of Physics; Mr Cudnik, Scientist of PVSO and Manager of Physics Laboratories, Mr Rahman and Ms Kolluri, Doctoral Students in Electrical Engineering for their unwavering time and dedication in all these efforts.

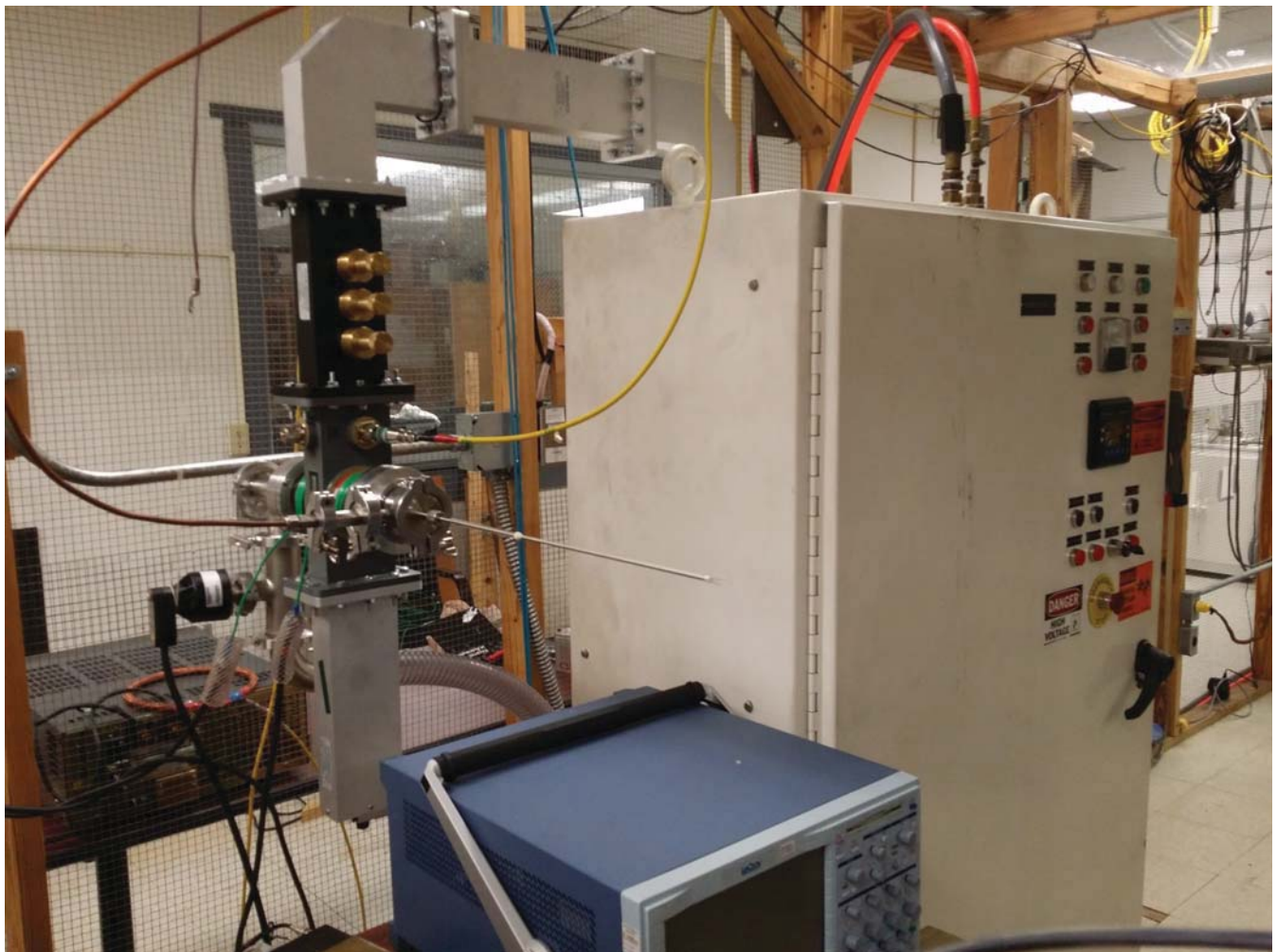
We acknowledge the Office of Research and Sponsored Program support - Dr Krishnamoorthi, Director, and Ms Mosley, Project Administrator II, for their continued support in the required documents needed for the preparation of this report. We thank Dr Kelley, Dean and Dr Keaton, Associate Dean for entrusting this responsibility that empowered us to accomplish this report.

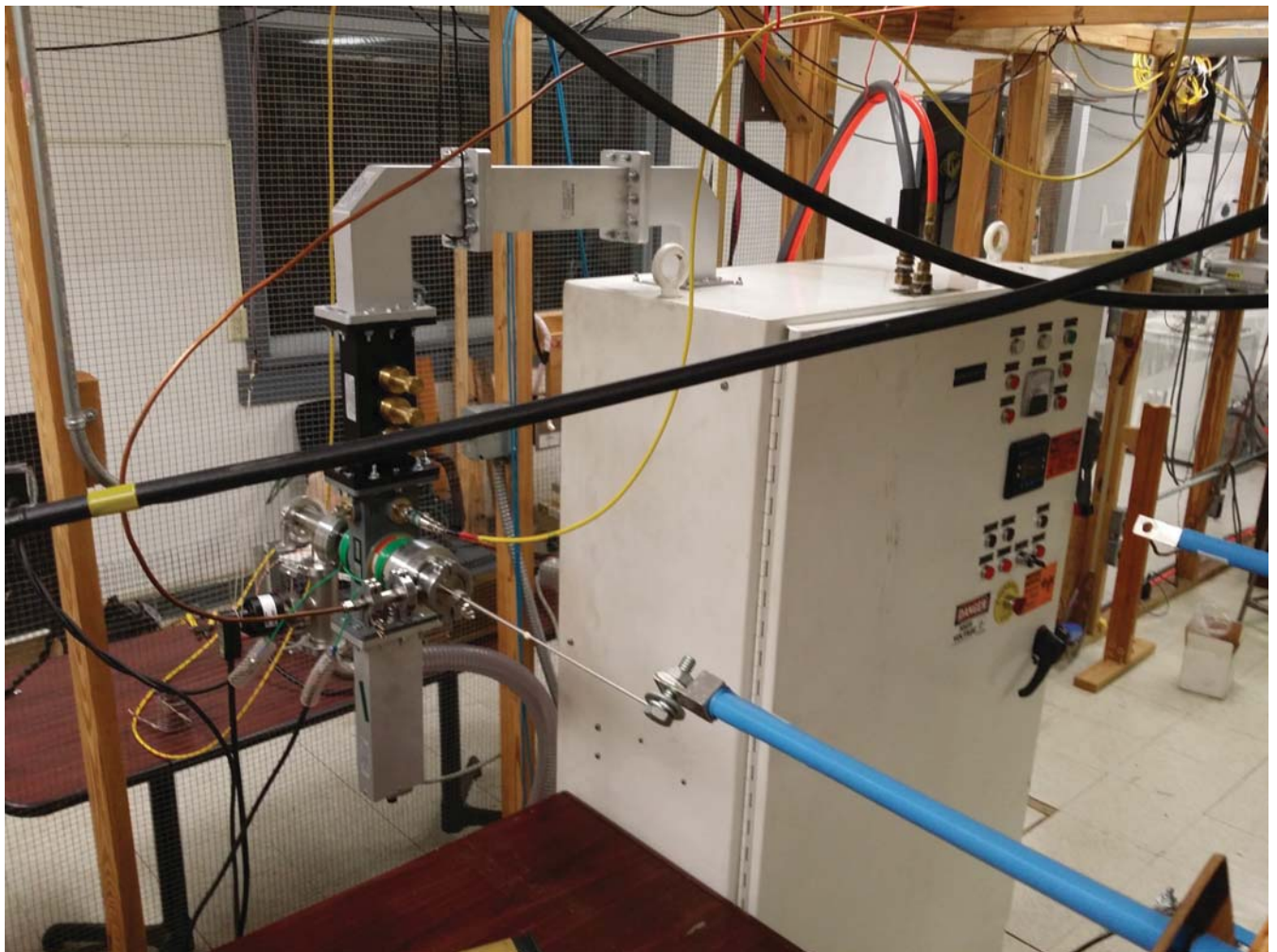
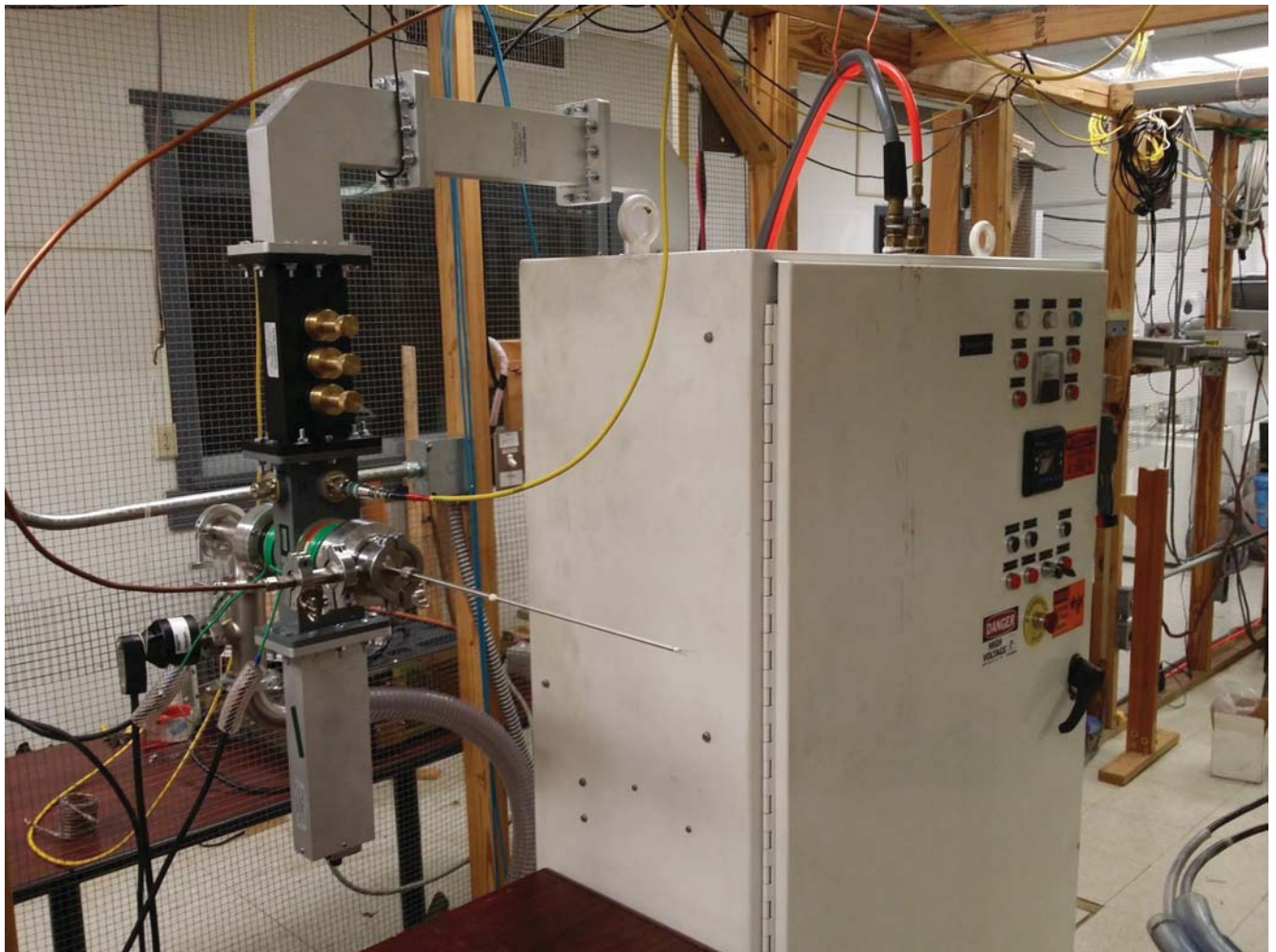
This report is compiled and submitted by Dr Saganti, Regents Professor (TAMU), Professor and Coordinator of Physics, as part of the over arching responsibility as a Research Director of the Chancellor Research Initiative (CRI), Radiation Institute for Science and Engineering (RaISE) at PVAMU. Any further questions or correspondence may be directed to pbsaganti@pvamu.edu

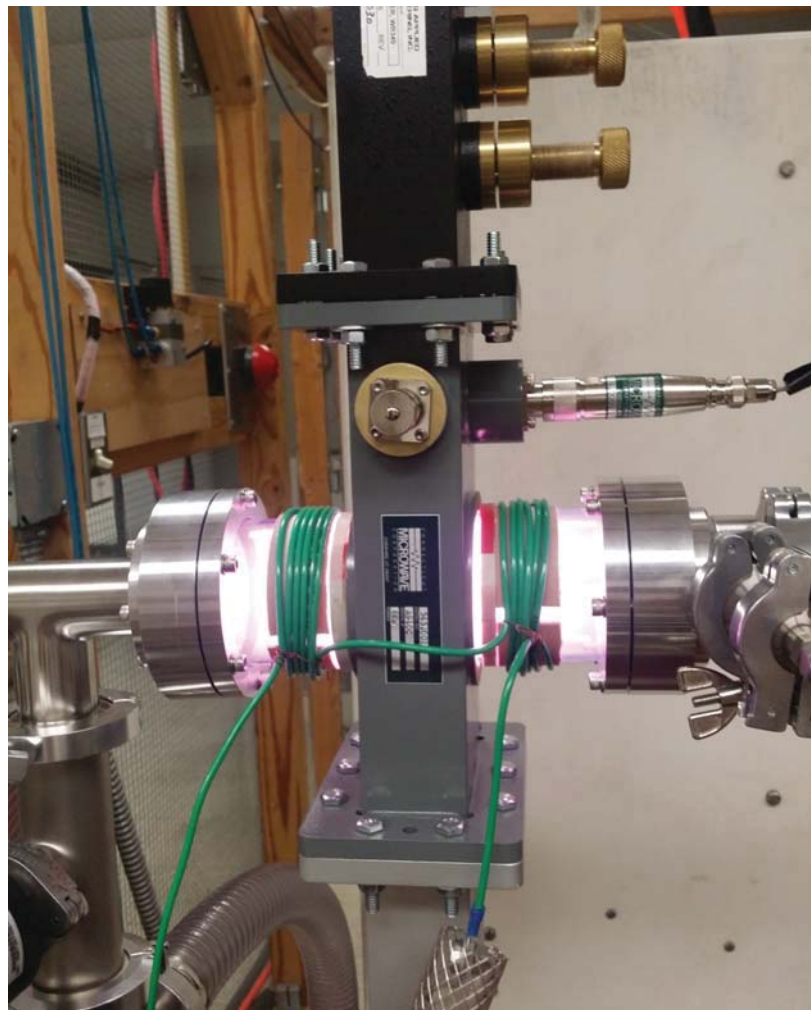
**Appendix-A: Assembled Microwave Plasma
System
(Pictures)**

Assembled Microwave Standing Wave-Fusion Plasma System

Prairie View A&M University
2015







Appendix-B: Propulsion System in Progress

(Pictures)

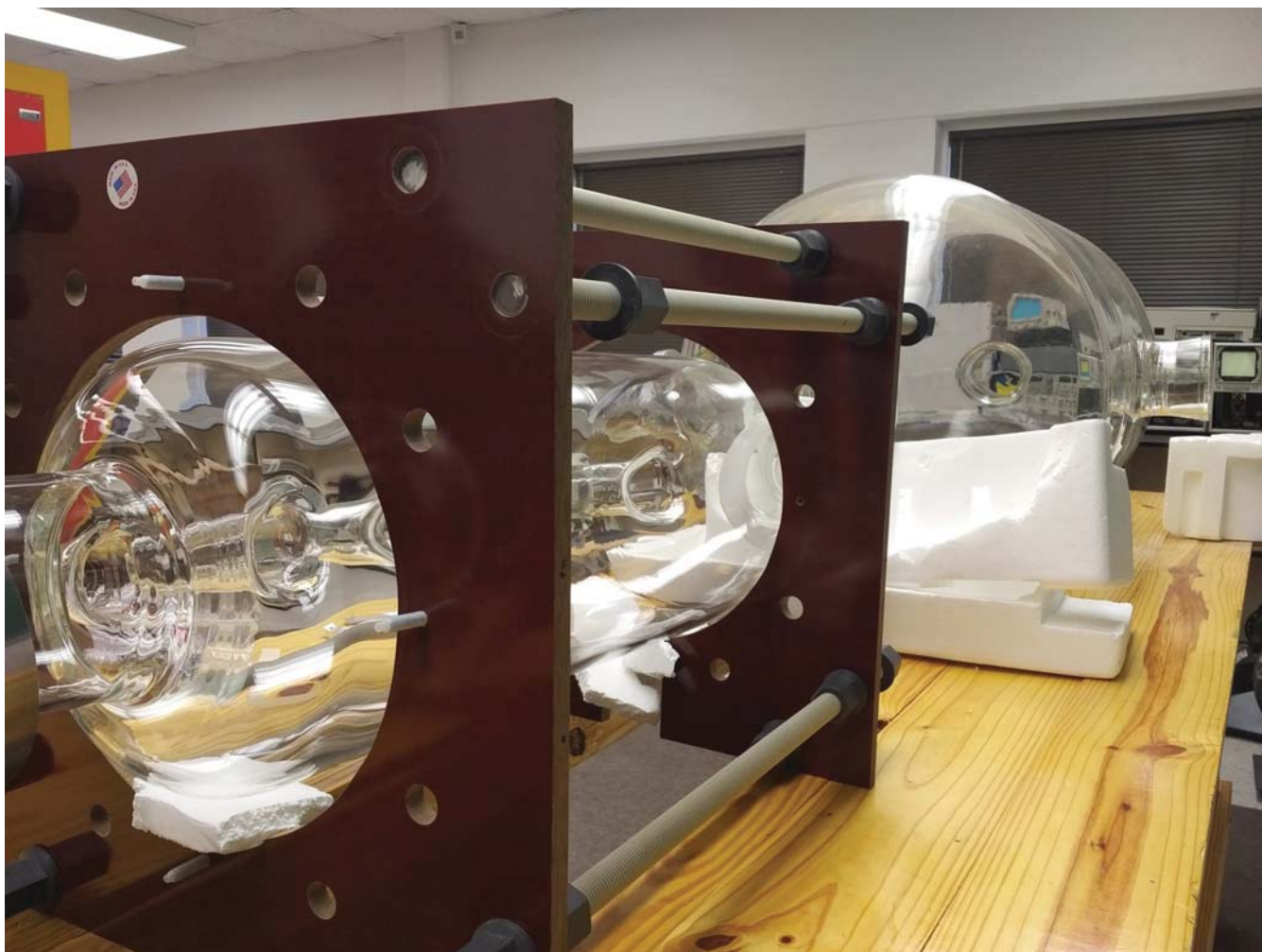
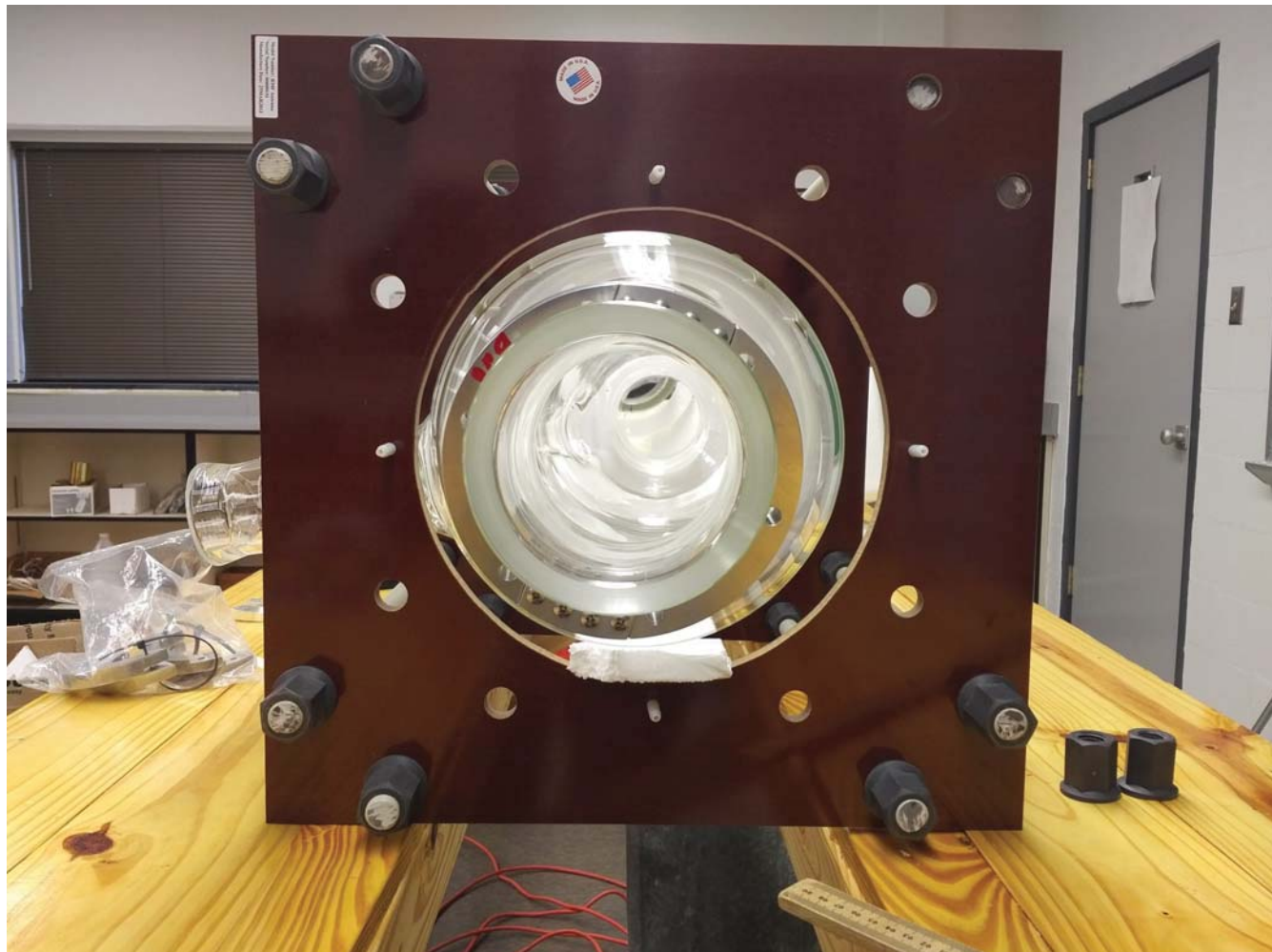
Propulsion System in Progress

Prairie View A&M University
2016



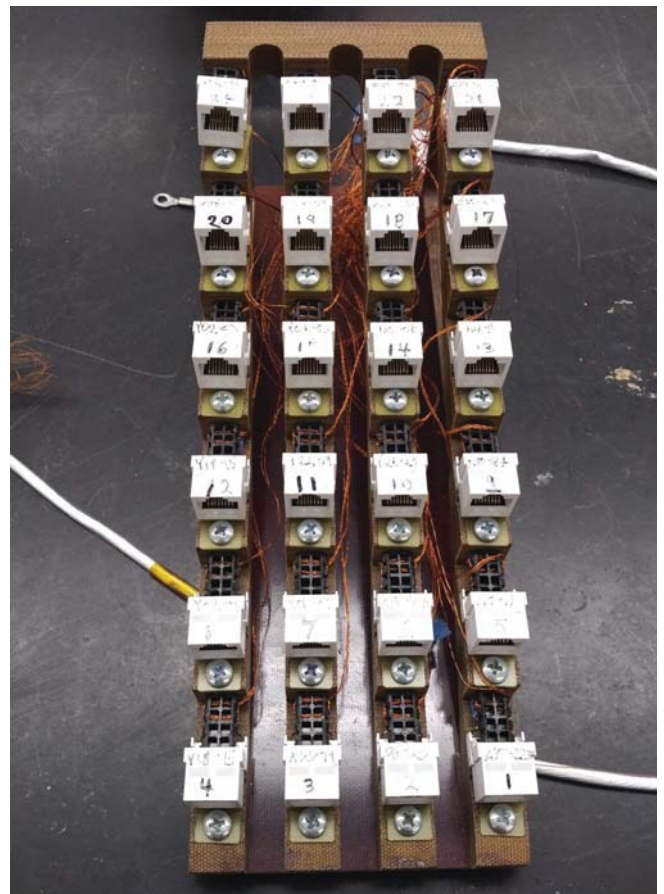
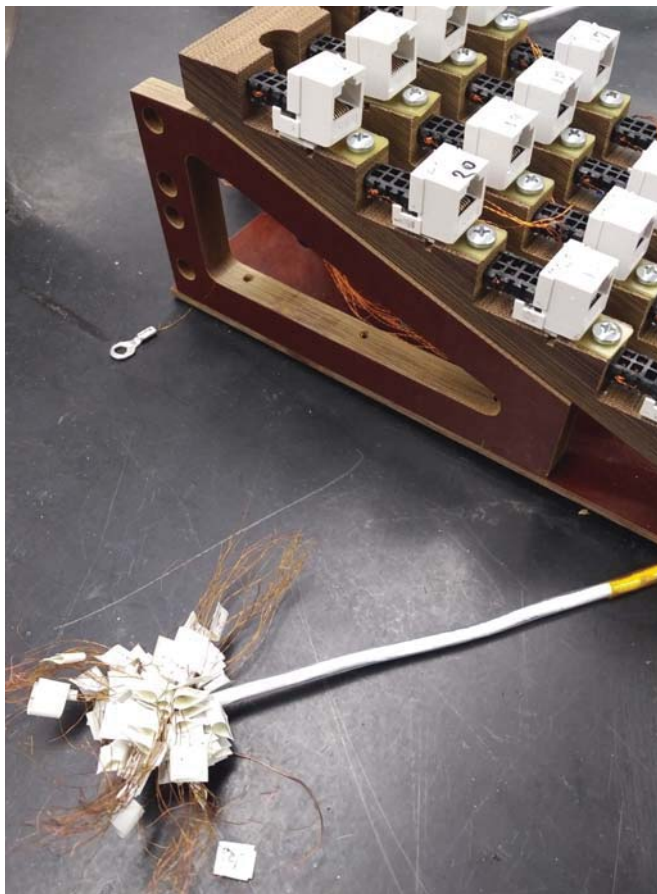


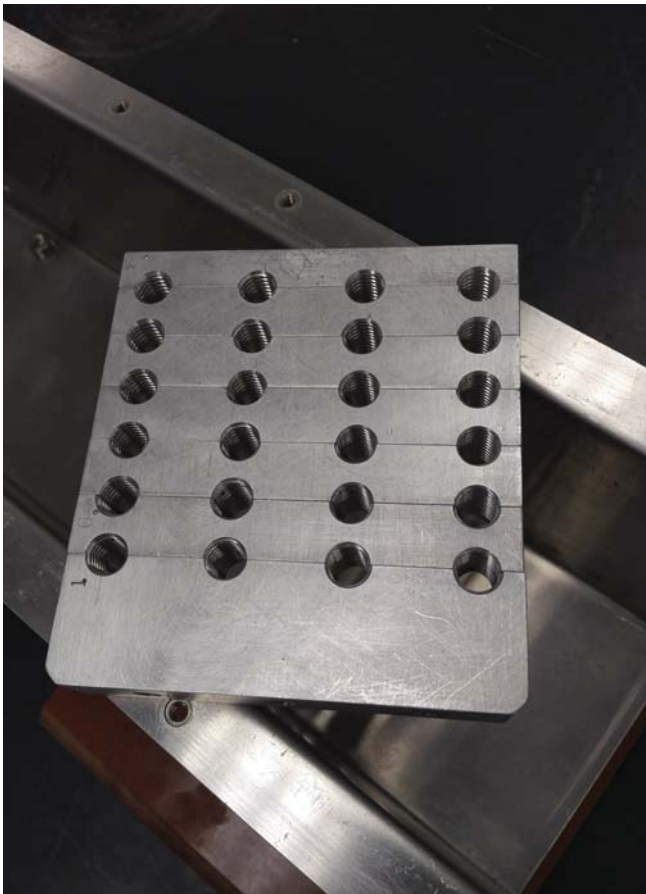
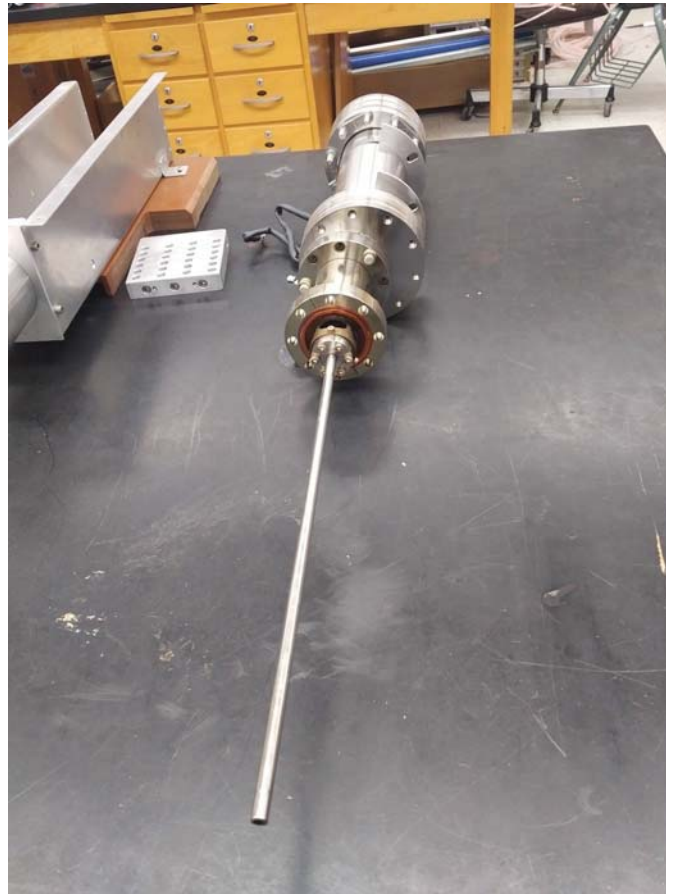
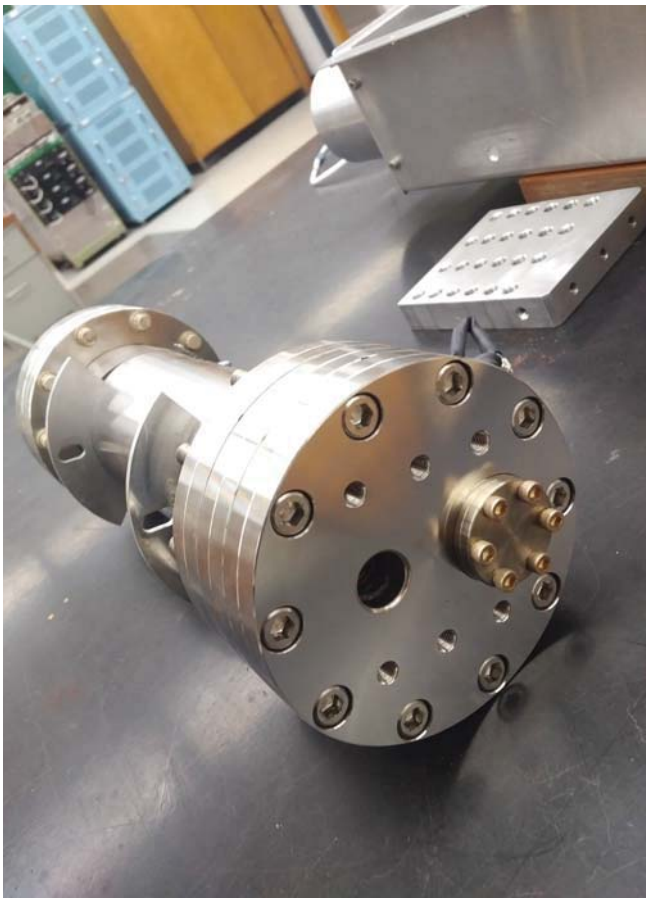












Appendix-C: Relevant Publications

(Re-Prints)

APPENDIX-C: Re-Prints of Relevant Publications

Re-Prints of Relevant Publications: Plasma Research at Prairie View A&M University with Support from DE-FG02-04ER54777 (2004-2015)

- 1. Microwave experiments on Prairie View Rotamak**
R. Zhou, M. Xu, and T.S. Huang
Physics of Plasmas 22, 05450, 2015
- 2. Improved Magnetic Reconnection Experiment at FRC Device**
M. Xu, R. Zhou, D. Vasquez, and T.S. Huang
Abstract Submitted for the DPP14 Meeting of the American Physical Society DPP14-2014-001073, 2014
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- 4. High resolution ion Doppler spectroscopy at Prairie View Rotamak**
S. Houshmandyar, X. Yang, and R. Magee
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SPECIAL ISSUE: Basic Plasma Processes in Solar-Terrestrial Activities April 2012 Vol.57
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- 7. Effect of toroidal magnetic field on $n = 1$ mode stability in Rotamak plasmas**
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- 8. Experiments on Rotamak plasma equilibrium and shape control**
T.S. Huang, X. Yang, Y. Wang, Y. Petrov
Phys of Plasmas, 17, 012506, 2010

9. Suppression of $n = 1$ tilt instability by magnetic shaping coils in Rotamak plasmas

T.S. Huang X. Yang, Y. Petrov

Physical Review Letters, 2009, Lett 102, 255004

10. Magnetic Field structure evolution in rotating magnetic field plasmas

T.S. Huang X. Yang, Y. Petrov

Physics of Plasmas 15, 072509, 2008

11. Comparison of Rotamak plasma discharges in spherical and cylindrical devices

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Plasmas Physics and Controlled Fusion, 50, 085020, 2008

12. The effect of toroidal field on the rotating magnetic field current drive in Rotamak plasmas

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Chinese Physics, Vol 16, 2007

13. Magnetic reconnection of two FRCs driven by rotating magnetic field

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Abstract Submitted for the DPP06 Meeting of the American Physical Society, 2006

14. Comparison of Rotamak plasmas in FRC and ST configurations

T.S. Huang, Y. Petrov, F.C. Zhong

Plasma Physics and Controlled Fusion, 47, 1517, 2005

15. High harmonic fields in a Rotamak plasma

T.S. Huang, Y. Petrov, F. Zhong

Physics of Plasmas 11, L1, 2004

Microwave experiments on Prairie View Rotamak

R. J. Zhou,^{1,2} M. Xu,¹ and Tian-Sen Huang¹

¹Prairie View A&M University, Prairie View, Texas 77446, USA

²Institute of Plasma Physics, Chinese Academy of Sciences, Hefei 230031, China

(Received 6 February 2015; accepted 29 April 2015; published online 14 May 2015)

A 6 kW/2.45 GHz microwave system has been added on Prairie View Rotamak, and a series of experiments with microwave heating in both O-mode and X-mode configurations have been performed. Effective ionization of hydrogen in the two configurations is observed when filling pressure of the hydrogen gas is under $p_f = 0.1$ Pa. Clear oscillations in plasma current I_p and magnetic field B_R are excited when microwaves are injected into plasma in the X-mode configuration. The higher the injected microwave power, the sooner the emergence of the magnetic oscillations in B_R , which implies the microwave may have decreased the elongation of the plasma. In the experiments, the efficiency of the current drive mechanism due to the injected microwave is about 0.2 kA/kW. © 2015 AIP Publishing LLC. [<http://dx.doi.org/10.1063/1.4921129>]

Rotamak is a compact torus configuration having the feature that the toroidal plasma current is driven by means of an externally applied rotating magnetic field (RMF). This plasma current is kept in equilibrium by an externally applied “vertical” field. If the plasma current is large enough, the “vertical” field is reversed on the symmetry axis, resulting in a typical field-reversed configuration (FRC). When a steady toroidal magnetic field is added, the Rotamak is able to operate as a typical spherical tokamak (ST). The Rotamak scheme retains the basic advantages of the general compact torus concept, such as FRC and ST, with further offering a quasi-static formation scheme and the promise of steady-state operation. For this, and other reasons, the Rotamak has attracted much attention, and significant progress has been made in experimental and theoretical studies.^{1–3}

Electromagnetic waves are widely used in fusion plasma experiments to heat plasma via electron cyclotron resonance heating (ECRH) or drive plasma current via electron cyclotron current driving (ECCD).^{4,5} However, typical electromagnetic waves in the electron cyclotron range of frequencies (ECRF) are not effective in plasma confinement devices with low magnetic field, low temperature and high density ($\omega_{ce} \ll \omega_{pe}$, where ω_{pe} , ω_{ce} are, respectively, the electron plasma and electron cyclotron frequencies). The plasma is optically thin for the incident microwave and the energy transfer at the fundamental resonance becomes inefficient.⁶ A special kind of electron cyclotron wave is the electron Bernstein wave (EBW), which is a short wavelength electrostatic wave in a magnetized plasma. EBW does propagate in this kind of plasma, which has the potential for providing localized, highly efficient heating or current drive.^{7,8} The potential of using the EBW for heating such plasmas was recognized in the theoretical work and experiments. The effectiveness of both the obliquely launched ordinary mode (O-mode) and perpendicularly launched extraordinary mode (X-mode) electromagnetic waves has been clearly demonstrated.^{9,10}

Microwaves have been used previously on the Rotamak-E II device as a pre-ionization system.¹¹ Recently, a 6 kW

microwave system was added on Prairie View Rotamak to study mainly the ECRH and ECCD effects of microwave on the Rotamak plasma, with $\omega_{ce} < 2\pi f < \omega_{pe}$, where f is the microwave frequency. Electron Bernstein waves are expected to be excited in the Rotamak plasma through electromagnetic wave mode conversion. A series of microwave experiments in both O-mode and X-mode configurations have been performed on Prairie View Rotamak for modes FRC and ST, respectively.

The Prairie View Rotamak is a rebuilt Rotamak based on the disassembled Flinders apparatus,² with plasma chamber, diagnostic tools, and control system upgrades. It focuses on the comparison of FRC and ST discharges with regard to the magnitude of driven plasma current, the study of active plasma shape control, and investigation of MHD instabilities and magnetic reconnection. A schematic of the Prairie View Rotamak is shown in Figure 1. The new chamber is a 0.8 m long/0.4 m diameter Pyrex glass vessel. The background vacuum in the chamber is maintained at 1.3×10^{-4} Pa. Working gas, which is Hydrogen, is continuously fed through the chamber at a pressure of $p_f \approx 2.7 \times 10^{-2}$ Pa. The RMF is produced by two pairs of coils, which are connected through a matching/tuning network to two 500 kHz/400 kW rf generators with a 90° phase shift between them.

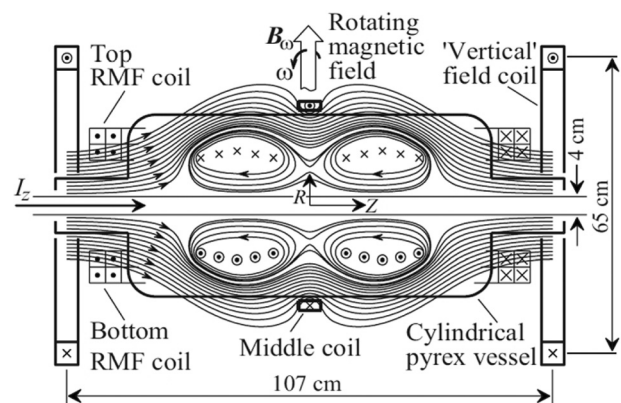


FIG. 1. The schematic of Prairie View Rotamak.

The equilibrium magnetic field system consists of two coils (vertical field coils) with diameter 0.65 m, separated by a distance of 1.07 m. In conducting microwave experiments, the microwave is injected into plasma through an upper right window. The setup of the microwave system in O-mode and X-mode configurations is shown in Figures 2(a) and 2(b), respectively.

The microwave generator is a complete self-contained microwave source. It is rated at a maximum 6 kW continuous power output and operates at the frequency $2.45 \text{ GHz} \pm 15 \text{ MHz}$. The generator uses a water-cooled magnetron to generate the microwave power from 1.5 kW to 6 kW. A 3-port circulator with water load protects the magnetron in the high-reflected power periods. Also, photoelectric arc detection is used to protect magnetron and waveguides.

A set of rectangular waveguides are used to transfer microwave to the plasma. In the waveguide system, two flexible waveguides with maximal insertion loss of 0.02 dB are used to couple with other waveguides. One E-bend corner-cut waveguide and one straight waveguide are used with maximal insertion loss of 0.20 dB. In addition, one 3-stub waveguide tuner is used for matching the plasma impedance. The total insertion loss of this waveguide system should be less than 6%. A vacuum sealed ceramics placed between the waveguides and the plasma chamber is used to enable high microwave power transmission and keep the high vacuum level in the Rotamak chamber. Copper meshes are used around the Rotamak chamber to prevent microwave from leaving the plasma chamber. The size of each mesh is about $0.2 \text{ m} \times 0.2 \text{ m}$ and a thickness of about 0.5 mm, and the distance between any two meshes is less than 50 mm. Each mesh is composed of a grid of about 1.5 mm spacing, much smaller than the microwave wavelength ($\lambda_{2.45 \text{ GHz}} \approx 122 \text{ mm}$). Connecting the meshes to each other would adversely affect the coupling between RMF and the plasma. In the experiments, the copper meshes are separated from each other to reduce this effect.

The dominant mode of the waveguide is TE_{10} . By adjusting the 3-stub tuner, the reflected microwave power can be decreased to 10% during the microwave experiments. Our experiments have been operated for both the O-mode and X-mode configuration. In the experiments, the plasma current is measured with a Rogowski coil, while the plasma temperature and density are measured with a double floating Langmuir probe. An array of 32 Mirnov coils installed on

the outside chamber surface is used to observe magnetic instabilities during the experiments.

A series of microwave experiments in both O-mode and X-mode configurations have been performed on Prairie View Rotamak. In the O-mode configuration, effective ionization of hydrogen has been observed when the filling pressure of the hydrogen gas is under $p_f = 0.1 \text{ Pa}$. However, we cannot drive plasma current by RMF when the filling pressure of the hydrogen gas is under $p_f = 0.1 \text{ Pa}$. Apart ionization, the plasma does not seem to respond to the injected 2.45 GHz microwaves in the O-mode configuration.

The density of the Rotamak plasma is high. The cut-off density of the X-mode is higher than the O-mode, thus the X-mode can be absorbed more effectively in the Prairie View Rotamak. Concerning power absorption of the X-mode microwave in the experiments, two scenarios are possible: direct absorption of X-mode microwave or direct coupling of the X-mode to the EBW and subsequent absorption of the EBW. In the X-mode configuration, effective ionization of hydrogen can also be observed when the filling pressure of the hydrogen gas is under $p_f = 0.1 \text{ Pa}$. Also, clear oscillations of the plasma current I_p and magnetic field B_R are excited when the microwaves are injected into the plasma in the X-mode configuration. Figure 3 shows four typical discharges with injected microwave power $P_{MW} = 0, 1, 2$, and 3 kW, respectively. These four shots are performed with the same discharge parameters except for the injected microwave power. Because the microwave power is relatively low compared to the typical power of the RMF generators, no obvious change in the plasma temperature and density is observed in the experiments. But clear oscillations of the plasma current I_p and magnetic field B_R are excited when microwaves are injected into the plasma.

Shot 1 is a reference discharge without input microwave power. The plasma current is about $I_p \approx 1.3 \text{ kA}$, with plasma density $n_e \approx 5 \times 10^{17} \text{ m}^{-3}$ and temperature $T_e \approx 11.5 \text{ eV}$. There are no obvious magnetic instabilities in this discharge condition. When microwaves are injected into plasma, clear oscillations in plasma current I_p and magnetic field B_R are excited, which can be seen in shots 2, 3, and 4.

Microwaves are injected into plasma during the entire discharge period, and oscillations in plasma current exist during the entire discharge period. The time-frequency spectra of the oscillations from plasma current signal in shot 1 and shot 2 are shown in Figures 4(a) and 4(b), respectively.

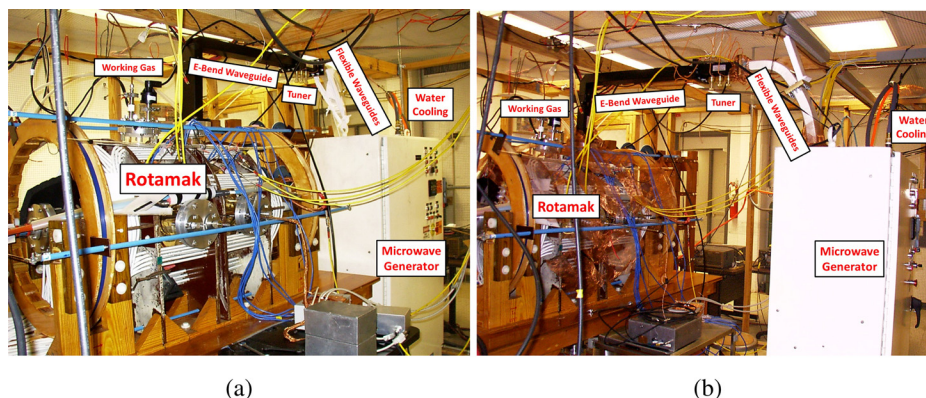


FIG. 2. Setup of the microwave system in (a) O-mode and (b) X-mode configurations.

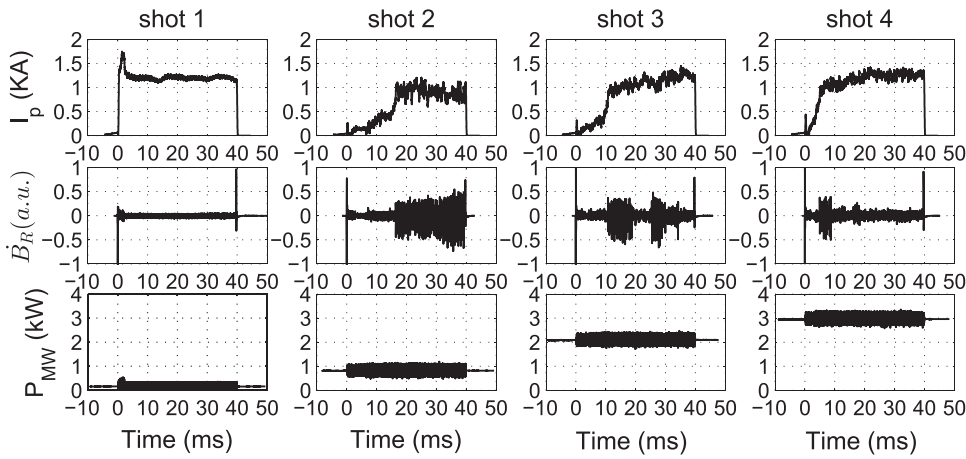


FIG. 3. Four typical discharges with injected microwave power $P_{MW} = 0$ kW in shot 1, $P_{MW} = 1$ kW in shot 2, $P_{MW} = 2$ kW in shot 3, and $P_{MW} = 3$ kW in shot 4.

In the figures, the observed oscillation at 8 kHz is noise, which comes from the vertical field's power supply system. This noise can also be observed during shot 1. In comparison with shot 1 when no microwaves are injected into plasma, oscillations with a wide frequency range of 0–5 kHz can be observed when microwaves are injected into plasma. This implies that the oscillations in plasma current resulted from the injected microwaves.

Figure 5 shows the time evolution of I_p and B_R , along with the B_R spectra from the Mirnov coils during shot 2. Oscillations in magnetic field B_R have much wider frequency range below 20 kHz and only emerged after a few milliseconds during the discharges. The higher the injected microwave power, the sooner the emergence of the magnetic oscillations in B_R . Those magnetic oscillations are identified as $n = 1$ radial shift mode (the phase shift between two sets of Mirnov coils is π). According to theory, the radial shift mode is expected to become unstable at sufficiently small elongation $E < 0.7$ ($E = Z_S/R_S$, with Z_S and R_S the separatrix half-length in the axial direction and separatrix radius, respectively).¹² This implies that the injected microwave may have decreased the elongation of the plasma. When the microwave power is higher, it can decrease the elongation of the plasma sooner to the threshold value. Then the radial shift mode emerges. Usually excitation of radial shift mode results in plasma disruption within 1 ms.¹³ But as seen from Figure 3, it is difficult to build a flat plasma current at the beginning of those discharges until the emergence of the magnetic oscillations in B_R . So, it implies the excitation of radial

shift mode seems to favor the building of the flat plasma current in our microwave experiments. We will further study this phenomenon in our future work.

Comparing the flat top phase of plasma current in shots 2, 3, and 4, it shows plasma current $I_p \approx 0.9$, 1.1, and 1.3 kA when injected microwave power $P_{MW} \approx 1$, 2, and 3 kW, respectively. The plasma current in flat top phase is higher when the injected microwave power is larger. It seems the microwave system is operating as a plasma current driving mechanism rather than a plasma heating mechanism. The efficiency of current driving of the microwave system is about 0.2 kA/kW in our experiments.

In summary, a microwave power injection system has been added on Prairie View Rotamak. The water-cooled microwave generator is rated at a maximum 6 kW continuous power output and operates at a frequency of 2.45 GHz. The total insertion loss of the waveguide system is less than 6%. By adjusting a 3-stub tuner, the reflected microwave power can be decreased to 10% with copper meshes surrounding the Rotamak chamber. Depending on the direction of the waveguide system with respect to the Rotamak's vertical field, the microwave can be launched into the plasma in O-mode or X-mode configuration.

A series of microwave experiments have been performed in both O-mode and X-mode configurations. Effective ionization of hydrogen is observed when the filling pressure of the hydrogen gas is under $p_f = 0.1$ Pa in both O-mode and X-mode configurations. Oscillations in plasma current I_p and magnetic field B_R are excited when the

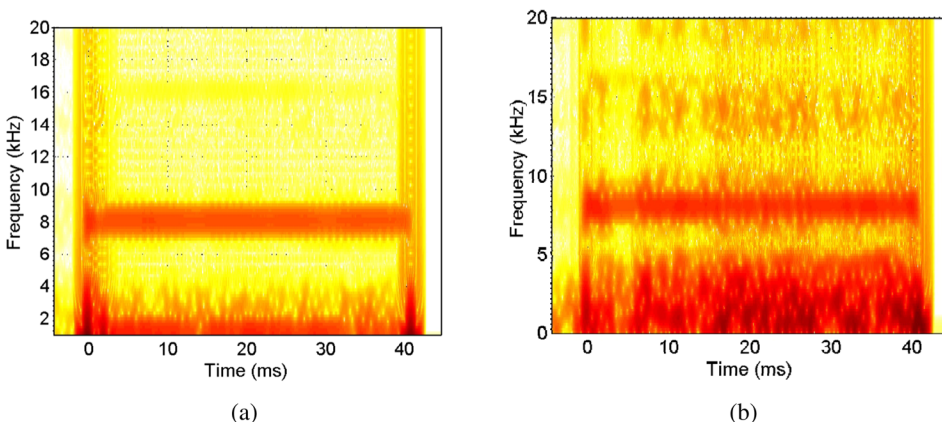


FIG. 4. Time-frequency spectra of the oscillations from plasma current signal in shot 1 (a), and shot 2 (b).

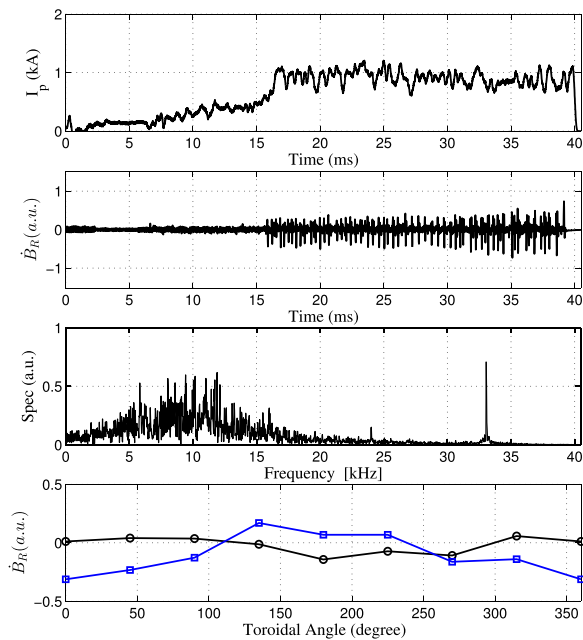


FIG. 5. Time evolution of I_p , B_R , spectra for B_R , and signals from two sets of Mirnov coils during shot 2.

microwave is injected into plasma in the X-mode configuration. The excited magnetic oscillations are identified as the $n = 1$ radial shift mode. The higher the injected microwave power, the sooner the emergence of the magnetic oscillations in B_R , which implies that the microwaves may have decreased the elongation of the plasma. The efficiency of the

current drive mechanism due to the injected microwave is about 0.2 kA/kW in the microwave experiments.

In further work, an upgraded microwave generator with higher power will be considered to get larger plasma temperature increase.

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Improved Magnetic Reconnection Experiment at FRC Device¹ MING XU, RUIJIE ZHOU, DANIEL VASQUEZ, TIAN-SEN HUANG, Prairie View AM University, PRAIRIE VIEW SOLAR OBSERVATORY TEAM — With experimental facility's improvement, magnetic reconnection has been further studied at Prairie View rotamak device. By adding one toroidal current in the central part of the rotamak device, the cutting of one magnetic field reverse configuration (FRC) as two FRCs in the experiment process becomes more obvious. Differing from the magnetic reconnection experiments conducted at other labs, where magnetic reconnection is formed with two wire-coiled currents buried in a chamber with large scale magnetic field, in our magnetic reconnection experiment the main source of the magnetic field is plasma current. Thus, the magnetic reconnection experiments conducted at rotamak device are closer to the one occurring in the space and on the sun. At the present stage, our experiments focus on the study of the change in electron temperature during the magnetic reconnection process. Furthermore, the ion temperature and plasma flow can be easily achieved from fast ion Doppler spectroscopy (IDS) diagnostic system [1], which makes the magnetic reconnection process more clearly. 1. Petrov, Y., X. Yang, et al., Physics of Plasmas, 15(7): 072509-072508 (2008) 2. Houshmandyar, S., X. Yang, et al., Review of Scientific Instruments 83(10): 10D506 (2012)

¹Improved Magnetic Reconnection Experiment at FRC Device

☐ Prefer Oral Session
☒ Prefer Poster Session

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High sensitivity, low noise Mirnov coil array on Prairie View rotamak^{a)}

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An array of 32 Mirnov coils with novel features of high sensitivity and low noise has been installed on the outside chamber surface of Prairie View rotamak. This B_R -oriented coil array has proven to be very reliable in the plasma driven by rotating magnetic field; it can resolve magnetic perturbation signals of 0.1 G. With this new diagnostic, the $n = 1$ tilt, radial shift, and kink modes are observed for the first time in rotamak plasmas. © 2012 American Institute of Physics. [<http://dx.doi.org/10.1063/1.4729675>]

I. INTRODUCTION

Rotamak is a compact torus configuration having the distinctive feature that the toroidal plasma current is driven in a steady state, noninductive fashion by means of an externally applied rotating magnetic field (RMF).¹ The experiments have proven that with this method, a sufficiently large plasma current can be generated to reverse the external equilibrium magnetic field on the symmetry axis so that a field-reversed configuration (FRC) is formed.^{2,3} Rotamak can operate in two regimes: the FRC (without external toroidal field) and the spherical tokamak configuration when a steady toroidal magnetic field (provided by an axial current I_z) is added. One of the most important issues about the FRC plasma is its stability to low- n (n is the toroidal number) MHD modes. This issue is extremely crucial when the rotamak is extrapolated to a fusion reactor. However, although rotamak research has been conducted theoretically and experimentally at Flinders University for more than two decades, the study on MHD instability was lacked because research interests were focused on the mechanism of RMF current drive.³ Nevertheless, the investigation of MHD instability is one of the important tasks on Prairie View (PV) rotamak.⁴

To conduct the experimental study of MHD instability, there is a need to develop nonperturbing diagnostics to detect the possible low- n instability modes in rotamak plasma. Mirnov coil array, which has been widely used in toroidal machines such as tokamak and spherical tokamak, has never been attempted on rotamaks and other machines^{5,6} in which the plasma current is driven by RMF. The previous concern of using Mirnov coil array for RMF-driven plasmas is that the magnetic perturbation signals might be too small to be detected, because (a) the MHD theory predicts that the presence of Hall current can provide additional stabilization for the tilt mode,⁷ (b) the instability modes observed with side-on interferometer on translation, confinement, and sustainment machine are relatively far from the chamber wall,⁵ (c) the RMF has stabilization effect on some low- n modes.⁵ Therefore, it

is a challenge to observe instability modes in the RMF-driven plasmas with Mirnov coil array. In this paper, the Mirnov coil array installed on PV rotamak is reported; three instability modes detected with the array are also presented.

II. MIRNOV ARRAY SYSTEM

Due to the unique property of rotamak plasma, high sensitivity, and shield against rf interference are special considerations when Mirnov coil array is designed. Installation of Mirnov coils around the chamber surface requires accurate orientation (B_R or B_θ) and alignment. Traditional rotamak uses a small spherical glass chamber; this makes it difficult to keep coil's accurate orientation and alignment on two sides of the mid-plane. The new PV rotamak, as illustrated in Fig. 1, is the first rotamak that has a cylindrical chamber. This chamber's modification makes coil's accurate installation easier than before. Different from those Mirnov array systems^{8,9} which usually use B_θ coils, we choose B_R over B_θ because the RMF has a small $\tilde{B}_R$ but a large $\tilde{B}_\theta$ component; in addition, rotamak FRC plasma has a relatively large self-generated B_θ component. With B_R -oriented coils, the magnetic perturbations can be easily detected as a difference from small signals; thus, coil's sensitivity is increased. There are four sets of coils being installed, respectively, at $Z = \pm 4$ and ± 30 cm along the axis of cylindrical chamber. At each z position, eight B_R coils are mounted around the outside chamber surface ($R = 20$ cm) at an equal interval of 45° toroidal angle.

The method to increase coil's pick-up sensitivity is to design coils with more turns. Each coil installed on PV rotamak consists of 1000 turns (38 gauge magnet wires) on a form with a square of $0.5 \text{ cm} \times 0.5 \text{ cm}$ and a length of 0.6 cm ; the coil sensitivity is 250 cm^2 , or $2.5 \times 10^{-3} \text{ Vs/G}$. The digitizer has a sensitivity of 1.12 mV/bit , therefore the array system can detect the changes in magnetic field around 500 G/s , and it can resolve magnetic perturbation signals of 0.1 G .

There are two types of rf sources used on PV rotamak: a 600 W , 13.56 MHz rf generator to preionize plasma and two 400 kW , 500 kHz generators to produce RMFs. Thus, attentions have been put on shielding rf interference and filtering rf signals. The twisted leads from each coil are covered by copper tubing; the outer braid of each triax cable and the connection box are wrapped with thick copper foil. After many

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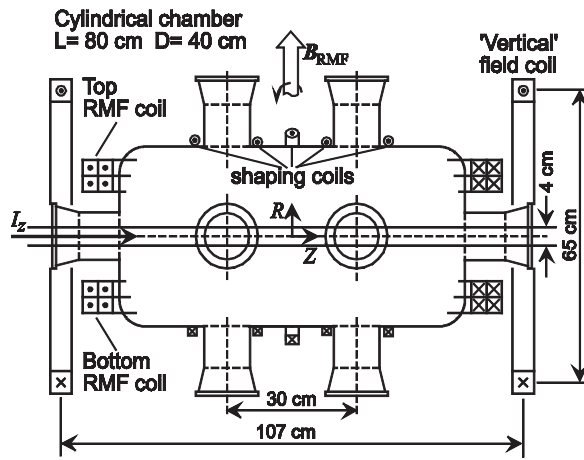


FIG. 1. The schematic of the PV rotamak.

trials for effective shielding, the array system can provide almost noise-free signals. Since the oscillating magnetic signals detected by Mirnov coils are often in the range 10–20 kHz, a low pass filter bank (below 50 kHz, for 32 channel coil signals) is used, thus the rf signals from generators are filtered.

From the beginning, one data acquisition board of NI PXI-6254 (32-channel) with 1 MHz sampling rate and 16 bit resolution was used. But in each plasma shot, this board can be used for maximum 16 coils because its sampling rate is not enough for all 32 coils. Now this board has been replaced by two NI PXIE-6368, which has 16 simultaneous analog inputs at 2 MS/s/Ch with 16 bit resolution. All signals from the coils are passively integrated and digitized at 1 μ s intervals.

The off-line analysis described in the reference ⁸ is used to process the B_R signals from the coils at four z locations. First, a numerical method is used to subtract the B_z background by comparing the measurements of a B_z probe at the same position and condition. After that, the Fourier analysis is used to obtain the amplitudes and the phases of instability modes with toroidal number $n = 0, 1, 2$, and 3, and to separate the axially even and odd components of the modes. Thus, MHD mode structure has been determined through Fourier analysis.

III. EXPERIMENTAL MEASUREMENTS

Experiments were typically conducted in 40 ms rotamak-FRC discharge with plasma current $I_p \approx 2.0$ kA. As shown in Fig. 2(a), the $n = 1$ mode is the dominant one that often appears in rotamak-FRC plasma discharge. Judging by the small phase shift between signals from two sets of Mirnov coils ($Z = -4$ cm and $+4$ cm), it is identified as a tilt mode. This mode rotates in electron direction; it usually grows within a period of 1–2 ms and then becomes saturated during the shot. Depending on the value of equilibrium magnetic field (“vertical field,” B_v) and filling gas pressure, the oscillation frequency of tilt mode is between 12 and 20 kHz; the growth time can be up to 10 ms (about 2000 of Alfvén transit time). This tilt mode is non-destructive, but it usually results in plasma current drop by 10%. As indicated in Fig. 2(a) and 2(b), when B_v is increased from 25 G to 28 G, the tilt mode starts later and grows more slowly, and its frequency and am-

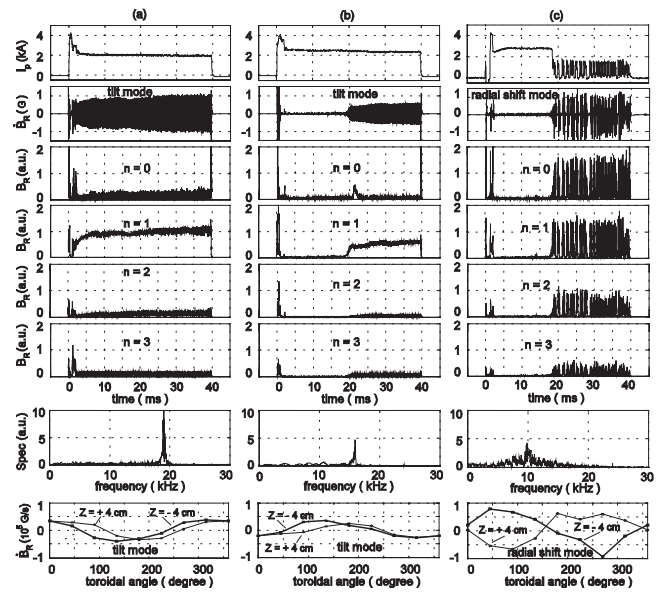


FIG. 2. Time evolution of plasma current, MHD oscillation signal, magnitude of $n = 0$ –3 perturbations to B_R , spectra, and signals from two sets of coils for (a) $B_v = 25$ G; (b) $B_v = 28$ G; (c) $B_v = 31$ G.

plitude gradually becomes smaller with higher B_v . When B_v is slightly larger than 30 G, tilt mode disappears, but another mode with oscillation frequency around 10 kHz appears. This mode is identified as $n = 1$ radial shift mode (the phase shift between two sets of Mirnov coils is π). Usually excitation of radial shift mode triggers plasma disruption within 1 ms as shown in Fig. 2(c). This result clearly demonstrates that the current termination observed in previous experiments¹⁰ is due to the excitation of radial shift mode when vertical field is above a critical value. Radial shift mode limits the achievable plasma current in rotamak-FRC discharges.

In experiments it is observed that the filling gas pressure has effect on tilt mode. As shown in Fig. 3, when hydrogen gas pressure increases, the tilt mode appears late in the discharge, its amplitude reduces and its oscillating frequency

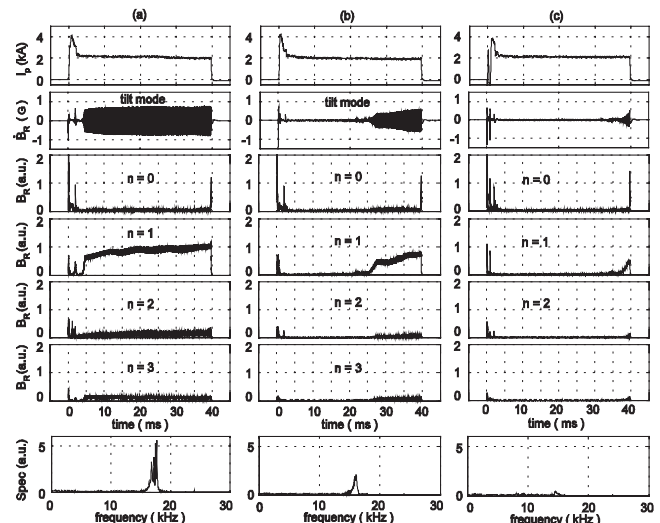


FIG. 3. Time evolution of plasma current, MHD oscillation signal, magnitude of $n = 0$ –3 perturbations to B_R and spectra for hydrogen gas pressure (a) $p_f = 1.3$ mTorr; (b) $p_f = 1.6$ mTorr; (c) $p_f = 1.8$ mTorr.

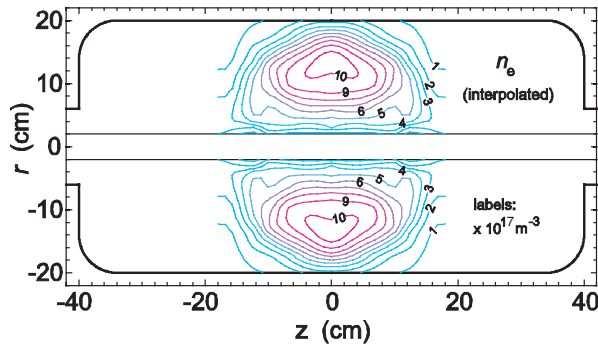


FIG. 4. Two-dimensional distribution of electron density in typical rotamak discharge.

downshifts. At higher gas pressure, the tilt mode disappears but plasma discharge becomes difficult to start as shown in Fig. 3(c) at $t = 0$ –2 ms. Therefore, gas pressure is one of the important factors in the study of MHD instability in rotamak-FRC plasma.

It is also noted that the signals detected by the coils located at $Z = \pm 30$ cm are too small to distinguish the perturbations of magnetic field. Two factors can explain this result: as shown in Fig. 4, the rotamak plasma is located mainly within the area of $|z| < 15$ cm, the position of maximum density is at $z = 0$ cm/ $r = 15$ cm; those two sets of coils are far from the middle plane due to the limitation of accessible space in rotamak machine. Under this situation, only two sets of coils located at $Z = \pm 4$ cm can provide useful information regarding MHD activities. Therefore, the coils at $Z = \pm 30$ cm are not frequently used for the study of MHD instability, unless the magnetic shaping coils are applied to extend plasma shape along the axial direction.¹¹

The transition experiments are conducted by ramping axial current I_z at $t = 0$ ms until end of discharge, while keep-

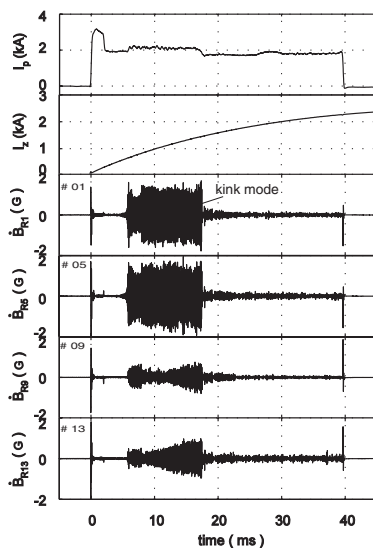


FIG. 5. Time evolution of plasma current, axial current, MHD oscillating signals of left side ($Z = -4$ cm) coil #1 and #5, and right side ($Z = +4$ cm) coil #9 and #13.

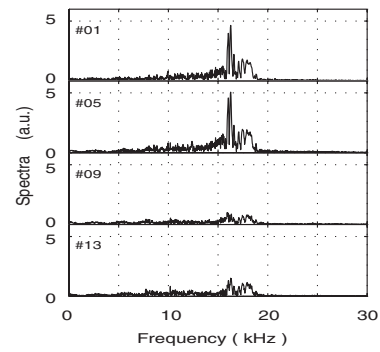


FIG. 6. MHD mode spectra detected by left side ($Z = -4$ cm) coil #1 and #5 and right side ($Z = +4$ cm) coil #9 and #13.

ing other parameters same as in the standard FRC discharge. As shown in Fig. 5, the discharge is free from any instability mode when I_z is below 0.5 kA. When I_z ramps above 0.5 kA, the $n = 1$ kink mode appears at $t = 6.5$ ms, grows quickly and almost saturates within 1 ms. The signal's amplitudes detected by the left side ($Z = -4$ cm) coils are about twice of the right side ($Z = +4$ cm) coils; here only two coil signals from each array are shown. This measurement result shows that kink mode has a strongly asymmetric perturbation structure along the middle plane. As seen from Fig. 6, kink mode has almost the same frequency as tilt mode, but with a wider frequency band. Kink mode continues to present in the discharge when I_z is in the range of 0.5–1.4 kA; it suddenly disappears when I_z is larger than a threshold value around 1.5 kA, as seen in Fig. 5. The transition experiment clearly demonstrates that the stability of kink mode strongly depends on toroidal magnetic field, or magnetic shear. In contrast to tilt mode which usually results in plasma current drop by 10%, kink mode can boost plasma current by 25% as long as it appears in the discharge.

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High resolution ion Doppler spectroscopy at Prairie View Rotamak^{a)}

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A fast ion Doppler spectroscopy (IDS) diagnostic system is installed on the Prairie View Rotamak to measure ion temperature and plasma flow. The diagnostic employs a single channel photomultiplier tube and a Jarrell-Ash 50 monochromator with a diffraction grating line density of 1180 lines/mm, which allows for first order spectra of 200–600 nm. The motorized gear of the monochromator allows spectral resolution of 0.01 nm. Equal IDS measurements are observed for various impurity emission lines of which carbon lines exhibit stronger intensities. Furthermore, the diagnostics is examined in an experiment where plasma experiences sudden disruption and quick recovery. In this case, the IDS measurements show $\sim 130\%$ increase in ion temperature. Flow measurements are shown to be consistent with plasma rotation. © 2012 American Institute of Physics. [<http://dx.doi.org/10.1063/1.4728091>]

I. INTRODUCTION

Spectroscopy techniques based on Doppler broadening and Doppler shift of the emission lines are widely used for ion temperature and flow measurements in low temperature plasmas,^{1–3} edge of fusion devices,⁴ and core tokamak measurements.⁵ Ion Doppler spectroscopy (IDS) is a routinely used plasma diagnostic that is based on line of sight average measurements and it has been well described in the literature.⁶ The fast response of the IDS systems has made them suitable diagnostics where ion heating is expected, e.g., magnetic reconnection experiments in spheromaks^{7,8} and reversed field pinch devices.⁹ IDS has also been used for measurements of supersonic toroidal $E \times B$ flow at Maryland Centrifugal Experiment,¹⁰ radially sheared axial flow at ZaP experiment,¹¹ and magnetic relaxations of helicity injections in compact torus experiment.¹²

As a limitation, IDS measurements rely on the radiation from impurity emission lines, rather than the majority species. Therefore, the ion temperature (a proxy) and flow must be inferred from these measurements. Nevertheless, it has been shown that the IDS measurements are consistent with other majority species temperature measurements.⁸

This paper describes the newly developed system of IDS diagnostic and some preliminary results on the Prairie View (PV) Rotamak. As a new diagnostics, the IDS system can provide useful information about the radial and toroidal flows and ion temperature measurements in magnetohydrodynamic (MHD) instability studies and plasma shape control via external feedback system, as well as magnetic reconnection.

II. EXPERIMENTAL SETUP

The PV-Rotamak is a compact torus configuration that can operate in field reversed configuration (FRC) or spherical tokamak (ST) regimes. Its toroidal plasma current is driven by a distinct feature which employs two pairs of external rotating magnetic field (RMF) in a non-inductive fashion. The RMF is produced by two 90° -dephased rf generators. Each generator can feed up to 400 kW of power (P_{rf}) to one pair of coils at frequency of 500 kHz. Another two coils provide a maximum external equilibrium (vertical) magnetic field (B_v) of 230 G (at the vessel's center). The RMF and vertical magnetic field windings embrace a cylindrical chamber. The chamber is an 80-cm long PyrexTM vessel with 40 cm ID. A KF-25 quartz window was installed on the feedthrough port at $z = 15$ cm. Fig. 1 shows the schematic of the PV-Rotamak. A Pfeifer Balzers TPU 510 turbo pump maintains a background pressure of $\sim 10^{-7}$ Torr. In the ST configuration (which is the case for the experiments reported here), a steady toroidal magnetic field (B_ϕ) is produced by an axial current, I_z , and it is measured by a Rogowski coil. This axial current is produced by discharging a capacitor bank through six wires that are pulled through the central axial stainless steel pipe, which is covered with a quartz tube. The RMF is applied for a 40 ms period;

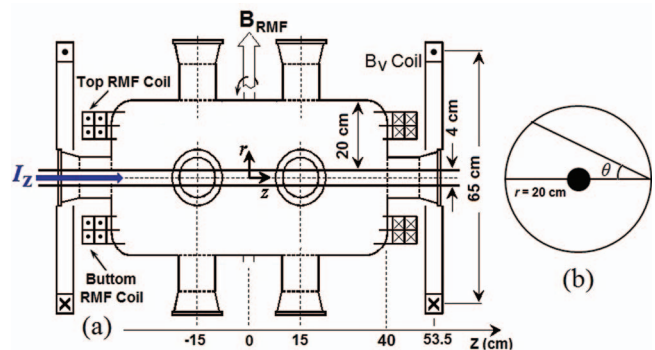


FIG. 1. (a) Schematic and (b) midplane ($z = 0$) cross section of the PV-Rotamak; the angle represents the chord direction as explained in Sec. IV.

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during this period, B_v and B_ϕ remain constant within 5%. A 650 W/13.56 rf generator pre-ionizes the hydrogen gas (flowing continuously into the chamber at filling pressure of 1.1 mTorr) for 13 ms; it has 7 ms overlap with the RMF. Five sets of magnetic shape control coils are wound close to the chamber's surface, of which the one at $z = 0$ (eight turns) is used for these experiments.¹² A Kepco ATE-6-100M provides a maximum current of 800 A to the middle coil. The plasma current (I_p) was measured by a Rogowski coil, which is pulled through the central pipe around the poloidal cross section. Electron temperature (T_e) and density (N_e) are measured with a double floating Langmuir probe. For the experiments reported here, $P_{rf} = 150$ kW, $B_v = 25$ G, and $I_z = 250$ A. Shown in Fig. 2 are the typical I_p , T_e , and N_e measurements at PV-Rotamak, when the middle coil was charged at $t = 10$ ms, for the rest of the discharge period.¹³

The IDS system used in this report includes a Jarrell-Ash 50 monochromator (JA) with a ruled diffraction grating of 1180 lines/mm. A telescope was built to collect collimated light with the ability of pivoting to perform measurements for different plasma chords. Another optics was designed to have an optimum light coupled into the monochromator. A fiber optics connects the two optics, which conveys the light from the plasma to the monochromator. The lenses inside of the two optics and the fiber optics material were chosen such that the combination can analyze the spectral lines from UV through visible. However, the PyrexTM chamber blocks the UV spectrum. Therefore, the quartz window was used for the spectral lines below 380 nm. A high gain, single channel photomultiplier tube (PMT) was mounted on the exit slit

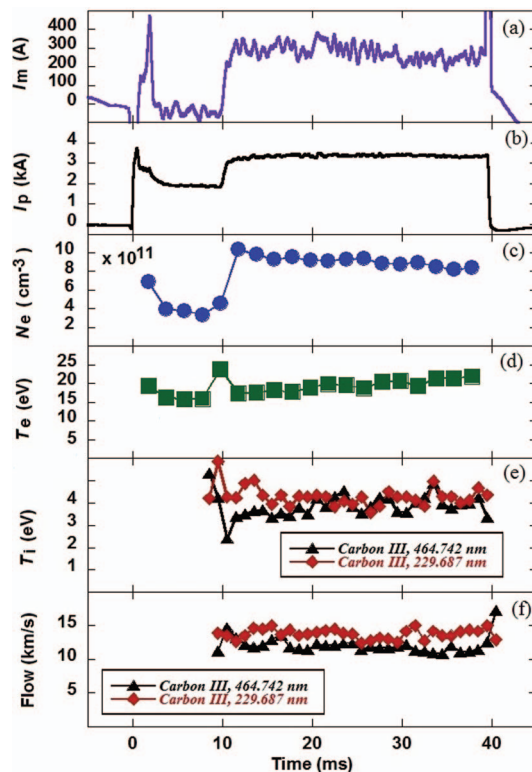


FIG. 2. Middle coil current (a), plasma current (b), electron density (c), and electron temperature (d) at midplane. The IDS measurements for ion temperature (e) and flow (f) with 1 ms integration time are performed through the quartz window at $z = 15$ cm.

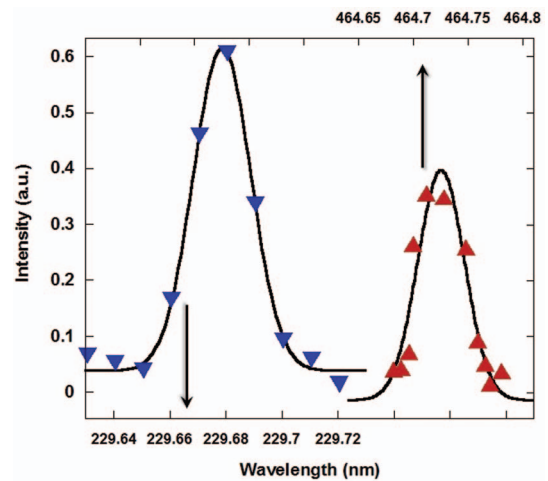


FIG. 3. Lineshape for C III impurity emission lines at 229.687 nm and 464.743 nm at $t = 35.538$ ms. The Gaussian fitting for temperature and flow derivations shows 4.05 eV and 11.74 km/s, and 4.28 eV and 10.7 km/s, respectively.

of the monochromator. A transimpedance amplifier was used to amplify the PMT signal with a gain of 10^4 V/mA. It was then recorded by a PCI-5112 NI-Scope at 100 MHz digitation rate, and $(5-50) \times 10^5$ record length. Hence, a spectrum (e.g., Fig. 3) is built up over several discharges. An HC-1 Hg-Ar calibration lamp was used for the monochromator calibrations. The motorized gears of the monochromator allowed wavelength resolution of 0.01 nm. The instrument temperature for carbon lines was measured to be 1.34 eV.

III. IDS MEASUREMENTS WITH DIFFERENT IMPURITY LINES

Carbon, oxygen, and silicon impurity emission lines were observed in plasmas discharges at PV-Rotamak. IDS measurements with O V at 278.101 nm, O V at 278.699 nm, Si II at 290.569 nm, C III at 229.687 nm, C III at 464.742, and C III at 465.025 nm showed equal temperature and flow measurements within 15% and 8% uncertainties, respectively. Fig. 3 shows lineshape comparison of two carbon lines at 229.687

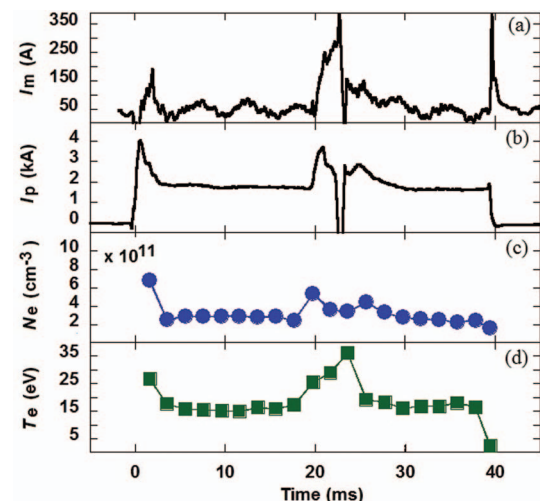


FIG. 4. Middle coil current (a), plasma current (b), electron density (c), and electron temperature (d). The middle coil was charged at $t = 20$ ms, for a 3 ms period.

nm and 464.742 nm at 35.538 ms. These measurements were performed through the quartz window. As the plasmas produced in PV-Rotamak are mostly located at midplane ($z = 0$), there was not enough light to be collected from the plasma. Thus, the middle coil was charged to elongate the plasma to higher axial extents.¹³ Figs. 2(e) and 2(f) show comparisons between measurements for temperature and flow for two different emission lines; the measurements are performed with 1 ms integration time.

IV. FAST IDS MEASUREMENTS

The IDS diagnostic installed on the PV-Rotamak has a minimum temporal resolution of $0.1 \mu\text{s}$. This resolution allows fast ion temperature variation investigations in plasma shape control experiments and ion heating observations for future magnetic reconnection experiments. In order to examine this high temporal resolution, IDS measurements were performed for a case in which the middle coil was charged to 250 A at $t = 20$ ms, for a 3 ms period. As a result, the plasma was severely disrupted (almost terminated) and then (surprisingly) recovered to its conditions prior to the disruption. Shown in Fig. 4 are the plots of middle coil current (I_m), plasma current (I_p), electron density (N_e), and electron temperature (T_e) as functions of time. Here the IDS measurements were performed at the midplane (through the PyrexTM), for the carbon line at 464.742 nm. Shown in Fig. 5 are the plots of ion temperature measurements with $1 \mu\text{s}$ temporal resolution, for plasma chords of $+40^\circ$, $+20^\circ$, 0° (on-axis view), -40° and -20° ; the angles and their signs are with respect to the on-axis view (Fig. 1(b)). These measurements show increase in the temperature up to $\sim 90\%$, before and after the disruption (e.g., 21 or 25 ms) for all of the chords except the 0° chord in which the ion heating ($\sim 130\%$) occurs in a very short period, right before the plasma termination and after the plasma reconstruction, i.e., ~ 23.1 and 23.25 ms. This behavior is yet to

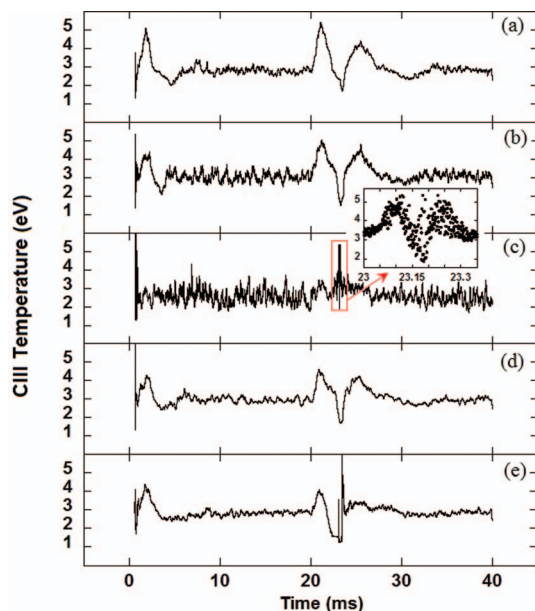


FIG. 5. Ion temperature measurements for $+40^\circ$ (a), $+20^\circ$ (b), 0° (c), -20° (d), and -40° (e) chords, with $1 \mu\text{s}$ temporal resolution. The zoom-in panel shows a $\Delta t = 350 \mu\text{s}$ span.

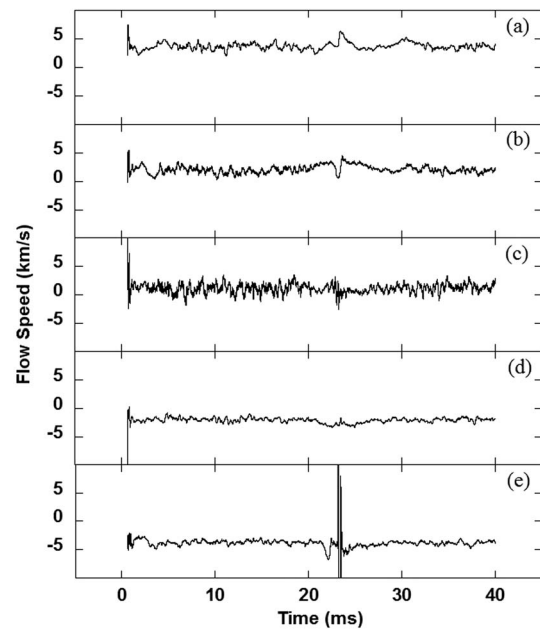


FIG. 6. Flow measurements for $+40^\circ$ (a), $+20^\circ$ (b), 0° (c), -20° (d), and -40° (e) chords, with $1 \mu\text{s}$ temporal resolution.

be investigated with higher time resolution. Shown in Fig. 6 are the plots of flow measurements with $1 \mu\text{s}$ temporal resolution, for plasma chords of $+40^\circ$, $+20^\circ$, 0° , -40° , and -20° , before and after the disruption. The direction of the flow (positive or negative sign) is consistent with the chords' direction and thus the plasma rotation.

V. SUMMARY

Here we report the first ion temperature measurements at the PV-Rotamak by installing a newly developed, high temporal resolution ($0.1 \mu\text{s}$) ion Doppler spectroscopy (IDS) system. Equal temperature and flow measurements are measured for different impurity emission lines. The high temporal resolution of the diagnostics was examined for a case of plasma disruption and recovery, which exhibits ion heating up to $\sim 130\%$.

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Euler potentials' discontinuity in the presence of field line-aligned currents

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Since their use in the study of charged particle motion in the 1960s, Euler potentials (α , β) have been widely employed as magnetic field coordinates in both space plasma and fusion plasma studies. People related them to magnetic vector potential $\mathbf{A}$ via the relation $\mathbf{A} = \alpha \nabla \beta$, subject to gauge condition $\mathbf{A} \cdot \mathbf{B} = 0$ ($\mathbf{B}$ is the magnetic induction). For a given magnetic field, the Euler potentials are often constructed with the relation $\mathbf{B} \cdot \Delta S = \Delta \alpha \Delta \beta$ on a surface that crosses the field lines, where ΔS is the area-element surrounding by two line-elements corresponding to the changes in α and β , then mapping the values of α and β along field lines into space. In this short paper, we show that in the presence of field line-aligned currents, the mapping does not work and the orthogonality gauge condition is not satisfied.

plasmas, space physics, Euler potential, drift motion

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By representing magnetic field as $\mathbf{B} = \nabla \alpha \times \nabla \beta$, Grad [1] was the first to use two scalar functions α and β as the field line coordinates, and referred to them as Euler potentials [2]. In Northrop and Teller's famous paper [3], the (α , β) coordinates were used in proving the second adiabatic invariant and deriving the dynamic equations of a charged particle:

$$\langle \dot{\alpha} \rangle = -\frac{1}{q} \frac{\partial K}{\partial \beta}, \quad \langle \dot{\beta} \rangle = \frac{1}{q} \frac{\partial K}{\partial \alpha},$$

where q is the particle's charge and K is sum of the particle's kinetic energy and electric potential energy plus a quantity $q\alpha\partial\beta/\partial t$. Euler potentials have since been widely employed in studying space plasma physics (e.g. [4–6]). The potentials are also used in the study of fusion plasma physics under the name of Clebsch coordinates [7]. The applicability of Euler potentials as magnetic field coordinates has been presented [4,8], but no strict proof has been presented. Below, we give an example to show that in the

presence of field line-aligned current, the Euler potentials are discontinuous whereas the magnetic field $\mathbf{B}$ and the vector potential $\mathbf{A}$ have no discontinuity.

1 Some concerns about the application of Euler potentials

In his mathematics text book [9], Phillips showed that a solenoidal and continuously differentiable vector function, such as a magnetic field $\mathbf{B}$ can be expressed as a cross-product of two gradients, $\mathbf{B} = \nabla \alpha \times \nabla \beta$. Northrop [8] gave one simple argument for the introduction of α and β as magnetic field line coordinates; that is roughly the same as Phillips' proof. Both Phillips' proof and Northrop's argument invoked the assumption that all magnetic tubes in the magnetic field maintain their integrity as they extend along the field lines, i.e. there exist two families of surfaces, $\alpha(x, y, z) = c_1$ and $\beta(x, y, z) = c_2$, for which surface intersections give all field lines. However, the assumption is not true in

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general (see the following section). In a strictly mathematical sense, setting the relationship $\mathbf{B} = \nabla\alpha \times \nabla\beta$ or $\mathbf{A} = \alpha\nabla\beta$ is dubious, although it is true for the simpler magnetic fields.

It is worth noting that for a given magnetic field, people often use the relation $\mathbf{B} \cdot \Delta S = \Delta\alpha\Delta\beta$ to establish an α - β grid on a surface S that crosses the field lines, then map the values of α and β on the surface along the field lines into the space. We note that in doing so the relation $\mathbf{A} = \alpha\nabla\beta$ might not be explicitly considered.

Unlike Northrop and Teller's paper [3], in which Euler potentials were used for time-dependent magnetic field, the magnetic fields in most studies applying Euler potentials are time-independent. People start with setting $\mathbf{A} = \alpha\nabla\beta$, introducing $\mathbf{A} \cdot \mathbf{B} = 0$ as a new gauge (e.g. [4,5]). However, no one imposes the restricted gauge transformation condition, such as those associated with the Lorentz gauge or Coulomb gauge.

We also note another related question that, if the vector potential $\mathbf{A}$ can be represented by Euler potentials α and β , then an electromagnetic field would be described with just three parameters α , β and ϕ instead of the four parameters A_x , A_y , A_z and ϕ . The physics of this seems unclear.

2 An example where α and β are inapplicable in the presence of field line-aligned current

In the presence of field line-aligned currents, a magnetic field might be divided into several zones, and the field lines in different zones could spiral in different ways. Across the boundary surfaces of contiguous zones, the Euler potentials could be discontinuous, although the vector potential and magnetic field might not have discontinuities on those surfaces.

One example is shown in Figure 1. Three distinct families of field lines that define three disjoint zones result from the fields of two straight currents I and $2I$ embedded in a uniform background field $\mathbf{B}_0$ oriented in the same direction as the current. On the boundary surfaces between the contiguous zones, the field lines either side make contact but spiral upward in distinct ways. Hence, the field lines in each of the different zones must be defined by different Euler potential systems. Another similar result can be drawn for two currents flowing in opposing directions.

The discontinuity in Euler potential discussed above has major consequences on charged particle's drift motion. For the sake of simplicity, the magnetic field showed in Figure 1 is one non-confinement to the charged particle motion. In

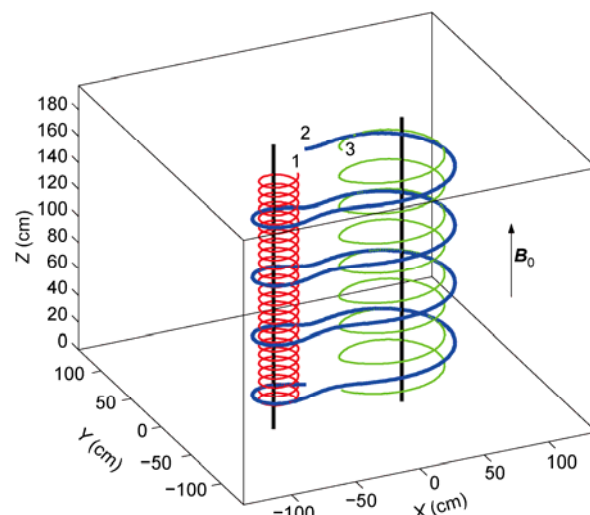


Figure 1 (Color online) Three families of field lines resulting from two straight current in a uniform $\mathbf{B}$ field.

the charged particle-confinement field, the Euler potential discontinuity such as one occurred in magnetic reconnection events makes particle drift complicated: the motion of particles limited inside each field line zone maintains relative independence, but particles drifting across the boundaries between the zones could lead to interactions of the plasmas located in these zones.

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Boost of plasma current with active magnetic field shaping coils in rotamak discharges

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A set of magnetic shaping coils is installed on the Prairie View (PV) rotamak for the study of active plasma shape control in the regimes with and without toroidal field (TF). In the spherical tokamak regime (with TF), plasma current I_p can be boosted by 200% when all five shaping coils (connected in series) are energized. The enhancement of current drive efficiency is mainly attributed to the radial compression and the substantially axial extension of the plasma column; this in turn improves the impedance matching and thus increases antenna input power. In the field-reversed configuration (without TF), plasma current can be boosted by 100% when one middle coil is used; the appearance of radial shift mode limits the achievable value of I_p . The experiments clearly demonstrate that the plasma shape control plays a role in effectively driving plasma current in rotamaks. © 2011 American Institute of Physics. [doi:10.1063/1.3622206]

The rotating magnetic field (RMF) was used in a compact torus device called Rotamak¹ to drive a steady state, non-inductive plasma current. The RMF-driven plasmas are typically formed as a spherical compact torus,^{2–4} or an elongated field-reversed configuration (FRC),^{5,6} mainly depending on the geometry of the equilibrium field coils and the confinement coils. It might be more difficult to implement the equilibrium field control system in RMF-driven plasmas, because the change of magnetic shape not only changes the value of $\mathbf{B}$ in the $\mathbf{J}_p \times \mathbf{B}$ force acting on plasma but also changes the power coupling from the rf antenna to the plasma. This in turn will change the magnitude of plasma current and its profile $\mathbf{J}_p$.

The study of active plasma shape control has been conducted recently on Prairie View (PV) rotamak. The preliminary experiments used three shaping coils and four copper rings, some interesting results were obtained, and a model was presented to explain the change in plasma shape and pressure.⁷ However, the initial experiments were not successful in the FRC regime due to the appearance of radial shift mode when more coils were used. Furthermore, the model could not explain the significant response of plasma current to the shape change.

Meanwhile, it has been noted in our experiments that when the shaping coil is energized in the FRC regime, the RMF appears to have a shallower penetration (2 cm near the chamber wall), similar to the property of the higher performance mode observed in the Translation, Confinement, and Sustainment (TCS) facility when the flux conserving copper straps are wrapped around the quartz tube.⁸ However, in our case, the plasma current is substantially boosted while in TCS there is no change. This phenomenon seems to be contrary to the mechanism of RMF current drive, which requires a good penetration to enhance current drive. Thus, the study of how the shaping coil affects the profile of current density is also needed.

To clarify those issues, we experimentally compare how the change of external magnetic shape affects plasma current in two operational scenarios: the FRC regime, which has no

toroidal field (TF), and the spherical tokamak (ST) regime, which has an added TF. Experimental results clearly indicate that the response of plasma to the change of magnetic field shape is substantially different in ST and FRC regime.

The cylindrical chamber PV Rotamak is illustrated schematically in Ref. 9. In the experiments discussed here, the radial and toroidal components of RMF are $\tilde{B}_r \approx 5 - 10$ G and $\tilde{B}_\theta \approx 10 - 20$ G at the chamber edge. The equilibrium field is produced by two “vertical” field coils. In these experiments, the vacuum field at the chamber center is set to 25 G and keeps the same value in both regimes. In the ST discharge, the toroidal field (25 G at the chamber edge) is added by feeding a 2.5 kA axial current I_z through the central stainless-steel pipe.

The recent system modification includes five magnetic shaping coils for plasma shape control. The shaping coils are wound close to the chamber surface and connected to a programmable current source with current-rise-time 1–2 ms. One “middle” coil (eight turns) is located at $z = 0$ and four “side” coils (four turns each) at $z = \pm 8$ cm and $z = \pm 30$ cm. The boundary poloidal flux is measured with six flux loops wound over the chamber surface at $z = \pm 3.5$ cm, ± 7.5 cm, and ± 22 cm. In addition, the new real time rf power measurement allows us to trace the total antenna input power and to calculate the effective impedance of the plasma (R_{eff}) seen by the rf amplifiers.

Experiments have been carried out in both regimes with different combinations of shaping coils being applied at $t = 20$ ms until end of the discharges. It is observed that the response of the plasma current to the change of magnetic field depends strongly on the coils’ position and total current in the coils ($I_{\text{coil}} = i_{\text{feed}} \times n_{\text{turn}}$). Moreover, the discharge appears to be less stable in the FRC regime when only side coils are used (middle coil being disconnected). Therefore, to simplify the comparison, in this report, we discuss only the three cases which include the middle coil: (a) one (middle) coil; (b) three coils (one middle and two side coils at $z = \pm 8$ cm); (c) five coils (one middle and four side coils).

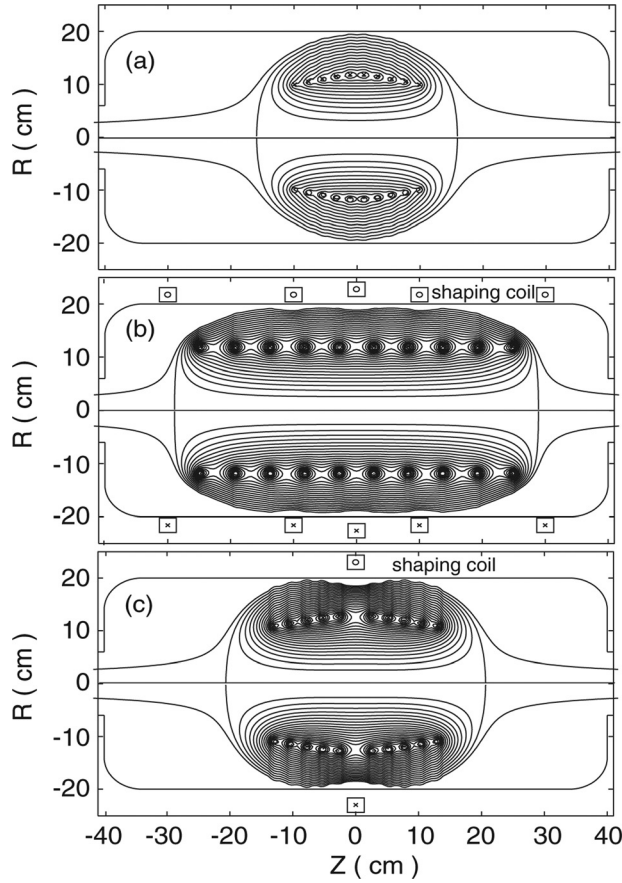


FIG. 1. Reconstructed plasma shape in the discharge of (a) the ST regime, (b) the ST regime with one middle coil and four side coils, and (c) the FRC regime with one middle coil.

Based on the measurements of the boundary poloidal flux profile and the total plasma current, we use the reconstruction process⁷ to plot the plasma shape. The plasma current I_p is represented by ten current loops; the radial and axial positions of the loops are adjusted until the boundary flux profile matches the experimental measurements. Fig. 1(a) shows the reconstructed plasma shape in the ST regime without the shaping coils. This plasma shape is confirmed by both an electron density scan with a Langmuir probe³ and a direct scan of the magnetic field with an internal magnetic probe.⁴ It is noted that the rotamak-ST and FRC plasmas have a similar shape, the minor difference is the plasma elongation (the ratio of the separatrix half-length to its radial extension) $E=1.0$ in the ST regime while $E=0.9$ in the FRC regime. As seen from Fig. 1(a), the closed magnetic surfaces are quite short along the z -axis, in spite of the fact that the RMF antenna extends to the end of the chamber. The power measurements show that the antenna input power is around 130–140 kW, which accounts for 70–80% of the generator output power.

In rotamak-ST experiments, plasma discharges are remarkably stable when shaping coils are applied. No instability modes have been detected by Mirnov coils. In the three cases with the middle coil, the coil current is gradually increased shot by shot until the discharges are terminated. The best shot in each case is plotted in Fig. 2. In the three cases, the antenna input power is almost identically 175 kW,

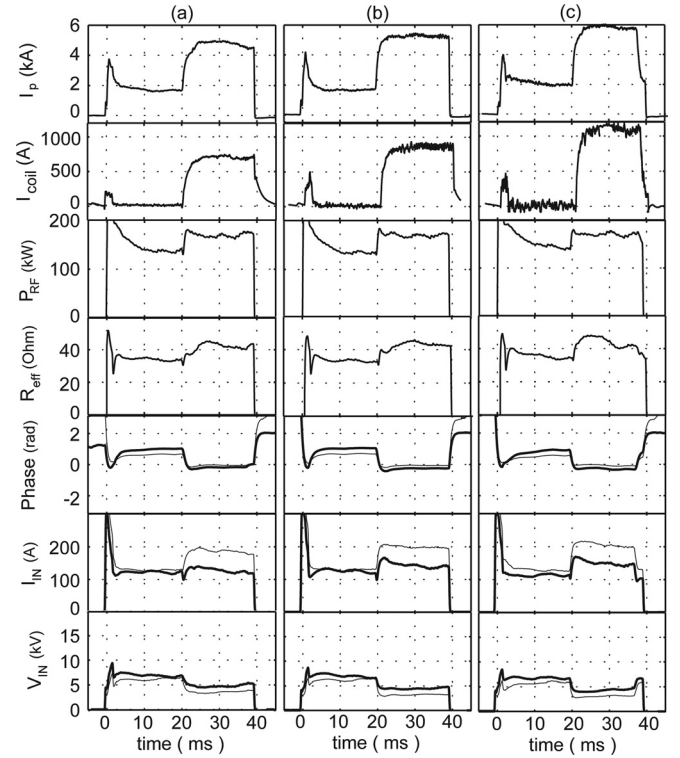


FIG. 2. Time evolution of rotamak-ST plasma current, total coil current, antenna input power, the effective impedance of the plasma, antenna phase difference, input current and voltage before and after applying (a) one middle coil, (b) one middle and two side coils (at $z = \pm 8$ cm), and (c) one middle and four side coils.

indicating an improved impedance matching between the rf generator and the antenna system. The difference among the three cases is that when more coils are used, larger coil current can be applied, and therefore larger plasma current can be driven. Fig. 2(c) shows the best result in the ST regime where plasma current is boosted by 200% from 2.0 kA to 6 kA when five coils are energized with a total current of 1.2 kA ($I_{\text{coil}} = i_{\text{feed}} \times n_{\text{turn}} = 0.050 \text{ kA} \times 24$). The input current of RMF antenna I_{in} and the effective impedance of the plasma R_{eff} are found to have the largest value. The plasma shape formed with five shaping coils is reconstructed and plotted in Fig. 1(b). Those results clearly indicate that the enhancement of plasma current drive is mainly due to the substantial extension of the plasma column uniformly along the z -axial direction. This would in turn improve impedance matching, and thus, larger antenna input power can drive more plasma current.

In the FRC experiments, plasma discharge is found to be less stable than in the ST regime. The tilt and radial shift modes are frequently detected by Mirnov coils even when I_{coil} is lower than the “disruptive” value. The best shot in each case is plotted in Fig. 3. The three cases have almost the same effective impedance of the plasma and antenna input power (160 kW). The difference among them is that when more coils are used, less coil current can be applied, and thus, less plasma current can be driven. This phenomenon is totally contrary to what is observed in the ST regime. Fig. 3(a) shows the best shot in the FRC regime where plasma current is boosted by 100% from 2.0 kA to 4.0 kA

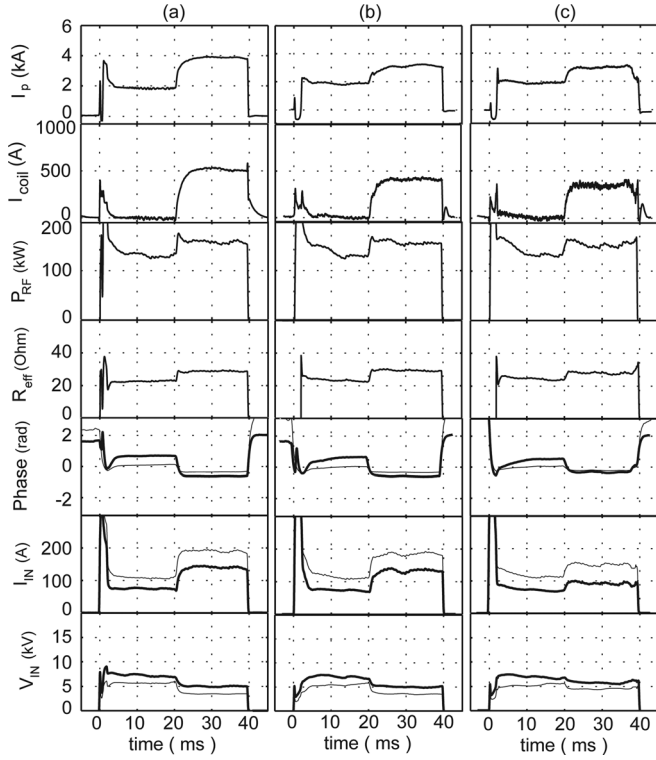


FIG. 3. Time evolution of rotamak-FRC plasma current, total coil current, antenna input power, the effective impedance of the plasma, antenna phase difference, input current and voltage before and after applying (a) one middle coil, (b) one middle and two side coils (at $z = \pm 8$ cm), and (c) one middle and four side coils.

when one middle coil is energized with a total current of 0.55 kA ($= I_{\text{coil}} = i_{\text{feed}} \times n_{\text{turn}} = 0.069 \text{ kA} \times 8$). The plasma shape formed with the middle coil is reconstructed and plotted in Fig. 1(c). This indented plasma has proven to be free from both the tilt and radial shift modes.⁹ This explains why larger plasma current can be driven with one middle coil in the FRC regime. However, if the I_{coil} is increased larger than 0.55 kA, which is the “disruptive” value for the FRC regime, the plasma discharge will be disrupted by the radial shift mode.

It has been found in our experiments that the enhancement of plasma current drive is partially due to the radial compression and axial extension of plasma column, during which the $L_s \times I_p$ flux is conserved (here L_s is the plasma self-inductance). In addition, an “extra” current is generated because as L_s is reduced, the impedance matching is significantly enhanced, and thus, the antenna input power is increased and more plasma current can be driven. Based on experimental measurements, we can estimate the geometrical parameters of the plasma shape before and after shaping coils are energized in the ST discharge. Before energizing the coils, at $t = 15$ ms, the magnetic axis position is $R_0 = 15.2$ cm, minor radius is $a = 7.2$ cm, and the ellipticity (the ratio of the separatrix half-length to the minor radius) $k = b/a = 2.0$. After the five shaping coils are applied, those parameters change to $R_0 = 13.2$ cm, $a = 6.6$ cm, and $k = 4.0$. Using the model equations for plasma self-inductance,¹⁰ we can estimate: $L_s(t = 15 \text{ ms}) = 0.085 \mu\text{H}$ and $L_s(t = 25 \text{ ms}) = 0.040 \mu\text{H}$, which drops by 53%. On the other

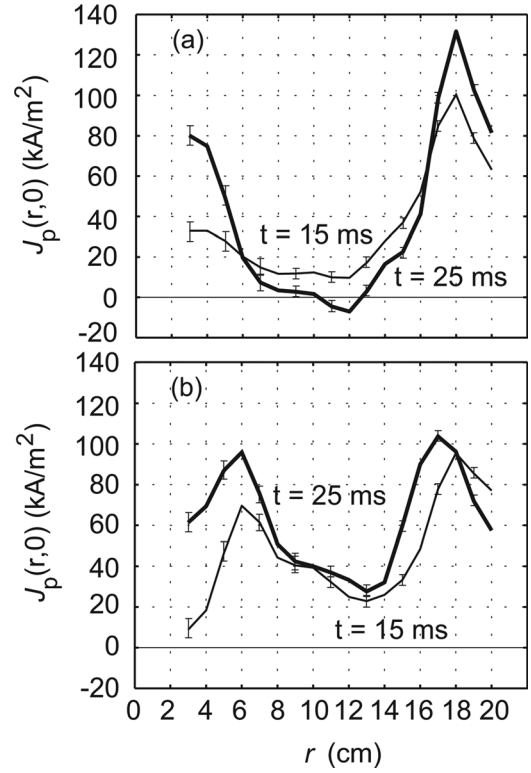


FIG. 4. Plasma current profiles along the $z = 0$ chord in (a) the FRC regime and (b) the ST regime before and after the middle coils are energized with 260 A current.

hand, the plasma current I_p is substantially boosted from 2.0 to 6.0 kA and the net increase in $L_s \times I_p$ flux is thus 40%. This is confirmed by a 25% increase of antenna input power, which grows from 140 to 175 kW, as seen in Fig. 2(c).

While in the FRC regime, after the middle coil is employed at 20 ms, the plasma surface becomes an indented FRC, stretching a little along the axial direction. Similarly, in the FRC regime, we have $R_0 = 16.3$ cm, $a = 6.6$ cm, and $k = 1.9$ at $t = 15$ ms, and $R_0 = 15.23$ cm, $a = 6.9$ cm, and $k = 2.5$ at $t = 25$ ms. The estimation shows that the plasma self-inductance L_s drops by 35% from 0.123 to 0.0795 μH . However, the plasma current grows from 2.0 to 4.0 kA, so that the $L_s \times I_p$ flux grows by 29%. This is confirmed by a 23% increase of antenna input power, which grows from 130 to 160 kW, as seen in Fig. 3(a).

To understand why the response of the plasma current to the shape change is drastically different in FRC and ST regimes, it is necessary to obtain the radial profiles of plasma current density before and after the shaping coils are energized. To ensure reliable comparison, one middle coil with the same current ($I_{\text{coil}} = 0.26$ kA) is used in both regimes. Based on the measured magnetic field, the profiles of plasma current density along $z = 0$ chord are plotted in Fig. 4. In both FRC and ST regimes, the profiles are hollow with two peaks located on both sides of the magnetic axis, forming the inner and outer current rings. Without the middle coil, in the FRC discharge, the peak current density of the outer current ring is roughly three times larger than the inner one, indicating a strong edge current drive property.

When the middle coil is applied in the FRC regime, as seen from Fig. 4(a), the plasma current density $J_p(r, 0)$ grows both near the chamber wall ($r < 17$ cm) and the central pipe ($r < 6$ cm), but meanwhile it drops in the central plasma region $r = 6\text{--}17$ cm, or even reverses in the plasma core at $10.2\text{ cm} < r < 12.5$ cm with the profile becoming more hollow. This property indicates that the action of the middle coil is to reshape plasma rather than to push it away from the chamber wall. Furthermore, it is interesting to note that with the increase of I_{coil} the position of peak current density stays the same, but the region of plasma current reversal widens quickly. Yet the mechanism of how the negative plasma current is generated in plasma core, and the drive source of radial shift mode with the increase of I_{coil} , is still not clear. This experiment result seems contrary to the theoretical simulation,¹¹ which predicts that smaller elongations $E \leq 0.7$ are favorable for the excitation of radial shift mode in FRC plasma. The radial shift mode is still observed in our rotamak-FRC experiments with plasma elongation $E > 1.5$.

While in the ST regime, as shown in Fig. 4(b), when the middle coil is energized with the same coil current as in the FRC regime, the plasma current is boosted in the whole plasma column at $r < 17$ cm, the outer peak plasma current density at $r \approx 18$ cm is moved away from the chamber wall by 1 cm. In addition, the current density is reduced at the edge region ($r > 18$ cm) near the chamber wall. The larger the I_{coil} is, the larger the distance the plasma column will be pushed. This indicates that the middle coil not only reshapes the plasma but also pushes it away from the chamber. This is consistent with the measured profile of electron pressure,⁷ which is increased by 50%–100% at plasma core while being decreased by 50% at the edge near the chamber wall; but electron temperature remains almost unchanged. In contrast to the FRC regime, there is no trend now for the plasma current reversal in the middle range value of r . However, if the coil current is larger than a critical value, the plasma column would be pushed too much inward and against the central pipe, resulting in the termination of discharge. This result suggests that an equilibrium system with active feedback control is needed for large scale rotamak devices.

In summary, the response of plasma current to the shape change of magnetic field is substantially different in the regimes with and without toroidal field. For a given rf generator power, in the ST regime, the plasma current can be boosted by 200% from 2.0 to 6.0 kA when five shaping coils in series connection are energized with a total current of 1.2 kA. The improvement of current drive efficiency is mainly attributed to the radial compression and the substantially axial extension of the plasma column; this in turn improves impedance matching and thus increases antenna input power. In the FRC regime, plasma current can be boosted by 100% when one middle coil is employed with a total current of 0.55 kA. It is observed that more side coils are used; less coil current can be applied. The radial profile of plasma current density shows a strong edge current drive property and the generation of negative current in the plasma core. The appearance of radial shift mode limits the achievable plasma current in FRC regime.

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Effect of toroidal magnetic field on $n = 1$ mode stability in rotamak plasmas

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To study the effect of toroidal magnetic field on $n = 1$ mode stability, a series of experiments with linearly ramping the axial current I_z , which makes field-reversed configuration (FRC) to spherical tokamak (ST) transition, have been conducted in rotamak. Results clearly demonstrate that the tilt mode can be completely suppressed by small I_z around 0.4 kA (in comparison with 2.0 kA plasma current). An unknown new mode with larger magnetic perturbations is triggered when I_z reaches 0.5 kA. This instability mode keeps saturation while plasma current is boosted when I_z is in the range of 0.6–1.4 kA. When I_z exceeds 1.6 kA, the new mode suddenly disappears and discharge is free from instability modes. © 2011 American Institute of Physics. [doi:10.1063/1.3622658]

One important modification in rotamak development is the addition of a steady toroidal magnetic field (TF) produced by an axial current I_z .^{1,2} With the appearance of external TF, the rotamak is able to operate as a spherical tokamak (ST) with an ultralow aspect ratio $A = R/a \sim 1.1$ (R and a are the major and minor radius). The earlier rotamak-ST experiments were focused on exploring the operational parameters which were achievable for high performance, and results have demonstrated that the rotamak-ST regime is more favorable for the penetration of rotating magnetic fields (RMFs) than the rotamak-field-reversed configuration (FRC) (without TF, $I_z = 0$) due to the excitation of rotating whistler wave modes.^{3,4} Later, experiments in the rotamak-ST regime further confirmed that the addition of TF can result in an essential improvement in the magnitude of the RMF-driven plasma current, in comparison with the rotamak-FRC regime for the same rf power level.^{5–7}

One of the advantageous features of the rotamak-ST is its improved stability. However, over a few decades of rotamak research, there was no report about the effects of TF on global plasma stability. Furthermore, the magnetohydrodynamic (MHD) stability of the rotamak-ST with an ultralow aspect ratio is expected to be significantly different from those of conventional tokamaks with large aspect ratios, but investigation on this topic is lacking as well. These issues are extremely crucial when the rotamak is extrapolated to a fusion reactor. With the recent development of new diagnostics in the Prairie View (PV) rotamak, the stability of RMF-driven plasma to low- n (n is the toroidal number) MHD modes were extensively studied in the rotamak-FRC regime. The $n = 1$ tilt and radial shift modes have been clearly observed in the experiments, and the mode stability is found to strongly relate to the shape of the magnetic field.⁸ Stimulated by these new findings, a new experiment campaign was launched to investigate two important issues: (1) Since the external TF can provide a strong stabilizing effect on tilt mode, at what critical value of I_z can the tilt mode be completely suppressed? (2) Is there any other low- n MHD modes existing in rotamak-ST discharges with different values of TF?

In this letter, we report the experiments of ramping axial current I_z over a 20 ms period in 40 ms rotamak discharge. The results indicate that rotamak-ST discharge is remarkably

stable when the ratio of axial current to plasma current I_z/I_p is larger than 70%. In addition, the experiments provide an effective way to obtain the optimal operation condition for rotamak-ST discharges.

The experiments are conducted at cylindrical chamber rotamak that has been illustrated schematically in Ref. 8. The steady-state confinement field is produced by two “vertical” field coils. In the experiments, the vacuum vertical field at chamber center is 25 G. In the rotamak-ST regime, the external TF is added by feeding an axial current I_z through a set of six wires inside the central stainless-steel pipe (od = 4.0 cm) covered by a quartz tube; the pipe is positioned along the major axis of the device. The I_z is generated by discharging a capacitor bank (maximum 152 kJ), which can ramp the I_z up to the maximum value of 18 kA. The RMF is produced by two pairs of coils, which are connected through a system of matching/tuning networks to two 500 kHz/400 kW rf generators with a 90° phase shift between them. The oscillating magnetic field (below 50 kHz) is measured with two sets of Mirnov magnetic coils located at $Z = -4$ cm (left side) and $Z = +4$ cm (right side). Each set consists of eight B_R -oriented coils mounted around the chamber surface at equal 45° toroidal angle intervals. The boundary poloidal flux is measured with six flux loops wound over the chamber surface at $z = \pm 3.5$ cm, ± 7.5 cm, and ± 22 cm. Hydrogen gas is continuously fed through the chamber at $p_f = 1.1$ –1.6 mTorr. The plasma current is measured with a Rogowski coil, which is pulled through the central steel pipe and looped around the poloidal cross-section. The electron temperature and density are measured by a double floating Langmuir probe. Typically, $T_e = 15$ –25 eV and $n_e = 1.2$ – 2.0×10^{12} cm⁻³.

Experiments were typically conducted in 40 ms rotamak-FRC discharge with $I_p \approx 2$ kA, with the axial current I_z ramping linearly from 20 ms to 40 ms during a shot. There were two series of ramping experiments: one series with small I_z from 0 to 1.4 kA, and another series with large I_z from 0 to 6 kA. Fig. 1 shows the measurements with the small I_z ramp. It is seen that the tilt mode appears at $t = 10$ ms in the discharge before I_z starts to ramp at $t = 20$ ms; the amplitude and onset time of the tilt mode depend on filling gas pressure and the vertical field.⁸ Unlike the radial shift

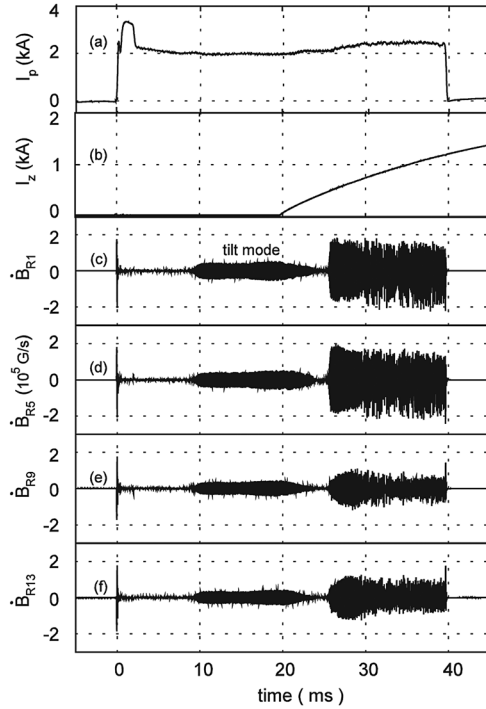


FIG. 1. Time evolution of plasma current, axial current, MHD oscillation signals of a left set coil #1 and #5, and a right set coil #9 and #13. Two coils on the same set are at 180° toroidal angle interval. I_z ramps almost linearly to 1.4 kA, $p_f = 1.1$ mTorr.

mode, which is the driver of plasma disruption in rotamak discharge, the tilt mode is non-destructive but can result in plasma current drop by 10%. Typically the tilt mode has an oscillation frequency in the range of 15–19 kHz and perturbation amplitude of 1 Gauss which is around 4% of the equilibrium field. Fig. 1 clearly shows the strong stabilizing effect of I_z on the tilt mode. The amplitude of the tilt mode is gradually reduced with the increase of I_z , and the tilt mode is completely suppressed when I_z reaches 0.4 kA at $t = 24$ ms. This value is about 20% of the plasma current. The result agrees well with the finding on TS-3,⁹ in which the tilt mode was completely suppressed when a small current in the center conductor of the device was added. We use the following equation originated from Ref. 9 to estimate the threshold value of I_z for suppressing the tilt mode:

$$\frac{I_z}{I_p} \geq \left(\frac{3}{2} \pi (-n^*) C_L A \right)^{1/2} \frac{A - 1}{k^{3/2}}. \quad (1)$$

Here $n^* = -(R/B_z)(\partial B_z/\partial R)$ is the field index at the magnetic axis, C_L is a coefficient (~ 1) that determines the self-inductance for plasma with a cross-section of half height b and minor radius a , and $k = b/a$ is the plasma elongation. In our case, $n^* \sim 0.2$, $A = 15.0/13.25 \approx 1.1$, and $k \approx 1.0$. Estimation from Eq. (1) gives $I_z/I_p \approx 20\%$, which is in excellent agreement with our experimental observation. This result indicates that in a rotamak-ST regime, the TF has a strong stabilizing effect on the tilt mode; small I_z can effectively improve its global tilt stability.

The most interesting result seen in Fig. 1 is that when I_z ramps above 0.5 kA ($I_z/I_p = 25\%$), a new rotating mode

appears soon after the suppression of the tilt mode. This mode grows very fast at $t = 24.5$ ms and saturates within 1 ms. Its amplitude remains maximum when I_z is in the range of 0.6–1.2 kA ($B_t = 5$ –10 G). In contrast to the tilt mode which usually leads to plasma current drop by 10%, this new mode boosts plasma current by 25% during its saturation period. In addition, the amplitude of magnetic perturbation of the new mode is different on two sides of the mid-plane: the magnetic perturbation signals of the left side Mirnov coil set ($Z = -4$ cm) (Figs. 1(c) and 1(d)) increase faster than that of the right side set ($Z = +4$ cm) (Figs. 1(e) and 1(f)); when this mode saturates, the signals' amplitudes of the left side set coils become as twice as that of the right side set coils. It is the first time to observe this type of mode in rotamak plasma.

Fig. 2 compares the tilt mode with the new mode. For tilt mode, the signals from two sets of Mirnov coils are almost identical; the tilt mode oscillates at a frequency around 16 kHz with about 45° phase shift between the signals from two sets of coils (Fig. 2(a)). The non-zero phase shift of the tilt mode in rotamak plasma is attributed to the observed multi-mode structure; the tilt mode is the dominant one among them. There is no pure $n = 1$ tilt or radial shift mode being observed in rotamak plasma. The new mode oscillates at a frequency near 14 kHz, lower than the tilt mode but higher than the radial shift mode (near 10 kHz); the phase shift is about 40° , and signal amplitudes of the left set coils are doubled in comparison with that of the right set coils (Fig. 2(b)). The measurements of boundary poloidal magnetic flux indicate that the axial profile of flux is symmetrical along the mid-plane before and after I_z ramping. This measurement excludes the possibility of the plasma column moving along the z -axis.

Additional I_z ramping experiments were carried out to clarify the new mode structure. To exclude the influence of the tilt mode in the discharge, the filling gas pressure p_f is increased to 1.6 mTorr to prevent the appearance of the tilt mode. Meanwhile, I_z starts to ramp from the beginning of discharge instead of at $t = 20$ ms, as shown in Fig. 3. The discharge is free from an instability mode when I_z is below 0.4 kA. A rotating mode with a low frequency (around 14 kHz) magnetic oscillation develops at $t = 6.5$ ms when I_z is above 0.4 kA; it saturates at $t = 8$ ms when I_z is larger than 0.5 kA, and maintains saturation until the end of the discharge where

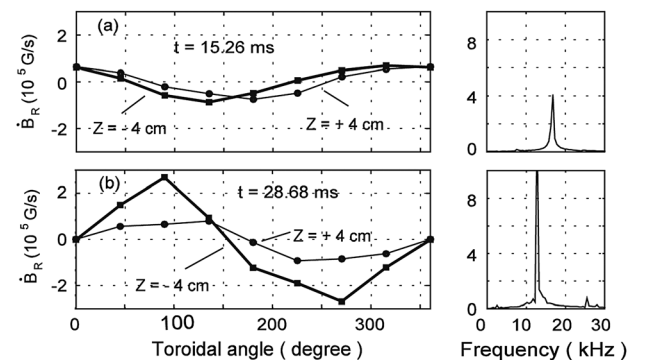


FIG. 2. Signals from two sets of Mirnov coils and their spectra before and after the ramp of B_t , $p_f = 1.1$ mTorr.

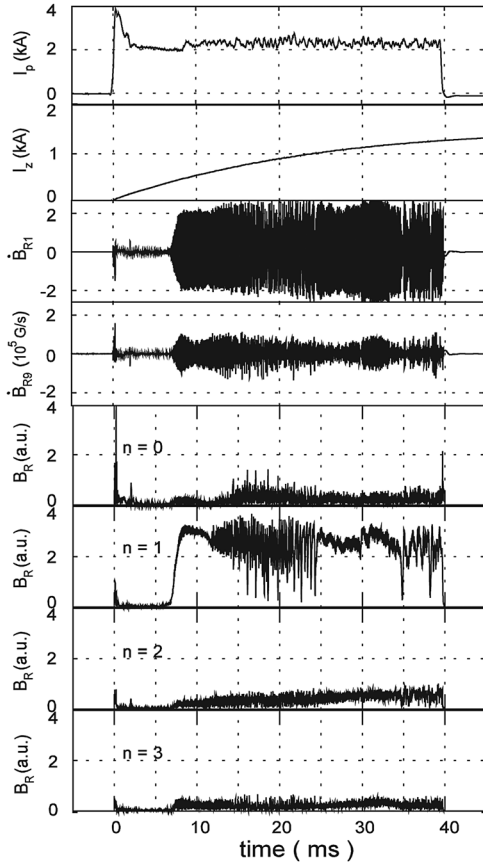


FIG. 3. Time evolution of plasma current, axial current, MHD oscillation signals from a left coil #1 and a right coil #9, magnitude of $n = 0-3$ perturbations to B_R . I_z ramps to 1.4 kA, $p_f = 1.6$ mTorr.

I_z approaches 1.2 kA. The signal amplitudes of the left set coils are even more than double the right set coils (only one coil signal from each set is shown). Furthermore, Fig. 3 clearly demonstrates that the $n = 1$ mode is the dominant perturbation. These results basically repeat what was shown in Fig. 1.

At present, it is not clear whether the observed new mode is a tilt-like mode or a kink-like one, or perhaps other unknown modes. In a plasma system that is symmetric about the middle plane, the ideal tilt and kink mode should have equal amplitudes of magnetic perturbation symmetrically about the middle plane; the ideal tilt mode has 0° phase shift while the ideal kink mode has a moderate phase shift. The phase shift associated with the new mode might be attributed to the multi-mode structure or a real phase shift related to the kink mode structure. However, it is not understandable why there is a factor of 2 difference between the amplitudes of magnetic perturbations measured by Mirnov coils. Moreover, it is difficult to explain why plasma current is boosted by 25% when the new mode saturates in the discharge. For this, further experimental and theoretical studies are still needed.

Another series of experiments was conducted with large I_z ramping and the tilt mode existing in the discharge at $t = 10$ ms before I_z ramps at $t = 20$ ms. As can be seen in Fig. 4, in the region where the value of I_z is less than 1.5 kA, the effects of TF on the tilt mode (suppression) and the new

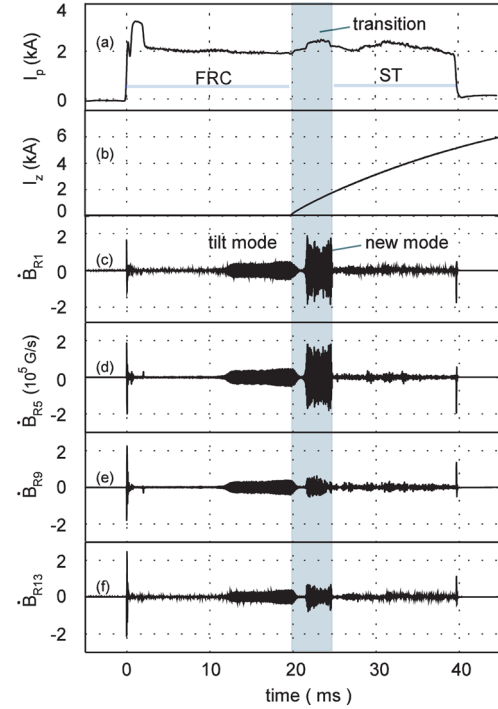


FIG. 4. (Color online) Time evolution of plasma current, axial current, MHD oscillation signals of a left set coil #1 and #5, and a right set coil #9 and #13. Two coils on same set are at 180° toroidal angle interval. I_z ramps to 6.0 kA, $p_f = 1.1$ mTorr.

mode (trigger) are exactly the same as demonstrated in Fig. 1 and Fig. 3. Since I_z ramps fast in this type of experiments, the tilt mode is suppressed quickly and the new mode saturates immediately after being triggered in the discharge. One interesting result shown in Fig. 4 is that when I_z approaches 1.6 kA, the new mode suddenly disappears; the corresponding ratio of I_z/I_p is around 70%. The discharge is free from any instability mode when I_z is larger than 1.6 kA. Repeated shots confirm that $I_z \approx 1.6$ kA is the threshold value to suppress the new mode, presumably due to the increase of magnetic shear by continuously increasing TF. Since the rotamak-ST regime typically operates under the condition of $I_z > 3.0$ kA and the ratio of $I_z/I_p > 150\%$, this result explains why no MHD mode was observed in previous rotamak-ST experiments.⁵

We conclude from Fig. 4 that the appearance of the new mode indicates the start of a transition from FRC to ST regime; strong magnetic perturbations and the increase of plasma current have been observed during the transition. The disappearance of the new mode can be regarded as the establishment of the rotamak-ST regime, no instability mode has been observed in this regime. This result clearly demonstrates the strong stabilizing effects of the toroidal field on the $n = 1$ MHD modes in rotamak plasma.

Our newly developed real time power measurement system allows us both to trace the total antenna input power and to calculate the effective impedance of the plasma. This helps us to understand how the RMF antennae and the plasma respond to the transition from rotamak-FRC to rotamak-ST regime. It is clearly seen from Fig. 5 that during the transition, there is a sudden change of the antenna phase-

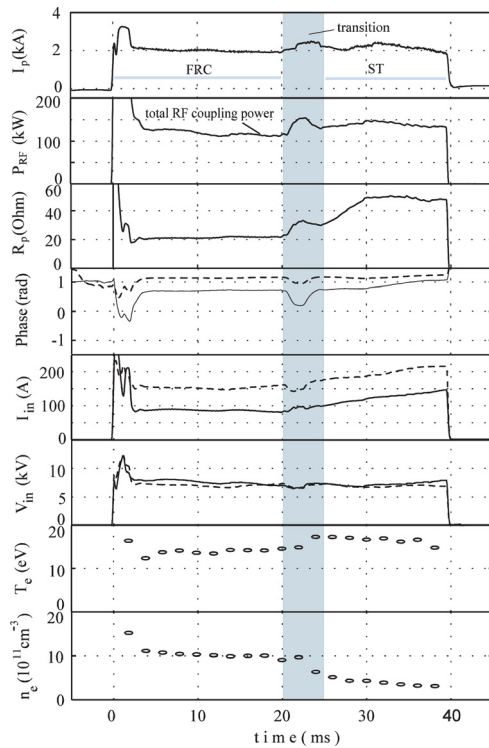


FIG. 5. (Color online) Time evolution of measured plasma current, effective impedance of plasma R_p , the antenna input power, phase difference, input current and input voltage, and electron temperature T_e , and density n_e at $r = 15$ cm. Experiments are in the case of large I_z ramping.

difference (θ) between the antenna input current and the input voltage. The reduce of the phase difference leads to the increase of antenna input power by 30% due to its $\cos \theta$ relationship with θ , indicating the improvement of the impedance matching between the rf generator and the antenna; therefore, more plasma current can be driven. It is also noted in Fig. 5 that near the outer edge area the electron temperature increases while its density decreases during the transition, indicating the enhancement of plasma confinement. After the transition, the response of plasma to the increase of I_z is similar to our previous experiments^{7,8} that there exists an optimal value of axial current for the largest plasma current, and this optimal I_z is different for each value of B_v . The I_z ramping experiment shown in Fig. 5 provides an effective

method to quickly find the optimal value of I_z for the rotamak-ST operation.

In summary, the I_z ramping experiments have been conducted to study the effects of external toroidal field on $n = 1$ instability modes in the rotamak-ST regime, which has an ultralow aspect ratio ($A = 1.1$). Because of this extreme limit, the toroidal field has strong stabilizing effects on global instability modes. It is clearly observed in experiments that the tilt mode is completely suppressed by small I_z (0.4 kA, $I_z/I_p \approx 0.2$). An unknown new mode with a strongly asymmetric perturbation structure appears and saturates in the discharge when I_z is in the range of 0.55–1.4 kA ($I_z/I_p \approx 0.26$ –0.6). It is noted that the appearance of the new mode triggers a strong interaction between the rf antennae and the plasma, leading to a sudden antenna phase change and the increase of rf power dissipated in the plasma, thus driving larger plasma current. The new mode suddenly disappears when I_z is larger than a threshold value around 1.6 kA ($I_z/I_p \approx 0.7$), which presumably produces the magnetic shear strong enough for mode suppression. Above this critical value, rotamak-ST configuration is established, and discharge is free from instability modes.

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Experiments on rotamak plasma equilibrium and shape control

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A set of magnetic shaping coils and copper rings is installed in cylindrical chamber rotamak to allow for an active equilibrium control in 40 ms plasma discharges. The coils, which are powered by programmable current source, are used to control both the plasma shape and the boundary poloidal magnetic flux. Without the active equilibrium control, the boundary flux drops from its vacuum value of 0.3 mWb to zero after the plasma current is generated. If the coils are activated, the boundary magnetic flux can be sustained within the 0.2–0.3 mWb range, thus keeping the separatrix away from chamber wall during whole period of the shot. The passive copper rings help in eliminating the fast variations of the boundary magnetic flux. The response of rotamak plasma to the active equilibrium control is drastically different in regimes with or without external toroidal field. A model is presented that describes the change in plasma shape, plasma current, and pressure under the effect of active equilibrium coils. © 2010 American Institute of Physics.

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I. INTRODUCTION

In rotamak, a steady-state plasma current is driven in toroidal direction by an externally applied rotating magnetic field (RMF) that results in reversal of the confining magnetic field on the symmetry axis.¹ The shape of the plasma is largely determined by the geometry of confinement coils. If the equilibrium system consists of two coils, a nearly spherical compact toroidal plasma is formed.^{2–4} On the other hand, if a set of identical confinement coils is used to produce a uniform axial field, then an elongated field-reversed configuration (FRC) is produced.^{5,6}

As the plasma current loop is formed, it expands and contacts chamber wall. In short-pulse devices, the sustainment is improved by using flux-conserving rings made of copper or aluminum, placed outside or inside chamber.^{5,6} The plasma boundary is kept away from the wall by field of induced image currents in the rings. However, the magnetic field still diffuses through the rings in a few milliseconds.

To address this issue, an alternative approach has been studied at the Prairie View (PV) rotamak. A hybrid active/passive equilibrium control system was recently installed in the cylindrical rotamak. The passive part consists of four copper rings that serve as a partial flux conserver. The active part consists of three coils wound over the chamber surface and connected to a programmable current source. The main purpose of the active coils is to keep the plasma separatrix off the chamber wall, similar to the use of flux-conserving rings in the University of Washington TCS experiment,⁵ and also to control the plasma shape. By programming the current in active coils, not only the elongation of plasma can be changed but also its triangularity. Although such equilibrium systems have been used in tokamaks since 1970s, their implementation in RMF-driven plasmas might be more difficult. First, the magnitude of plasma current density is determined by the separatrix radius, which depends on external magnetic field so that the confining force $\mathbf{J} \times \mathbf{B}$ is a nonlinear function of the external field $\mathbf{B}$. Second, reshaping the

plasma can trigger global instabilities, especially when there is no significant toroidal field in plasma.

We examined the effect of the active equilibrium system on rotamak plasmas in two sets of experiments: FRC regime, which has no external toroidal field, and spherical torus (ST) regime, which has an added toroidal magnetic field. As we will show, the presence of toroidal field makes discharges very stable, allowing the equilibrium system to keep the separatrix away from chamber wall throughout the 40 ms discharge, and substantially increases the plasma elongation. In rotamak-FRC regime, on the other hand, the increase in current in the active equilibrium system above certain limit results in excitation of $n=1$ tilt and radial shift instabilities and disruption of the discharge.

This paper is organized as follows. Parameters of rotamak, active equilibrium control coils, and diagnostic tools are described in Sec. II. The results for rotamak-ST regime are discussed in Sec. III, and results for rotamak-FRC regime in Sec. IV. A summary is given in Sec. V.

II. EXPERIMENTAL SETUP

The sketch of the PV rotamak is shown in Fig. 1. The background vacuum in the 80 cm long 40 cm id Pyrex chamber is maintained at 5.0×10^{-7} Torr. In the experiments discussed here the working gas is hydrogen, which is continuously fed through the chamber at $p_f \approx 1$ mTorr.

The RMF is produced by two pairs of coils, which are connected through a system of matching/tuning network to two 500 kHz/400 kW rf generators with a 90° phase shift between them. In a typical experiment, the magnitudes of the radial and toroidal RMF components are $\tilde{B}_r \approx 5\text{--}10$ G and $\tilde{B}_\theta \approx 10\text{--}20$ G.

The steady-state confinement field is produced by two coils (“vertical” field coils). In these series of experiments, the vacuum field at the chamber center is 25 G.

A recent modification of the system includes addition of

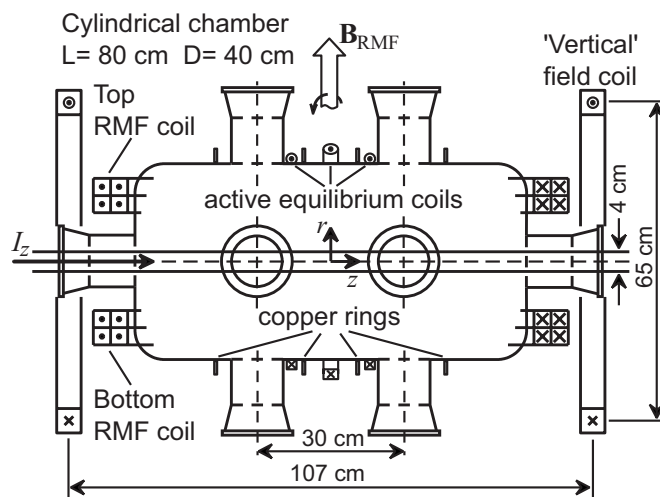


FIG. 1. The schematic view of PV rotamak.

three active magnetic coils that can be energized during the 40 ms plasma shot. The active coils are wound close to the chamber

surface and connected to a programmable current source with current rise time of 1–2 ms. One “middle” coil (eight turns) is located at $z=0$, two “side” coils (four turns each) at $z=\pm 8$ cm. Another addition to the equilibrium system is a set of four copper rings, 27.5 mm wide/6 mm thick, placed at $z=\pm 5.5$ cm and $z=\pm 23.5$ cm just outside the chamber surface.

In rotamak-ST regime, the external toroidal magnetic field is added by feeding a current I_z through the central stainless-steel pipe; the latter is covered by a 45 mm outside diameter quartz tube. The boundary poloidal flux is measured with six flux loops wound over the chamber surface at $z=\pm 3.5$, ± 7.5 , and ± 22 cm. The oscillating magnetic field (below 50 kHz) is measured with two sets of Mirnov magnetic coils located at $z=-4$ and 4 cm. Each set consists of eight B_r -oriented coils mounted around chamber surface at equal 45° toroidal angle interval. The magnetic field in plasma is scanned with internal magnetic probe.³

The plasma current is measured with the Rogowski coil, which is pulled through the central steel pipe and looped around the poloidal cross section. The current of the active coils that also enters the area enclosed by the Rogowski coil is subtracted by measuring the feeding current to the active coils. The unaccounted current in the copper rings is opposite

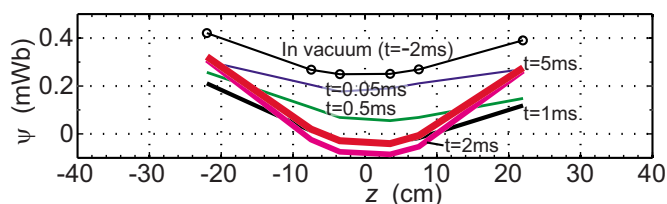


FIG. 2. (Color online) Change in the boundary poloidal flux in rotamak-ST discharge without active coils and copper rings. The position of flux loops is indicated by circles. The flux in vacuum is produced by the vertical magnetic field only.

to the plasma current, resulting in underestimation of plasma current during first 3–5 ms of discharge.

The plasma temperature and density are measured with double floating Langmuir probes that can scan almost the entire cross section of plasma. Typically, $T_e=15\text{--}25$ eV (uniform radial profile) and $n_{e,\text{peak}}=(1.2\text{--}2.0)\times 10^{18}$ m⁻³ (peaked profile).

III. EQUILIBRIUM CONTROL IN ROTAMAK-ST REGIME

In experiments discussed in this section, the rotamak-ST configuration is produced by adding the axial current $I_z=2.5$ kA. As seen in the earlier experiments,^{2,7,8} adding toroidal magnetic field allows doubling or tripling the RMF-driven plasma current due to whistler excitation.^{8,9} The highest efficiency is achieved when the axial I_z current is 60%–80% of the magnitude of plasma current.^{9,4} For the present chamber, the highest plasma current $I_p\approx 5$ kA is achieved when the vertical field is 45 G and $I_z\approx 3$ kA, while for the spherical chamber the largest current $I_p\approx 5$ kA was generated at $B_v=45$ G and $I_z\approx 5$ kA.⁴ For comparison, in Flinders rotamak-ST the plasma current $I_p=6.5$ kA was produced at $B_v=60$ G and $I_z=9.65$ kA.¹⁰

In present series of experiments, the vertical field was set to 25 G in order to maintain same conditions as in the rotamak-FRC regime. For this value of vertical field, the plasma current is $I_p\approx 2.2$ kA in both rotamak-ST and rotamak-FRC regimes (without active equilibrium control).

A. Experimental results

Without active coils and copper rings, the temporal change in the boundary poloidal flux is shown in Fig. 2. The boundary flux drops sharply by 60%–80% during first 0.1–0.2 ms of discharge, then continues to decline during the next 1–2 ms. For the rest of 38 ms the boundary flux remains at approximately zero level for $|z|<7$ cm, indicating that the separatrix is resting on chamber wall.

Based on measured boundary flux profile and total plasma current, the plasma shape is reconstructed, as displayed in Fig. 3. The plasma current is represented by ten current loops; the radial and axial positions of the loops are adjusted until the boundary flux profile matches that measured in experiment. Previously, a direct scan of magnetic field with internal magnetic probes has shown a similar, nearly spherical shape of flux surfaces.³ The oblate shape of plasma can be attributed to the high mirror ratio of the vertical field and ponderomotive force due to nonuniform RMF that pushes plasma toward the midplane.¹¹

In contrast, when three active coils (connected in series) are energized with total current $I_c\approx 1$ kA (feeding current is 65 A), the boundary flux is sustained at 0.2 mWb level during most of the discharge duration, as demonstrated in Fig. 4. In the shot, the active coils are energized 0.5 ms before starting RMF in order to reduce initial drop in boundary flux; the response time of the current source is limited by inductance of active coils. The copper rings that could eliminate fast variations of poloidal flux were disabled (cut open) in this shot.

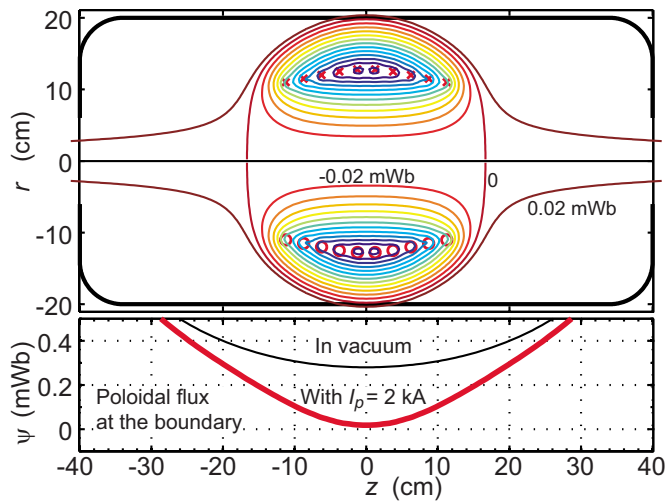


FIG. 3. (Color online) Model for poloidal flux surfaces in rotamak-ST regime without active coils.

The profile of the boundary flux along z axis becomes almost uniform after 2–3 ms, as seen in Fig. 5. Based on the data from six flux loops and measured plasma current, the reconstructed plasma shape is shown in Fig. 6. The plasma

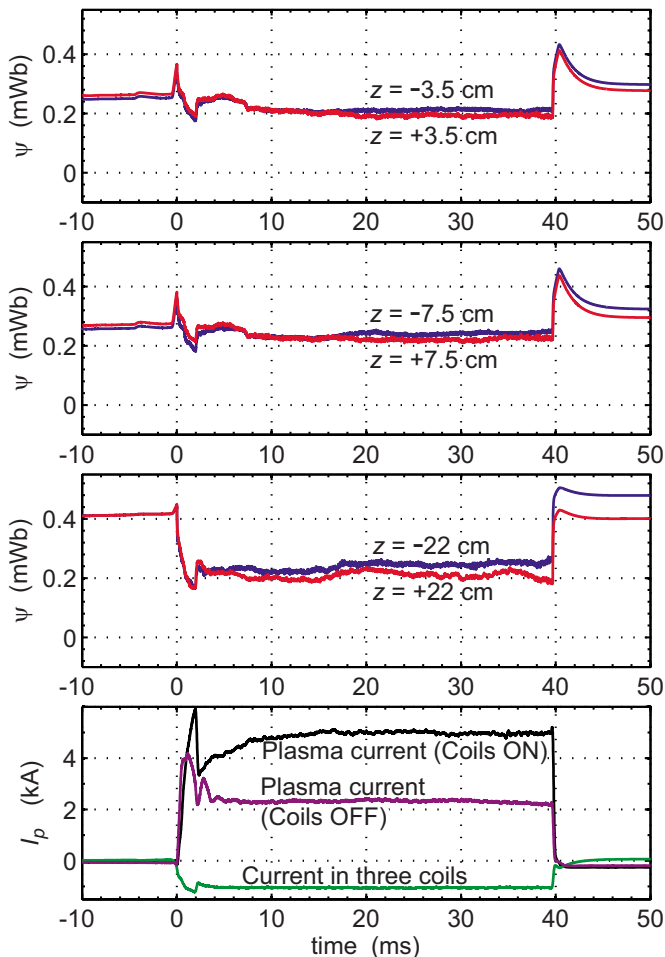


FIG. 4. (Color online) Measured boundary poloidal flux at different z locations and plasma current during rotamak-ST discharge. Three active coils are energized from $t = -0.5$ ms to $t = 40$ ms. For comparison, plasma current in discharge without active coils is also shown.

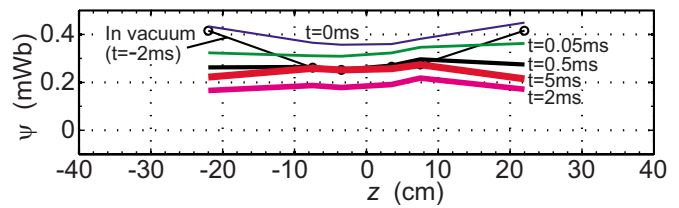


FIG. 5. (Color online) Change in the boundary poloidal flux in rotamak-ST discharge with active coils.

becomes more prolate, and the separatrix is well separated from chamber wall.

Further improvement is achieved with adding four copper rings. Although the magnetic field still diffuses through the rings in 2–3 ms, the presence of the rings helps to reduce the initial sharp drop of the flux to only 5%–10%, which is then replenished by the active coils. This is demonstrated in Fig. 7 for a shot in which three active coils are energized with $I_c \approx 1$ kA. It is seen that the initial jump is much smaller than that shown in Fig. 4.

Langmuir probe scans show that with the active coils, the plasma density becomes higher by about 50% at $r < 13$ cm; the electron temperature jumps from 20 to 25 eV at $r > 12$ cm, and from 19 to 21 eV at smaller radii. The resulting change for the radial profile of electron pressure at $z = 0$ plane is shown in Fig. 8.

For the current in active coils below or about 1 kA, the plasma remains very stable. For $I_c \approx 1.2$ kA, the plasma is pushed too much against the center column, resulting in a brief drop of I_p to a 3–3.5 kA level, usually followed by recovery to 4–4.5 kA. The boundary flux in such case is about 0.35 mWb at $|z| < 10$ cm and 0.25 mWb at $|z| \approx 22$ cm, indicating the formation of doublet-shape plasma.

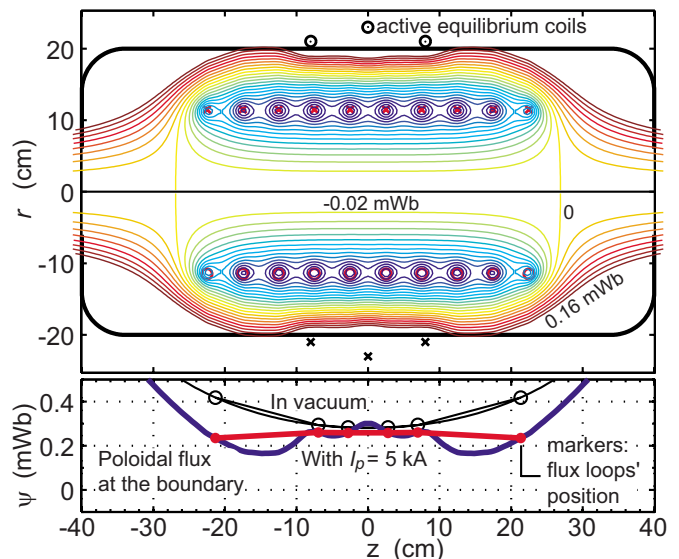


FIG. 6. (Color online) Model for poloidal flux surfaces in rotamak-ST regime with active coils.

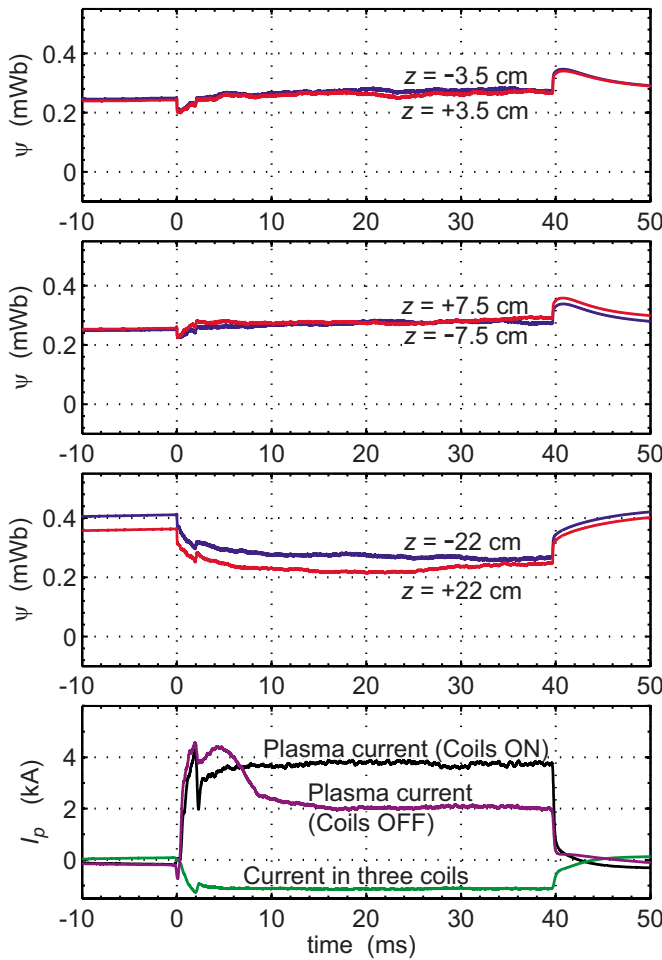


FIG. 7. (Color online) Boundary poloidal flux and plasma current in rotamak-ST with copper rings and active coils. Three active coils are energized during $t=0-40$ ms. For comparison, plasma current in discharge without active coils is also shown.

B. Theoretical analysis

The observed response of RMF-FRC plasma to external equilibrium field can be explained using the pressure balance condition in (r, z) plane,

$$\nabla p = \mathbf{J} \times \mathbf{B}, \quad (1)$$

which leads to the Grad-Shafranov equation. In cylindrical coordinates, it can be written as

$$r \frac{\partial}{\partial r} \left(\frac{1}{r} \frac{\partial \psi}{\partial r} \right) + \frac{\partial^2 \psi}{\partial z^2} + (2\pi)^2 \left(\mu_0 r^2 \frac{\partial p}{\partial \psi} + r B_\theta \frac{\partial (r B_\theta)}{\partial \psi} \right) = 0, \quad (2)$$

where $\psi = 2\pi \int_0^r B_z r dr$. The first two terms in Eq. (2) correspond to $(-2\pi r \mu_0 J_\theta)$, where J_θ is the toroidal plasma current density; in our case, it is driven by RMF. The term with toroidal field B_θ can be ignored because in our conditions, $B_\theta \sim 1/r$. Strictly speaking, in weakly ionized plasma with unmagnetized ions, the force balance equation can only be written for electron component. By adopting Eq. (1) for our conditions (with p being the electron pressure) we ignore currents and electric fields in (r, z) plane. We also ignore the

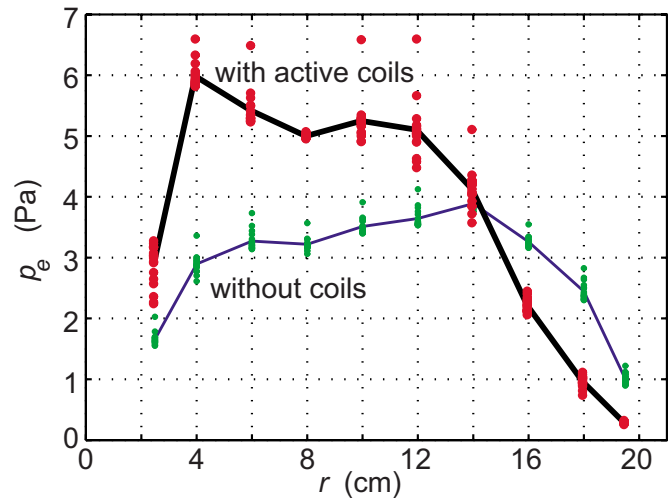


FIG. 8. (Color online) Comparison of electron pressure profiles along $z=0$ plane in discharges with and without active equilibrium control.

time-average $\langle (-J_{z, \text{rmf}} B_{\theta, \text{rmf}}) \rangle$ term which could be contributing to the radial confining force.²

Further, we consider the Solov'ev model,¹² which assumes that the plasma pressure is a linear function of the poloidal magnetic flux ψ . The solution can be written in the following form:

$$\psi(r, z) = \psi_0 \left(2 \frac{r^2}{r_0^2} - \frac{r^4}{r_0^4} - \frac{z^2 r^2}{E^2 r_0^4} \right), \quad (3)$$

$$p(\psi) = p_0 + (\psi - \psi_0) \frac{2\psi_0}{\mu_0 \pi^2 r_0^4} \left(1 + \frac{1}{4E^2} \right). \quad (4)$$

Here, ψ_0 and p_0 are the values at the magnetic axis ($r=r_0, z=0$) and elongation $E \equiv z_s/r_s$ is the ratio of the separatrix half length to its radial extension. Magnetic field and current density are found from Eq. (3),

$$\begin{aligned} \mathbf{B}(r, z) &= \frac{\nabla \psi \times \nabla \theta}{2\pi} + r B_\theta \nabla \theta \\ &= \frac{2\psi_0}{\pi r_0^2} \left\{ \left(1 - \frac{r^2}{r_0^2} - \frac{z^2}{2E^2 r_0^2} \right) \hat{\mathbf{e}}_z + \frac{zr}{2E^2 r_0^2} \hat{\mathbf{e}}_r \right\} + B_\theta \hat{\mathbf{e}}_\theta, \end{aligned} \quad (5)$$

$$J_\theta(r) = \frac{4\psi_0 r}{\pi \mu_0 r_0^4} \left(1 + \frac{1}{4E^2} \right). \quad (6)$$

Equations (2)–(4) only apply inside the separatrix. Current density is assumed to be zero outside the separatrix. From Eq. (3), one can find $r_s = \sqrt{2}r_0$, and from Eq. (4),

$$\psi_0 = -\pi r_0^2 \left\{ \frac{\mu_0 (p_0 - p_s)}{2[1 + 1/(4E^2)]} \right\}^{1/2}, \quad (7)$$

where $p_s \equiv p(y_s)$ is plasma pressure at the separatrix. It is seen that the solution depends on the pressure difference only, $\Delta p \equiv p_0 - p_s$; the value of p_s is irrelevant. Also, from Eq. (5),

TABLE I. Parameters calculated from the model and measured in experiment without active coils.

Parameter				Model	Experiment
Input for the model	Field at separatrix	$B_{z,s}$	(G)	27.0	25–30
	Plasma pressure	Δp	(Pa)	3.80	3.5–4.0
	Current density on axis	J_0	(kA/m ²)	−40.0	−30 to −60
	Separatrix radius	r_s	(cm)	19.90	≈20
	Magnetic axis radius	r_0	(cm)	14.04	≈15
	Plasma elongation	E		0.898	0.85–0.95
	Poloidal flux on axis	ψ_0	(mWb)	−0.084	−0.13 to −0.15
	Plasma volume	V	(m ³)	0.0297	
	Plasma current	I_p	(kA)	1.34	2.1–2.3
	Input power	P_{rf}	(kW)	95	95–105

$$\psi_0 = -B_z(r_s, 0) \pi r_0^2 / 2 = -B_z(r_s, 0) \pi r_s^2 / 4, \quad (8)$$

where the value of $B_z(r_s, 0)$ corresponds to the point $z=0$, $r=r_s$. A specific equilibrium is defined by three independent input parameters, for example, r_s , E , and ψ_0 . Physically, the equilibrium state is established by magnitude of external magnetic field, by rf power, which affects the value of driven plasma current, and by global power balance, which determines the peak density and temperature in plasma. Therefore, for input parameters we choose the value of magnetic field at the separatrix, $B_{z,s} \equiv B_z(r_s, 0)$, the current density on magnetic axis, $J_0 \equiv J_\theta(r_0)$, and the pressure difference $\Delta p \equiv p_0 - p_s$. Then, from Eqs. (6)–(8), the geometry of plasma is determined by

$$r_s = -4\sqrt{2} \frac{\Delta p}{B_{z,s} J_0}, \quad (9)$$

$$E = \frac{1}{2} [2\mu_0(\Delta p/B_{z,s}^2) - 1]^{-1/2}. \quad (10)$$

For the case without active coils, the parameters derived from the model are given in Table I, together with the values obtained from direct scans of plasma with magnetic probes.³ The value of Δp is found from Fig. 8. However, it can also be estimated from the global power balance.¹³ For our case of weakly ionized plasma, the atomic processes are dominated by electron-neutral collisions that result in ionization and radiation losses. The power balance reduces to¹³

$$n_e(T_e) = \frac{P_{\text{rf}}[W]/(en_n V)}{[(\chi/D + 3\alpha/2)T_e + E_{\text{ion}}]\langle\sigma v\rangle_{\text{ion}} + E_{\text{rad}}\langle\sigma v\rangle_{\text{rad}}}, \quad (11)$$

where P_{rf} is the total heating power. Here, the neutral particle density n_n is set to $2.25 \times 10^{19} \text{ m}^{-3}$ (for $p_f = 0.7 \text{ mTorr}$), ionization energy for hydrogen is $E_{\text{ion}} = 13.6 \text{ eV}$, ionization rate coefficient is $\langle\sigma v\rangle_{\text{ion}} \approx 1.26 \times 10^{-14} \text{ m}^3 \text{ s}^{-1}$ for $T_e = 19 \text{ eV}$, and electron-neutral impact excitation rate coefficient $E_{\text{rad}}\langle\sigma v\rangle_{\text{rad}} \approx 1.92 \times 10^{-13} \text{ eV m}^3 \text{ s}^{-1}$. The ratio of thermal conduction to particle diffusivity χ/D is set to zero, and parameter α , the ratio of edge to central electron temperature, is set to 1. For estimated $P_{\text{rf}} = 95 \text{ kW}$ and calculated volume of plasma within the separatrix $V = \frac{4}{3} \pi r_s^3 E$

$= 0.0297 \text{ m}^3$, Eq. (11) gives $n_e \approx 1.23 \times 10^{18} \text{ m}^{-3}$ that corresponds to $p_e \approx 3.73 \text{ Pa}$. Both values are close to those measured in experiment.

Table I shows a good agreement between experiment and the model, except the value of plasma current. In the model, the plasma current is calculated as

$$I_p \equiv -I_\theta = - \int_{-z_s}^{z_s} dz \int_0^{\sqrt{r_s^2 - z^2/E^2}} J_\theta dr \\ = \frac{8}{3} r_s E \left[\frac{2\Delta p}{\mu_0} \left(1 + \frac{1}{4E^2} \right) \right]^{1/2}, \quad (12)$$

based on current density given by Eq. (6), which is a linear function of radius. The actual profile of current density has two peaks,³ $J_p \approx 60 \text{ kA/m}^2$ at $r \approx 6\text{--}8 \text{ cm}$ and $J_p \approx 90 \text{ kA/m}^2$ at $r \approx 16\text{--}18 \text{ cm}$. Changing the value of J_0 in the model from -40 to -60 kA/m^2 , however, would result in a much smaller r_s than observed in experiments. The issue can be addressed with a more sophisticated model for plasma equilibrium that has realistic types of $J_\theta(r)$ profiles.

The results above are shown for configuration without active coils, as displayed in Fig. 3. When the active coils are energized, the total plasma current is increased from 2.2 to 5 kA, as seen in Fig. 4; the peak plasma density is increased from $(1.2\text{--}1.3) \times 10^{18}$ to $(1.6\text{--}1.8) \times 10^{18} \text{ m}^{-3}$. The plasma current density profile derived from the scan of magnetic field^{3,14} is also significantly changed; the inner current peak is nearly doubled in magnitude. Although a simplified equilibrium model cannot accurately describe the observed changes in $J_\theta(r, z)$, we notice from experimental data that the value of ψ_0 remains constant, even during reshaping the plasma to a doublet configuration.^{3,4,14} From Eq. (8), the condition $\psi_0 = \text{const}$ is equivalent to $B_{z,s} r_s^2 = \text{const}$. Using Eq. (9), we obtain

$$J_0/(r_s \Delta p) = \text{const}. \quad (13)$$

This condition describes how the model current density should be changed to satisfy $\psi_0 = \text{const}$. It is much less restrictive than assumption of $J_\theta(r) = -n_e e w_{\text{rf}} r$ (limit of RMF full penetration) used in equilibrium analysis of RMF plasmas.^{15–17}

TABLE II. Parameters calculated from the model and measured in experiment with active equilibrium coils.

Parameter			Model	Experiment
Input for the model	Field at separatrix	$B_{z,s}$	(G)	35.0
	Plasma pressure	Δp	(Pa)	5.20
				30–35
				5.0–5.5
	Current density on axis	J_0	(kA/m ²)	–48.1
				–35 to –60
	Separatrix radius	r_s	(cm)	17.48
				≈17
	Magnetic axis radius	r_0	(cm)	12.36
				≈12
	Plasma elongation	E		1.934
				1.7–1.9
	Poloidal flux on axis	ψ_0	(mWb)	–0.084
				–0.12 to –0.14
	Plasma volume	V	(m ³)	0.0433
	Plasma current	I_p	(kA)	2.68
				4.2–5.0
	Input power	P_{rf}	(kW)	200
				170–180

The parameters derived from the model are given in Table II, alongside with experimental values. The value of J_0 in this case is obtained by using Eq. (13),

$$J_0^{(2)} = J_0^{(1)} \frac{\Delta p^{(2)}}{\Delta p^{(1)}} \left(\frac{B_{z,s}^{(1)}}{B_{z,s}^{(2)}} \right)^{1/2}, \quad (14)$$

where the upper index refers to Table I (without active coils) or Table II (with coils). The value of $\Delta p^{(2)} = 5.20$ Pa is found from Fig. 8, but it can be obtained self-consistently from global power balance. The Ohmic heating power can be calculated as

$$P_{\text{Ohm}} = 2\pi\eta \int_{-z_s}^{z_s} dz \int_0^{\sqrt{r_s^2 - z^2/E^2}} J_\theta^2 r dr = \eta \frac{256\pi\Delta p}{15\mu_0} \left(1 + \frac{1}{4E^2} \right) r_s E = \eta \frac{6\pi}{5} \frac{I_\theta^2}{r_s E}. \quad (15)$$

Substituting values from Tables I and II, we see that P_{Ohm} is increased by factor of 2.11 after the active coils are energized. Assuming that P_{rf} is proportional to P_{Ohm} , we obtain $P_{rf}^{(2)} \approx 2.11 P_{rf}^{(1)} = 200$ kW. Substituting this value for P_{rf} and $V = 0.0433$ m³ from Table II into Eq. (11), we obtain $n_e \approx 1.64 \times 10^{18}$ m^{−3} that corresponds to $p_e \approx 5.26$ Pa. This value is consistent with experiment and with input Δp value for the model.

Although the model is not accurate, it can be used to predict how the geometry of plasma would change under the effect of programed equilibrium field, at least for our range of parameters.

IV. ROTAMAK-FRC REGIME

In the rotamak-FRC regime, the use of active coils is found to be far less efficient than in the ST case. The rotamak-FRC plasma becomes unstable if the magnitude of current in the active coils exceeds certain level.

Without the active coils, the change in boundary poloidal flux is very similar to that in rotamak-ST case (shown in Fig. 2). At $|z| < 7$ cm, the boundary flux drops to zero in 1 ms after starting RMF.

With the active equilibrium control, the maximum current I_c that could be applied to active coils is found to be

0.6 kA, twice smaller than in ST regime. The temporal change in the boundary poloidal flux and plasma current is shown in Fig. 9. It is seen that for $|z| < 7$ cm the boundary flux is sustained at the level of 0.05–0.08 mWb, far worse than the results achieved in rotamak-ST regime (Fig. 4). When $I_c = 0.6$ kA is applied, the peak electron pressure is

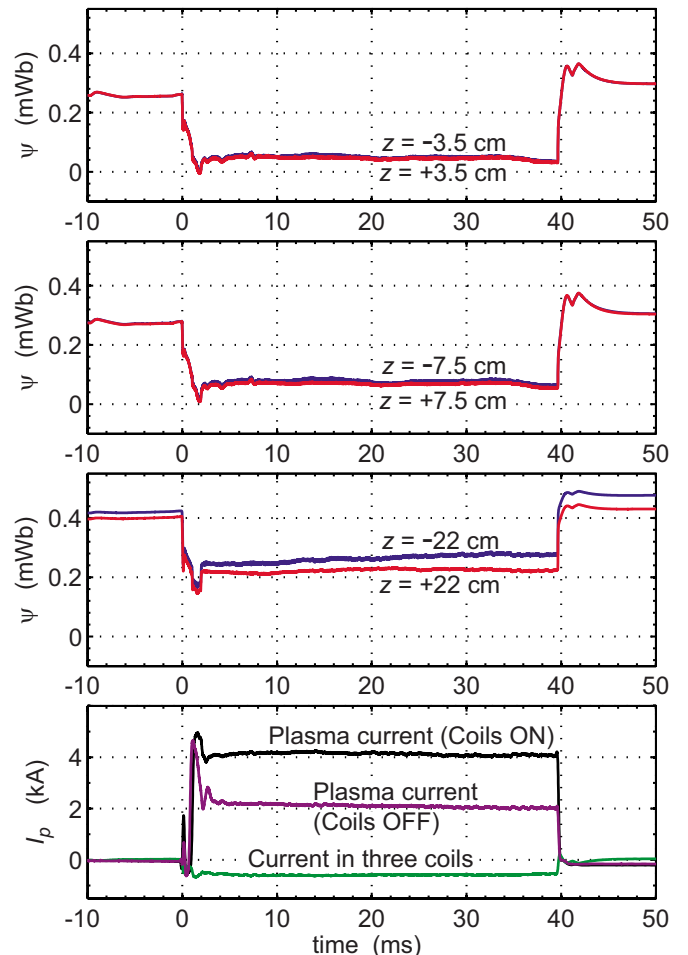


FIG. 9. (Color online) Measured boundary poloidal flux and plasma current during rotamak-FRC discharge with active equilibrium coils. For comparison, plasma current in discharge without active equilibrium coils is also shown.

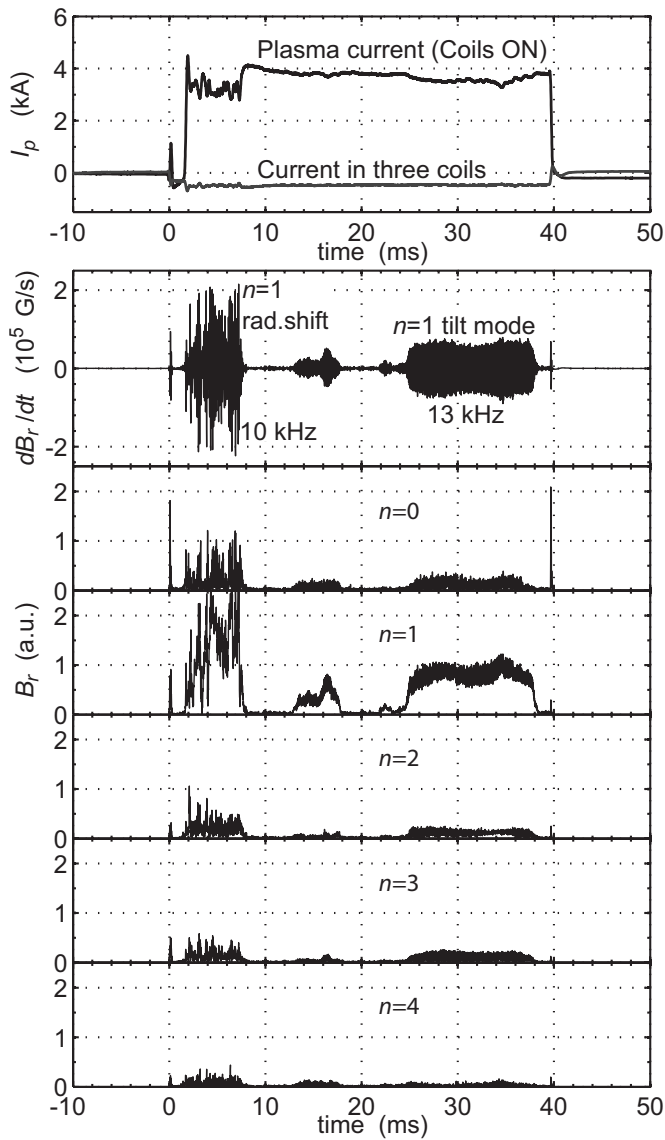


FIG. 10. Plasma current, MHD fluctuation signal from one of Mirnov coils, and toroidal modes' magnitudes during rotamak-FRC discharge with active equilibrium control.

changed from 3.8 to 4.8 Pa. Unlike the ST case, it remains flat within $r=6\text{--}17$ cm.

Based on the measurements of magnetic field fluctuations at plasma edge we conclude that the limitation for the current in the active coils is due to magnetohydrodynamic (MHD) instabilities in rotamak-FRC regime. One typical example is shown in Fig. 10. Fluctuations of B_r component are observed during rotamak-FRC discharge, with $n=1$ toroidal mode having the largest amplitude. Mode types are identified by phase difference between two sets of Mirnov coils at $z=-4$ cm and $z=4$ cm. When the phase difference is small (less than $\pi/4$), the mode is recognized as a tilt mode; when the phase difference is π , it is identified as a radial shift mode.

From Fig. 10, we can see that the radial shift mode starts at $t=1.9$ ms, as soon as the plasma current is generated. The oscillating frequency of the mode is approximately 10 kHz, and the magnitude is 2–3 G. It is seen that the excitation of

the radial shift mode results in a significant, about 20%, drop of plasma current. When the current in the active coils exceeds $I_c=0.6$ kA, the radial shift mode usually becomes so large that the plasma current is disrupted. In Fig. 10, corresponding to $I_c=0.5$ kA, the mode was stabilized after $t=8$ ms.

The results shown in Figs. 9 and 10 correspond to configuration without copper rings. When the four rings are added to the outside of chamber, as shown in Fig. 1, the rotamak-FRC plasmas become less stable. A brief gap in $I_p(t)$ seen in Fig. 9 during $\Delta t=0.2\text{--}1.0$ ms widens in the presence of copper rings to $\Delta t=0.25\text{--}4.5$ ms. After $t\approx 4.5$ ms, the plasma current emerges, accompanied by a strong radial shift mode (about 50% larger than that shown in Fig. 10), which later disappears.

It is also seen in the figure that during $t=13\text{--}18$ and $25\text{--}38$ ms another mode appears which is identified as the tilt mode. The oscillating frequency is 13 kHz and the amplitude is up to 1 G. Usually this mode grows in 1–2 ms and then becomes saturated; its magnitude slowly changes in time (comparing to eruptive changes in the radial shift mode magnitude). It should be noted that even without active equilibrium control, the tilt mode is present in rotamak-FRC discharges. Adding four copper rings does not lead to stabilization of this mode.

The tilt mode rotates in electron direction with angular speed $2\pi \times 13$ kHz $\approx 8 \times 10^4$ rad/s. The two-fluid Hall-MHD theory¹⁸ yields the dispersion relation for the internal tilt mode that for the case of immobile ions can be written as

$$\hat{\omega}^2 - \hat{\omega} \frac{11\hat{\gamma}^2}{6S_*} \left(E + \frac{1}{E} \right) + \hat{\gamma}^2 = 0. \quad (16)$$

Here, E is the plasma elongation and $\hat{\gamma}^2 = (15E^2 - 31)/(11 + 11E^2)$ corresponds to the growth rate in the ideal MHD limit normalized to the Alfvén inverse time V_A/z_s , and $\hat{\omega} = \omega/(V_A/z_s)$ is the normalized perturbation frequency. Kinetic parameter $S_* = r_s/(c/w_{pi})$, which is the ratio of the separatrix radius to the ion skin depth, is $S_* \approx 1$ in our plasmas. The solution of Eq. (16) in this case possesses $\text{Re}(w) \approx 0.25V_A/z_s \approx 6 \times 10^4$ rad/s for the range $0.6 < E < 1.2$, which is close to the measured angular speed.

According to our recent experimental study,¹⁴ the tilt mode can be completely suppressed in a doublet-shape rotamak-FRC, which is produced by applying $I_c \approx 0.25\text{--}0.50$ kA total current to the midplane active coil (side coils being inactive). In many present RMF-driven FRC plasmas,^{5,6} the tilt mode is stabilized by kinetic effects due to large ion Larmor radii and high elongation of plasma.¹⁹

In contrast, the radial shift mode could not be stabilized by reshaping the plasma with active coils. According to theoretical prediction,²⁰ the radial shift mode should be suppressed for $E > 0.69$, where elongation of plasma E is defined as the ratio of the separatrix half length to its radial extension. Without the active equilibrium control, the elongation of plasma is approximately 0.9 (Fig. 3); no radial shift mode is observed in this case. When each of the side coils is energized with $I_c \geq 0.2$ kA (midplane coil being discon-

nected), the elongation of plasma is reduced to $E \leq 0.7$; the discharge is disrupted by large amplitude radial shift mode. This observation confirms the theoretical prediction that smaller elongations $E \leq 0.7$ are favorable for excitation of radial shift mode. However, when all three active coils are energized resulting in $E > 1$, the radial shift mode is still present and often leads to plasma disruption, as shown in Fig. 10. Moreover, when the four copper rings are added to the equilibrium system, the amplitude of the radial shift mode becomes even larger, contrary to expectation that the conducting structures should stabilize it.²¹ A possible explanation for high growth rate of the radial shift mode in our plasmas is that the active coils and copper rings produce a nonuniform field at the boundary. In our device, the gap between the copper rings could not be reduced because of ports' position, and the active coils could not be moved away from chamber surface because a higher power current source would be needed to produce same magnitude field.

V. SUMMARY

An active equilibrium control is implemented for the first time in RMF-driven plasma. The effectiveness of the system depends on the presence of external toroidal magnetic field. In rotamak discharges with toroidal field, the total current in active equilibrium coils can be as large as 1.2 kA without causing disruptions. Applying 1 kA total current to active coils results in sustainment of the magnetic boundary flux during the 40 ms shot. The plasma elongation is doubled and the plasma current is increased from 2.2 to 5 kA comparing to discharges without active equilibrium control. The increases in total current and absorbed power are mainly due to the increase in FRC length when the active coils are employed. The change in plasma geometry is consistent with the Solov'ev equilibrium model and global power balance. The plasma pressure is increased by 50%–100% in plasma core, but plasma temperature remains almost unchanged. The plasma is still weakly ionized, unable to overcome radiation barrier because of impurities. Presumably the large influx of impurities from chamber wall is produced by unmagnetized ions hitting the (untreated) glass surface.

In rotamak discharges without external toroidal field, the working range for the equilibrium control system is found to be limited by MHD instabilities, particularly by $n=1$ radial

shift mode. Only 0.6 kA total current could be applied to active coils before the plasma is disrupted. Presumably a more complicated equilibrium control system would be needed, which could provide an axially uniform magnetic field. The active coils should be positioned further away from plasma edge, and energized with higher current.

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Suppression of $n = 1$ Tilt Instability by Magnetic Shaping Coils in Rotamak Plasmas

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Measurements from the array of Mirnov magnetic coils provide the first evidence for $n = 1$ tilt and radial shift instabilities in a 40 ms field-reversed configuration (FRC) driven by rotating magnetic field. External plasma-shaping magnetic coils are utilized to suppress the $n = 1$ instability modes. It is demonstrated that by energizing the middle shaping coil with 250–500 A current, the tilt mode is completely suppressed when a doublet FRC with an internal figure-of-eight separatrix is formed.

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Rotamak is a device that can form and sustain a field-reversed configuration (FRC) confined plasma with no or negligible toroidal magnetic field [1]. Such configuration is considered as a promising alternative for a fusion reactor because of its compact and simple design, natural divertor, and high plasma β (ratio of plasma pressure to magnetic field pressure). In a rotamak, the toroidal plasma current is driven in a steady-state, noninductive fashion by means of an externally applied rotating magnetic field (RMF) [2,3]. With this method, a sufficiently large plasma current can be generated to reverse the external equilibrium magnetic field on the symmetry axis so that a FRC is formed [3–10].

The principal concern about an FRC plasma is its stability to low- n (n is the toroidal number) magnetohydrodynamic (MHD) modes. Among them, the $n = 1$ tilt or shift modes are predicted to be most dangerous for global stability. The tilt instability originates from the trend of the plasma current ring to flip in order to align its magnetic moment with the external magnetic field. The radial shift instability occurs when the magnitude of external magnetic field decreases with radius, causing the Lorentz forces on the opposite sides of the plasma current ring to become increasingly unbalanced if the current ring is shifted in the radial direction. Theoretical models predict the tilt mode to be unstable for elongations of plasma $E > 0.5$, where $E = Z_S/(R_S - R_i)$ is the ratio of the separatrix half-length to its radial extension, and the radial shift mode to be unstable for $E < 0.7$ [11]. Thus there is no range for E where the plasma is expected to be free from $n = 1$ instabilities.

Although the instabilities are predicted to destroy plasma at a time scale of Z_S/V_A , where V_A is the Alfvén speed, most of experiments to date demonstrate that FRCs remain grossly stable, with a lifetimes factor of hundreds longer than the Alfvén transit time. The stabilization of $n = 1$ modes is partially attributed to nearby passive conducting structures, to high viscosity of present FRC plasmas, and finite Larmor radius effects [11–13]. Experiments have shown that the amplitude of tilt mode is reduced by a factor of 10 when the value of S_*/E is smaller than 3, for a range of $S_* \sim 8$ –40 and $E \sim 3$ –8 [14]. Kinetic parameter $S_* = R_S/(c/\omega_{pi})$, which is the ratio of the separatrix radius to the ion skin depth, also counts the number of thermal ion

gyroradii in case of $T_i \gg T_e$, when c/ω_{pi} approximately equals the ion gyroradius in the external field.

But if this empirical stability criterion is extrapolated to fusion reactor plasmas, where the values of $S_* \sim 100$ –200 are needed for achieving ignition [14], the elongation and length of an FRC becomes unrealistic. Therefore, the study of additional stabilizing techniques is still needed for future large-size FRCs.

The newly constructed Prairie View (PV) Rotamak is uniquely positioned for such study. The 40 ms long duration of plasma discharge allows for observing global MHD instability modes even if their growth time is much longer than the Alfvén transit time. In addition, a set of shaping magnetic coils permits plasmas to form with $E < 1$ (oblate FRC), $E > 1$ (prolate FRC), and doublet-shape configurations with two magnetic axes. The main questions for the study are: (1) What kind of MHD instability occurs in our rotamak FRC plasmas and which modes result in plasma disruption? (2) Can MHD instabilities be completely suppressed with shaping magnetic coils?

In this Letter, we report the first observation of the $n = 1$ tilt and radial shift modes in rotamak FRC plasmas. Experimental results demonstrate that in RMF FRC plasmas, the $n = 1$ tilt mode becomes saturated after a long growth time, which can be up to 10 ms (~ 2000 of Alfvén transit time). In contrast, when the $n = 1$ radial shift mode occurs, it leads to a fast destruction of plasma. As shown in this Letter, stability of modes is strongly related to plasma shape. Tilt mode is completely suppressed when a doublet-shape FRC with an internal figure-of-eight separatrix is formed by energizing the shaping magnetic coil at the midplane. We show that there is an optimum doublet-shape configuration that is completely free from both tilt and radial shift modes.

The Rotamak device [9] is illustrated schematically in Fig. 1. The glass chamber includes the center column covered with quartz tube; there are no conducting surfaces facing the plasma. Three shaping magnetic coils are wound over the chamber surface and connected to a programmable current source with current rise time of 1–2 ms. One middle coil (8 turns) is located at $Z = 0$, two side coils (4 turns each) at $Z = \pm 8$ cm. To characterize MHD insta-

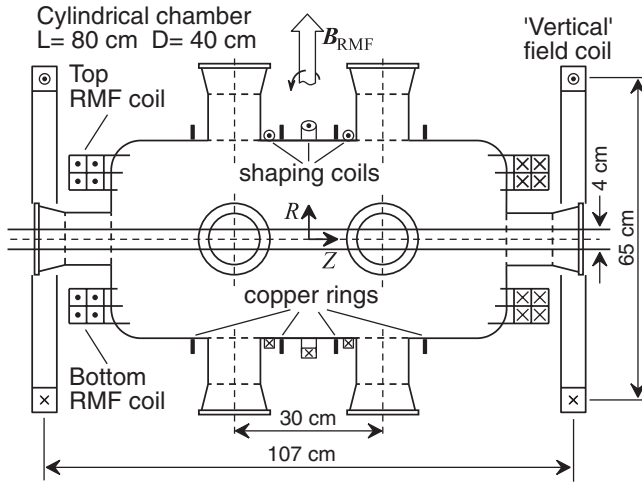


FIG. 1. The schematic of the PV Rotamak.

bility, two sets of Mirnov magnetic coils are located at $Z = \pm 4$ cm. At each Z position, eight B_R -oriented coils are mounted around chamber surface at equal 45° θ intervals (θ is the toroidal angle). The “vertical” field coils provide the vacuum magnetic field $B_v = 25$ G at the chamber center. No external toroidal field is used in this study. Hydrogen gas is continuously fed through the chamber at filling pressure $p_f = 1.3$ mTorr. The rf power ($f_{\text{rmf}} = 500$ kHz) delivered to the plasma is 200 kW that produces RMF components $\tilde{B}_R \approx 10$ G and $\tilde{B}_\theta \approx 20$ G at the plasma edge. The plasma temperature and density are measured by double floating Langmuir probes. Typically, $T_e = 15$ –20 eV and $n_e = 1.2$ – 2.5×10^{12} cm $^{-3}$.

It is seen that the plasma is weakly ionized: the filling pressure $p_f = 1.3$ mTorr corresponds to the neutral density $n_n = 4.5 \times 10^{13}$ cm $^{-3}$, about 20–30 times larger than the electron density. The collision rate of electrons with neutrals is $\nu_{en} = n_n \sigma_{en} V_{Te} \approx 2 \times 10^6$ s $^{-1}$, about 3 times higher than ν_{ei} . The electron collision rate is still much smaller than their gyrofrequency $\omega_{ce}/2\pi = 7 \times 10^7$ s $^{-1}$. For ions, assuming $T_i = 5$ eV, the collision rates are $\nu_{in} = n_n \sigma_{in} V_{Ti} \approx 5 \times 10^5$ s $^{-1}$ (based on $\sigma_{in} = 5 \times 10^{-19}$ m 2) and $\nu_{ii} \approx 8 \times 10^4$ s $^{-1}$, while $\omega_{ci}/2\pi \approx 4 \times 10^4$ s $^{-1}$. Therefore, the ions are not magnetized and can be treated as an immobile background, while the electron fluid is described by Ohm’s law

$$\mathbf{E} - (\mathbf{J} \times \mathbf{B} - \nabla P_e)/n_e e = \eta \mathbf{J}. \quad (1)$$

The main difference from the ideal MHD model is the presence of the Hall current, which can provide additional stabilization for the tilt mode [15].

A two-component internal magnetic pickup probe was used to perform radial scans at axial positions $Z = 0, \pm 5, \pm 11, \pm 17$ and ± 22.5 cm. After the 2D scan is completed, the field components are interpolated to intermediate radial and axial points. As seen in Fig. 2(a), when no shaping coils are energized, the closed magnetic surfaces are quite

oblate, in spite of the fact that the RMF antennae extend to the end of the chamber. This plasma shape is confirmed by an electron density scan with a Langmuir probe [9]; the plasma is located mainly within the area of $|Z| < 15$ cm, the position of maximum density being at $Z = 0$ cm and $R \approx 15$ cm. The nearly spherical shape of the plasma might be the consequence of two factors: the high mirror ratio of the equilibrium vertical field, and the ponderomotive force originating from the inhomogeneous oscillating electromagnetic field (RMF) that pushes plasma towards the midplane [16]. From Fig. 2(a) we estimate that when no shaping coils are energized, the elongation of the plasma is $E \approx 0.9$.

For plasma discharges without shaping coils, the $n = 1$ mode often appears during discharge, as seen in Fig. 3. Judging by the small phase shift between signals from two sets of Mirnov coils ($Z = -4$ cm and $+4$ cm), we identified it as a tilt mode. This mode usually grows within a period of 1–2 ms, and then becomes saturated during the shot. Depending on vertical field and filling gas pressure, the oscillation frequency of the mode is between 15 and 19 kHz; the growth time can be up to 10 ms. This tilt mode is nondestructive, but it usually results in a plasma current drop by 10%. As indicated in Fig. 3, when B_v is increased from 25 G to 28 G, the tilt mode starts later and grows more slowly, and its frequency and amplitude gradually becomes smaller with higher B_v . Measurements with poloidal flux loops show the elongation E reduces from 0.9 to 0.7 when B_v is increased from 25 G to 30 G. This result qualitatively agrees with theoretical prediction [11] that the growth rate for tilt mode drops for smaller elongation, and the mode is completely suppressed at $E < 0.5$. When B_v is slightly

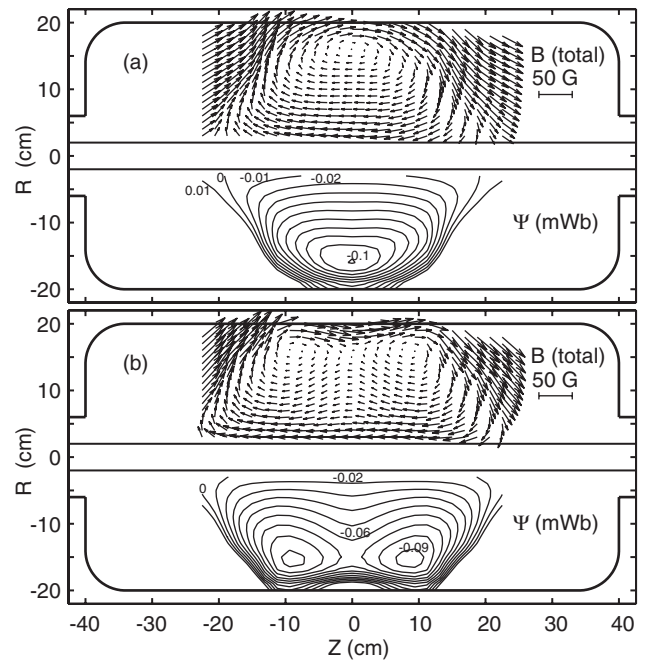


FIG. 2. Measured magnetic field and poloidal flux (a) before, and (b) after the middle shaping coil is energized.

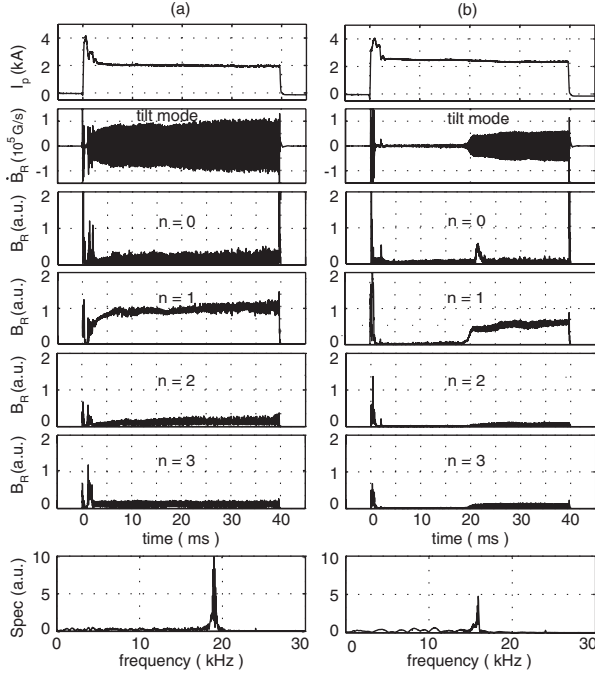


FIG. 3. Time evolution of plasma current, MHD oscillation signal, magnitude of $n = 0-3$ perturbations to B_R and spectra for (a) $B_v = 25$ G; (b) $B_v = 28$ G.

larger than 30 G, tilt mode becomes very small or even disappears, but another mode with oscillation frequency around 10 kHz appears. It is identified as $n = 1$ radial shift mode (the phase shift between two sets of Mirnov coils is π). Usually excitation of radial shift mode results in plasma disruption within 1 ms. According to theoretical study [11], the radial shift mode is expected to become unstable at sufficiently small elongations ($E < 0.69$), in good agreement with our measurements.

From measurements, the tilt mode rotates in electron direction with angular speed $2\pi \times 16$ kHz $\approx 10^5$ rad/s. The two-fluid Hall MHD theory [15] yields the dispersion relation for the internal tilt mode ($E > 1$) that for the case of immobile ions can be written as

$$\hat{\omega}^2 - \hat{\omega} \frac{11\hat{\gamma}^2}{6S_*} \left(E + \frac{1}{E} \right) + \hat{\gamma}^2 = 0. \quad (2)$$

Here, $\hat{\gamma}^2 = (15E^2 - 31)/(11 + 11E^2)$ corresponds to the growth rate in the ideal MHD limit normalized to V_A/Z_S , and $\hat{\omega} = \omega/(V_A/Z_S)$ is the normalized perturbation frequency. In our case of $S_* = R_S/(c/\omega_{pi}) \approx 1$, the solution of Eq. (2) possesses $\text{Re}(\omega) \approx 0.25V_A/Z_S \approx 6 \times 10^4$ rad/s for the range $0.6 < E < 1.2$, which is close to the measured angular speed of 10^5 rad/s. The linear growth rate is given by $\text{Im}(\omega)$ which is positive in a narrow range of elongation, $1.437 < E < 1.645$. The highest growth rate, $\text{Im}(\omega) \approx 0.25V_A/Z_S \approx 6 \times 10^4$ rad/s, is reached at $E = 1.54$. It should be noted, however, that for $E \approx 1$ the mode evolves into external tilt, for which Eq. (2) is not accurate. In any case, due to $V_A/Z_S \ll 2\pi\nu_{in}$, the collisions are expected to

stabilize the tilt mode for both $E < 1$ and $E > 1$ in our plasmas.

In general, the tilt mode stability is strongly affected by plasma shape. Experiments with external field shaping coils [13] proved that the tilt mode can be stabilized in very oblate plasmas ($E < 0.5$).

We examined how the plasma shaping could affect stability of tilt mode in the range of $E \sim 0.7-1.5$. The typical value of $E = 0.9$ for our plasmas can be reduced to $E \sim 0.7$ by energizing side shaping coils only (at $Z = \pm 8$ cm). In this case the tilt mode is not suppressed; besides, the radial shift mode appears, which results in plasma disruption. When all three shaping coils are energized, the elongation can be increased up to $E \sim 1.5$. In such case the tilt mode is still present, although with a smaller amplitude (sometimes undetectable).

The most striking result is seen when only the middle shaping coil (at $Z = 0$) is energized, producing doublet-shape FRCs, as shown in Fig. 2(b). As seen in Fig. 4(a), the $n = 1$ tilt mode starts growing at 4th ms and saturates at 5th ms with its oscillation frequency $f \sim 17.5$ kHz; the excitation of this mode results in I_p drop from 2.3 kA to 2.1 kA. In this shot, the middle shaping coil is energized with $I_m = 400$ A current during $t = 20-40$ ms. It is seen in Fig. 4(a) that after the middle coil is energized, the $n = 1$ tilt mode is completely suppressed. The change of elongation from $E \approx 0.92$ to $E \approx 1.37$ after the middle coil is energized cannot explain by itself the suppression of the tilt mode. As mentioned above, the tilt mode is still observed in elliptically shaped plasmas with $E \approx 1.5$, when all three shaping coils are energized.

Repeatable discharges demonstrate that the tilt mode can be completely suppressed by energizing the middle coil with current in the range of $I_m \sim 250-500$ A. When I_m current is higher, 500–600 A, the tilt mode is suppressed, but the $n = 1$ radial shift mode is excited and plasma is disrupted within 2–3 ms after energizing the coil, as shown in Fig. 4(b). For $I_m > 600$ A, plasma is disrupted immediately as the coil is energized.

Plasma global stability to $n = 1$ tilt or shift modes can be examined with the rigid-body model [12,13] which approximates plasma current by several rigid current rings that can tilt or shift in unison with each other. The onset of tilt instability is determined by value of

$$N_{\text{decay}} = -\frac{R}{B_z} \left[\frac{\partial B_z}{\partial R} - \frac{Z}{R^2} \frac{\partial}{\partial R} (2RB_R + ZB_z) \right], \quad (3)$$

which characterizes the curvature of equilibrium magnetic field at the location of current rings. The torque on the plasma is due to the $\mathbf{J} \times \mathbf{B}$ force from equilibrium field onto tilted plasma current rings. The torque acts to restore the current ring to its equilibrium position if $N_{\text{decay}} > 1$, or to increase the tilt if $N_{\text{decay}} < 1$. For our conditions, before the middle coil is energized, $N_{\text{decay}} < 0.4$, so the tilt instability should be present. This result is in agreement with experiment. When the middle shaping coil is energized, for

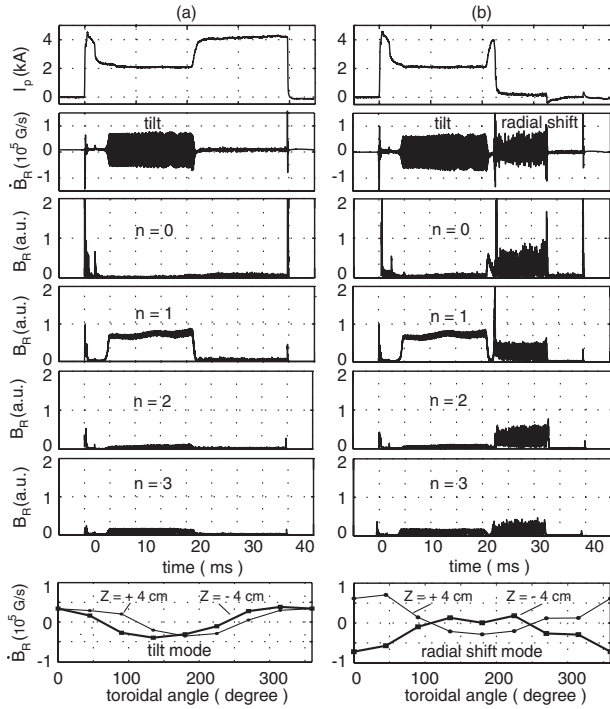


FIG. 4. Time evolution of plasma current, MHD oscillation signal, magnitude of $n = 0-3$ perturbations to B_R and signals from two sets of Mirnov coils. I_m current is fed to the middle coil during 20–40 ms (a) $I_m = 400$ A; (b) $I_m = 550$ A.

example, with $I_m = 360$ A current, N_{decay} still remains lower than 0.5 in most area of plasma. However, in experiments, the tilt mode is fully suppressed.

The apparent absence of tilt mode could be attributed to the fact that in a doublet FRC the tilt perturbation becomes localized near the FRC ends, too far from magnetic probes. To verify the measurements, a following simulation is performed. The plasma current is represented by a set of 10 current rings with different radii and axial positions to fit experimental data on the boundary magnetic flux and the structure of magnetic field. In one simulation, the current rings are uniformly positioned within $-11.25 \text{ cm} \leq Z \leq 11.25 \text{ cm}$. Each current ring is carrying -200 A current (totally -2 kA), while the two equilibrium coils at $Z = \pm 53.5 \text{ cm}$ carry 4.3 kA current each. When the current rings that represent the plasma current are tilted by $\pm 2.5^\circ$, the calculated B_R component of magnetic field at Mirnov coil's position is changed from -6.8 G to -4.8 G. Assuming that this change occurs during the half-period of the oscillating field, $\Delta t = T/2 = 0.5/18 \text{ kHz}$, the amplitude of dB_R/dt oscillations can be estimated as $|\Delta B_R/\Delta t| = 0.7 \times 10^5 \text{ G/s}$, close to that shown in Fig. 4.

In another simulation, an additional coil with $I_m = 400$ A is added at the midplane. The plasma current (-4 kA) is represented by 10 current rings positioned within $-15 \text{ cm} \leq Z \leq 15 \text{ cm}$, to reproduce the field structure shown in Fig. 2(b). The tilting angle is set to 2.5° for the current rings at $Z = \pm 15 \text{ cm}$, but gradually reducing to

zero for the current rings closer to the midplane. The calculated B_R component of magnetic field at Mirnov coil's position is 10.2 G and 11.1 G, corresponding to a different sign of the tilt angle. Therefore, the amplitude of dB_R/dt oscillations can be estimated as $|\Delta B_R/\Delta t| = 0.3 \times 10^5 \text{ G/s}$. This value is still an order of magnitude higher than the noise level of $0.05 \times 10^5 \text{ G/s}$ seen in Fig. 4(a) for $t > 20$ ms, so the tilt mode could be detected if it were present.

Thus we are inclined to conclude that it is the doublet configuration that makes the plasma stable to the tilt mode. The improved stability of doublets against other types of MHD instabilities (pressure-driven high- n interchange modes) has been predicted analytically [17]; the stability improves when E gets smaller but indentation remains moderate. It was speculated that the doublet configuration should also be stable to the cointerchange (tilt) mode.

In summary, for the first time in rotamak FRC plasmas, the $n = 1$ tilt and radial shift modes have been clearly observed and identified. The tilt mode is partially stabilized by collisional and Hall-current effects. By using a middle shaping coil with a moderate current of $250-500$ A, the doublet FRCs are formed which are completely free from both the tilt and radial shift modes. Experimental results suggest that doublet FRCs are more stable than standard oblate or prolate FRCs.

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Magnetic field structure evolution in rotating magnetic field plasmas

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A study of magnetic field structure evolution during 40-ms plasma discharge has been performed in a new device with 80 cm long/40 cm diameter cylindrical chamber, in which a plasma current $I_p \approx 2$ kA was driven and sustained by a rotating magnetic field. The main focus of the experiments is on how the changes in externally applied magnetic field affect the current profile and magnetic field in plasma. During plasma discharge, a pulse current was briefly fed to a magnetic coil located at the midplane (middle coil). The magnetic field in cross section of plasma was scanned with pickup probes. Two regimes were studied: without and with an external toroidal field (TF) produced by axial I_z current. With a relatively small current ($I_m \leq 600$ A) in the middle coil, the plasma current is boosted up to 5 kA. The magnetic flux surfaces become extended along the axial Z direction, sometimes with the formation of doublet shape plasma. The regime without TF appears to be less stable, presumably due to the reversal of plasma current in central area of plasma column. © 2008 American Institute of Physics. [DOI: 10.1063/1.2952293]

I. INTRODUCTION

Rotamak is a concept suggested in late 1970s as a method to produce a compact torus configuration by driving steady toroidal plasma current with externally applied rotating magnetic field (RMF).^{1,2} The experiments have proven that with this method, a sufficiently large plasma current can be generated to reverse the external equilibrium magnetic field on the symmetry axis, so that a field reversed configuration (FRC) is formed.^{3,4} Although extensive studies have been done for the mechanism of RMF current drive,⁵⁻¹² there is still no satisfactory model for the three-dimensional magnetohydrodynamics equilibrium in such plasmas; such a task would require extensive numerical calculations.^{13,14} Furthermore, no experimental or theoretical study has been done on how the RMF-driven plasma would respond to a change of external equilibrium field. In presently operated short-pulse devices with RMF-driven current, the basic equilibrium is provided by a stationary equilibrium “vertical” field and by flux conserving coils.¹⁵⁻¹⁷ In future long-pulse devices, an active equilibrium feedback system will be needed, similar to the systems used in modern tokamaks. Therefore, there is a need to study how the RMF-driven plasma responds to the shape change of an external magnetic field during a shot. Complicating the description is the fact that as the outermost magnetic surface of plasma is pushed or deformed, the coupling between RMF antenna and plasma is changed, and so is the driving torque on electrons, which in turn will change the magnitude of the plasma current and its distribution. The main question is: What range and what kind of changes in the equilibrium magnetic field can keep the plasma away from the chamber wall without inducing instabilities or ceasing the RMF-driven plasma current?

As the first step in studying this problem, we conducted a set of experiments in our newly built cylindrical chamber rotamak. The distinctive feature of the new device is the additional magnetic coil in the midplane placed in close proximity to the chamber side wall. In the experiments, the

coil was energized for a period of 10 ms during 40-ms plasma discharges; the magnetic field components inside plasma were measured directly by magnetic pickup coils.

The paper is organized as follows. Device parameters and diagnostic tools are described in Sec. II. In Sec. III, we show how the structure of magnetic field inside plasma is changed in FRC discharges when the current of different magnitude is briefly applied to the middle coil. In Sec. IV, the results from another series of experiments are given, where rotamak was operated with an additional external toroidal magnetic field, referred to as spherical tokamak (ST) regime. A summary is given in Sec. V.

II. EXPERIMENTAL SETUP

The new chamber is an 80 cm long, 40 cm diameter Pyrex glass vessel (Fig. 1). Its background vacuum is about 5.0×10^{-7} Torr. Hydrogen gas is continuously fed through the chamber. The gas pressure is maintained at $p_f = 1.3$ mTorr.

The preionization system powered by a 13.6 MHz/650 W rf generator is started 10 ms before the RMF is applied, and ceases 5 ms after the beginning of discharge.

The plasma current is driven by RMF that is produced by two pairs of coils. Each pair is connected through a system of matching/tuning capacitors and inductances to a separate 500 kHz/400 kW rf generator; there is a 90° phase shift between the two generators. The waveform is programmed in such a way as to maintain steady plasma current for a period of 40 ms. The experiments are conducted by using 200 kW of rf power. The plasma discharges are very stable and repeatable from shot to shot.

The equilibrium magnetic field system consists of two coils (vertical field coils) with diameter 65 cm, separated by a distance of 107 cm. In all experiments discussed here, it provides the vacuum magnetic field 25 G at the chamber center, almost steady during each plasma shot.

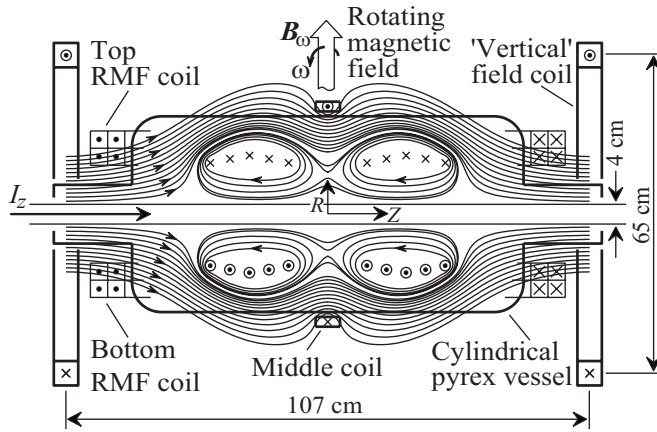


FIG. 1. The schematic of the cylindrical chamber rotamak.

The additional midplane coil (“middle coil”) is made of eight turns of welding cable (two layers by four turns) wound over the chamber’s side surface. The current in the coil is produced by discharging a capacitor. In experiments discussed in the paper, the electronic switch that connects capacitor to the coil is triggered at $t \approx 19$ ms of discharge. The current reaches its maximum at $t \approx 21$ – 22 ms, and then relaxes to zero within the following 10 ms. The value of peak current in the coil is set by capacitor charging voltage.

The device also includes a system for producing a toroidal magnetic field. A set of six wires is fed through the central axial stainless-steel pipe. The pipe is covered with quartz tube to reduce metal sputtering and to avoid development of high electric field at the pipe surface.¹⁸ The discharge current from capacitor bank threads the central axis six times resulting in a net I_z current. In experiments discussed in Sec. IV (ST regime), the value of I_z is 2.5 kA, almost unchanged during the plasma shot.

The plasma current I_p is measured by a Rogowski coil, which is pulled through the central steel pipe and looped around the poloidal cross section. The current in the middle coil that also enters the area enclosed by the Rogowski coil is later subtracted by making a separate “vacuum shot” without RMF (and hence, no plasma current).

A two-component magnetic pickup coil is used to perform complete radial scans at axial positions $|Z| = 0, 50, 110, 170$, and 225 mm. The coil is positioned inside the sealed L-shaped glass tubing, which itself can be moved in radial direction. In this way, the pickup coil can reach almost any point in poloidal cross section. The signal is filtered and numerically integrated. After the two-dimensional scan is completed, the field components are interpolated to intermediate radial and axial points.

The plasma temperature and density are measured by double floating L-shaped and straight Langmuir probes. Typically, $T_e = 15$ – 20 eV and $n_e = 1.2$ – 2.5×10^{12} cm⁻³.

III. MAGNETIC FIELD STRUCTURE CHANGE IN ROTAMAK-FRC REGIME

The response of the plasma to the magnetic field of the middle coil is found to depend strongly on the current I_m in the coil. The value of peak current in the coil is set by

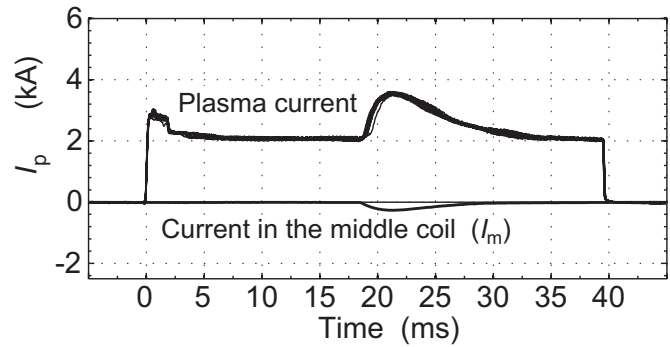


FIG. 2. Time history of the plasma current when a small current is applied to middle coil during rotamak-FRC plasma discharges. The data from 15 shots is plotted to show the reproducibility of discharges. Current in the middle coil is also shown; the negative sign corresponds to the direction opposite to plasma current.

capacitor charging voltage. Below, the results for two different charging voltages are discussed, i.e., 5 and 15 V, which correspond to the total ($I_m = i_{\text{feed}} \times 8$) peak current in the coil equal to 263 and 1080 A.

A. Small current in the middle coil: $I_{m,\text{peak}} = 263$ A

When the coil current $|I_m|$ is below a certain level, the plasma current responds with a growth, the increase being a few times larger than the magnitude of the current in the middle coil, as seen in Fig. 2.

For the initial transitional stage of 0–3 ms, when the middle coil is not yet energized, the magnetic field components are shown in Fig. 3. In addition, the current density profiles calculated with $\mathbf{J} = \nabla \times \mathbf{B} / \mu_0$ are shown. The posi-

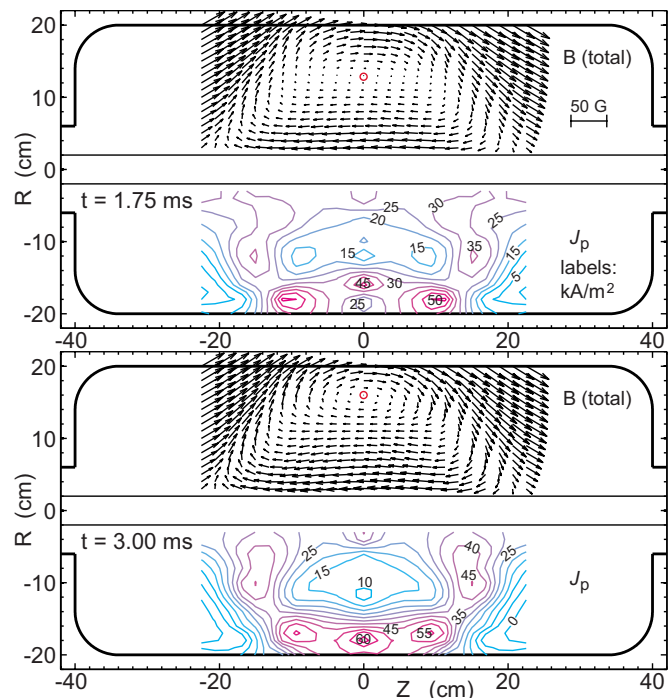


FIG. 3. (Color online) Magnetic field and plasma current density during startup stage. The magnetic field is measured with pickup coils and interpolated to a finer grid.

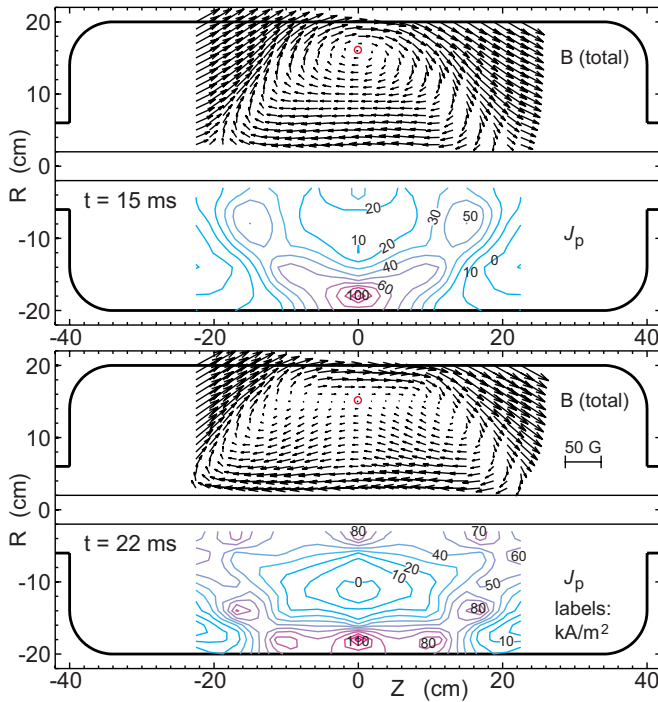


FIG. 4. (Color online) Magnetic field and plasma current density before the current in the middle coil is started (15 ms), and when it is started and reaches $|I_m| = 263$ A at $t = 22$ ms.

tion of the magnetic axis is circled. During the first 2 ms, the magnetic axis quickly moves towards the edge. After 4 ms, the magnetic structure remains almost unchanged; the plasma current is formed mostly within a ring at the mid-plane, close to the chamber side wall. It is interesting that during the initial stage ($t < 2.5$ ms), the plasma current is formed mostly within three rings: two at $Z \approx \pm 10$ cm, near the chamber wall, and one at $Z = 0$, but further away from the wall; later all three merge. The triplex profile of plasma current could not be caused by induced current in the middle coil, because the coil was not shorted out.

After the current in the middle coil is started at $t = 19$ ms, the magnetic surfaces become extended along the Z axis, as seen in Fig. 4. The profile of plasma current den-

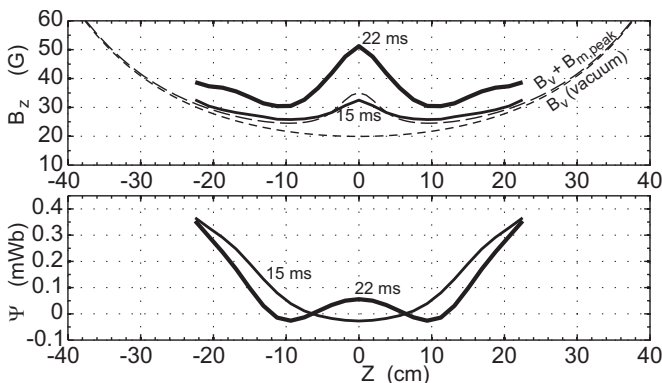


FIG. 5. Axial magnetic field and poloidal flux at the chamber wall ($R = 20$ cm) before (thin lines) and after (bold lines) the middle coil is energized. Dashed lines show the vacuum field at the chamber wall produced by the vertical field coil alone and together with middle coil.

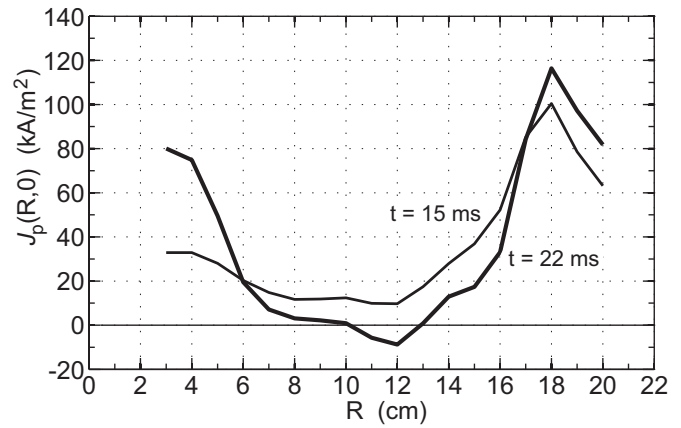


FIG. 6. Profiles of the plasma current density along $Z = 0$ chord in FRC regime, when the current in the middle coil is zero ($t = 15$ ms), and when it reaches 263 A ($t = 22$ ms).

sity becomes more elongated, and larger current is generated at $R < 5$ cm.

Based on measured magnetic field, the axial variation of B_z field and poloidal flux Ψ at the chamber wall can be plotted; they are shown in Fig. 5 with solid lines. The dashed lines show the vacuum field produced by the vertical field coils alone and together with middle coil, calculated from known currents in the coils. It is seen that the vertical field could only be considered uniform within $|Z| < 10$ cm. This might be the main reason the plasma axial length is much shorter than the RMF antennas, which lowers the current drive efficiency. Comparing with the spherical chamber rotamak,¹⁸ the plasma current in the present chamber is about 20% lower for the same input rf power. The addition of a middle coil field acts mostly to reshape the plasma rather than to push it away from chamber wall. This is confirmed by the plot of $\Psi(R = 20$ cm) in Fig. 5, which shows that when $|I_m|$ reaches a maximum ($t = 22$ ms), the flux becomes higher under the coil, but lower at $|Z| = 8 - 20$ cm.

As seen from Fig. 6, the current density grows near the chamber wall and central pipe, but at the same time it drops in the region $R = 7 - 16$ cm, so the profile becomes more hollow. The overall enhancement of the total plasma current results from larger current density near the wall and central pipe, and from axial elongation of the plasma.

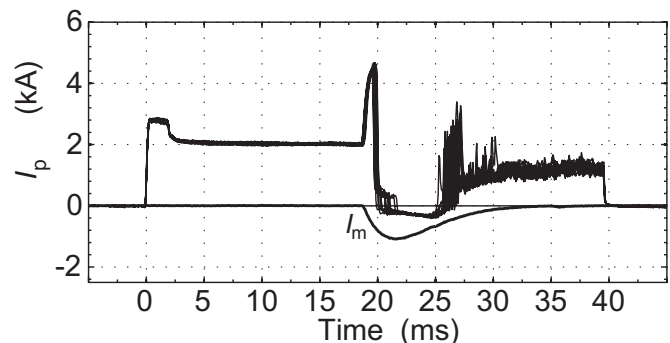


FIG. 7. Time history of plasma current (15 shots are plotted) when a large current is applied to the middle coil during rotamak-FRC plasma discharges. Current in the middle coil is also shown.

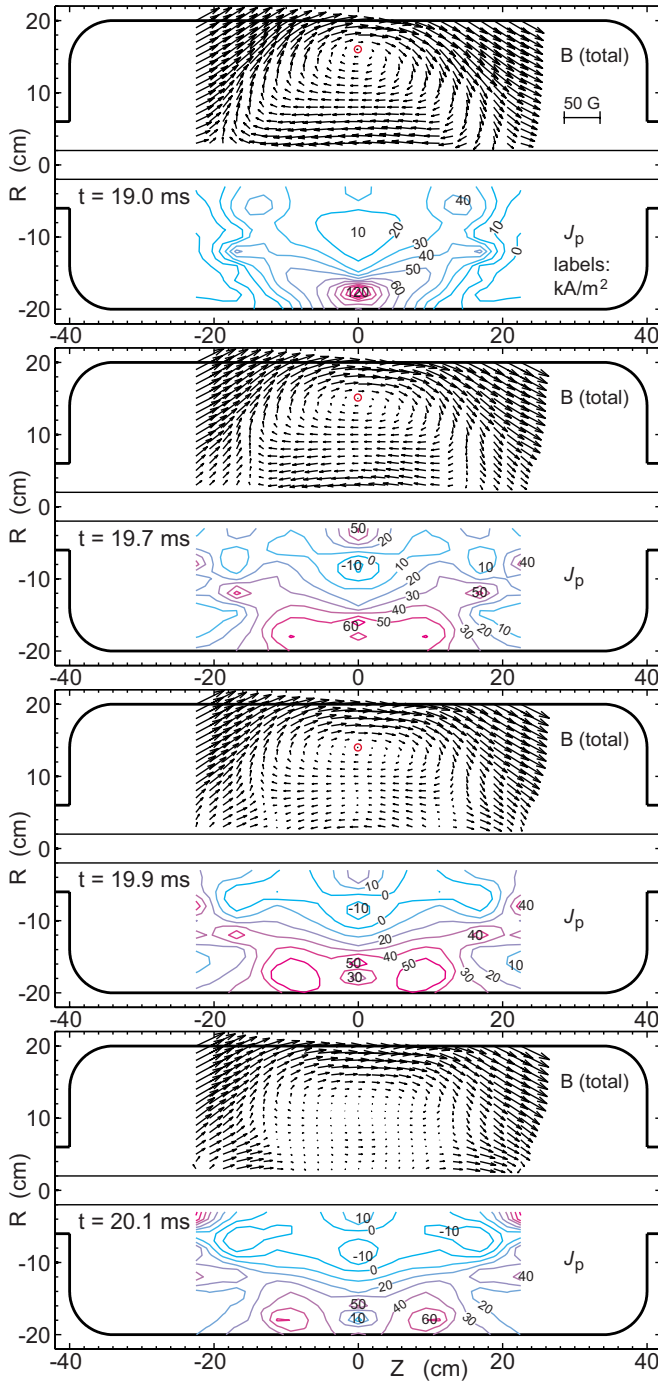


FIG. 8. (Color online) Magnetic field and plasma current density; current in the middle coil is started at $t=19$ ms, reaches disruptive level of $|I_m| \approx 600$ A at $t=19.7$ ms, then continues to grow up to 1080 A at $t=22$ ms.

It might occur that the plasma current enhancement happens because of possible conservation of $L_s I_p$ flux, where L_s is the plasma self-inductance. From the levels of $\Psi(R, Z)$ (not shown), we estimate geometrical parameters for plasma shape before and after the middle coil was energized. At $t=15$ ms, the magnetic axis position is $R_0=16.3$ cm, minor radius is $a=6.6$ cm, and the elongation is $\kappa=1.9$. After the middle coil is energized, at $t=22$ ms, these parameters change to $R_0=15.4$ cm, $a=6.9$ cm, and $\kappa=2.5$. Using model equations for the plasma self-inductance,¹⁹ we estimate:

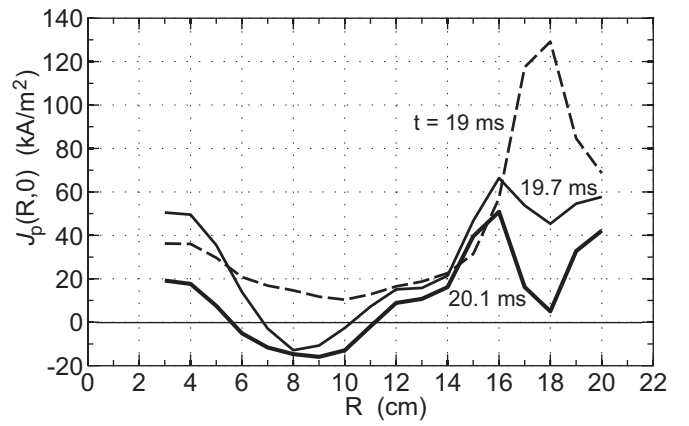


FIG. 9. Profiles of the plasma current density along $Z=0$ chord, when the current in the middle coil is just started ($t=19$ ms), reaches disruptive level of 600 A ($t=19.7$ ms), and then further grows to 850 A ($t=20.1$ ms) when the total plasma current is nearly zero.

$L_s(t=15 \text{ ms})=0.123 \mu\text{H}$ and $L_s(t=22 \text{ ms})=0.083 \mu\text{H}$, which is a 33% drop. On the other hand, the plasma current grows from 2.07 to 3.55 kA, i.e., by 71%; the net increase in $L_s I_p$ flux is thus 15%.

Therefore, the increase of plasma current is partially due to the trend of $L_s I_p$ flux to be conserved. Besides, additional current is generated because of enhancement of coupling between RMF coils and plasma. This is confirmed by a 15%–20% increase in the current flowing through the RMF coils when the middle coil is energized.

B. Large current in the middle coil, $I_{m,\text{peak}} > 600$ A

When the current in the middle coil exceeds certain level, the plasma current is disrupted. As seen from Fig. 7, for the case of $|I_{m,\text{peak}}|=1080$ A (charging voltage=15 V), the plasma current initially grows, reaching 4.5 kA at $t \approx 19.7$ ms; at that instant the current in the middle coil is $|I_m| \approx 600$ A. As the current in the middle coil exceeds this level, referred to as “disruptive” level, the plasma current suddenly drops.

Figure 8 shows that at $t \approx 19.7$ ms, when $|I_m| \approx 600$ A and plasma current is at maximum, the magnetic axis is slightly pushed away from the edge. However, at $t \approx 19.9$ – 20.1 ms the total plasma current drops to a level not sufficient to reverse the magnetic field in the center; the magnetic surfaces become open. Even though the measured total plasma current is almost zero, the current density is not. As seen from Fig. 9, while the peak in current density near the edge is reduced and moved away from the wall, a considerable negative current is generated in the region $R=6$ – 11 cm. In fact, a small negative current already appears at $t=19.7$ ms, when the total plasma current is at maximum. At later stage of $t=21$ – 25 ms, when the total plasma current remains zero or slightly negative, the region with negative current density widens to $R=4$ – 13 cm. In addition, the plasma current density becomes reversed at $R > 16$ cm, down to -80 kA/m^2 at $R=18$ cm, $Z=0$, although at $R=18$ cm, $Z=\pm 11$ cm, it remains positive at 40 – 50 kA/m^2 .

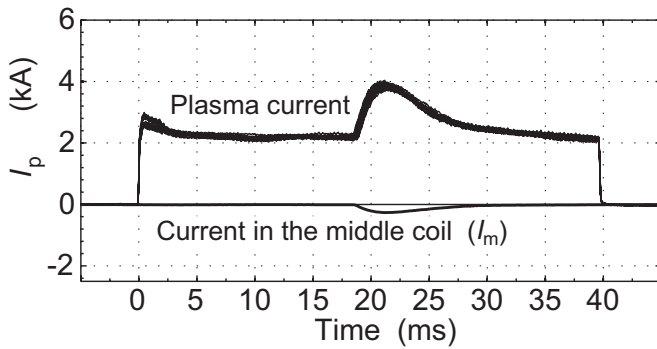


FIG. 10. Time history of plasma current (15 shots are plotted) when a small current is applied to the middle coil during rotamak-ST plasma discharges. Current in the middle coil is also shown; the negative sign corresponds to the direction opposite to plasma current.

As seen from Fig. 7, after the current in the middle coil reduces below the disruptive level (after $t \approx 25$ ms), the plasma is unable to return to the steady state configuration of $t = 3 - 18$ ms that existed before the current in the middle coil was applied. Although the plasma current reaches a 1 kA level, it remains very unstable, and the FRC configuration is not formed. This happens in most of discharges with $|I_{m,peak}| = 1080$ A (charging voltage set to 15 V); only in a few shots the plasma current returns to the 2 kA level.

Thus, when a large enough current is applied to the middle coil, the plasma current can be doubled. However, as soon as $|I_m|$ exceeds 600 A, the plasma current is disrupted and drops to nearly zero, or an even slightly negative level. In fact, we could never find such value of $|I_{m,peak}|$ that the plasma current drops to an “intermediate” magnitude of around 2 kA; it always drops to such a level that FRC is

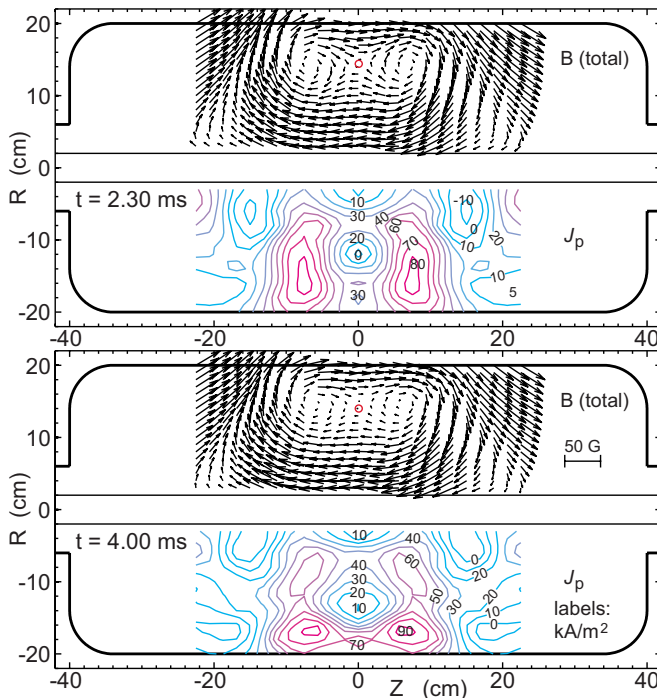


FIG. 11. (Color online) Magnetic field and plasma current density during startup stage in the ST regime.

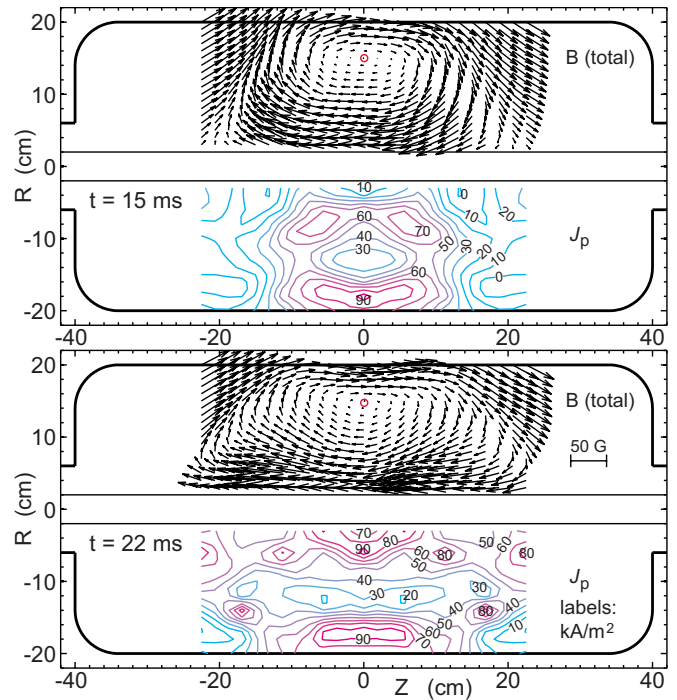


FIG. 12. (Color online) Magnetic field and plasma current density before the current in the middle coil is started (15 ms), and when it is started and reaches $|I_m| = 263$ A at $t = 22$ ms.

destroyed. A possible reason for such sensitivity to I_m is the reversal of plasma current in the inner region of $R = 4 - 13$ cm, which quickly widens with the increase of $|I_m|$. To understand how the negative plasma current is generated requires advance numerical modeling.

IV. MAGNETIC FIELD STRUCTURE CHANGE IN ROTAMAK-ST REGIME

The rotamak-ST regime is formed when an external toroidal magnetic field is added by feeding the axial I_z current through the central pipe. In all experiments discussed in this paper, the axial current is set to 2.5 kA, while the vertical field is set to the same value as in FRC regime: $B_v(0,0) = 25$ G. Below, the results for two charging voltages

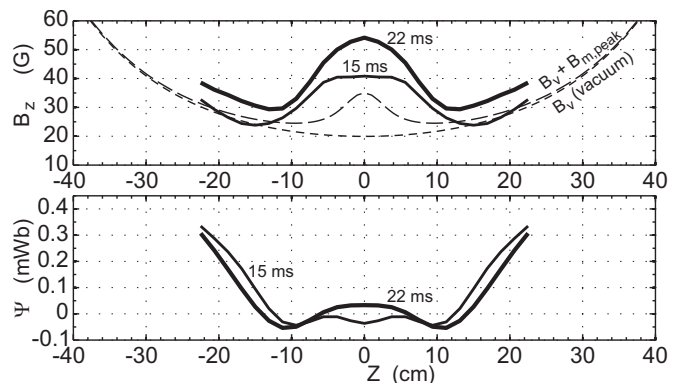


FIG. 13. Axial magnetic field and poloidal flux at the chamber wall ($R = 20$ cm) before (thin lines) and after (bold lines) the middle coil is energized. Dashed lines show the vacuum field at the chamber wall produced by the vertical field coil alone and together with middle coil.

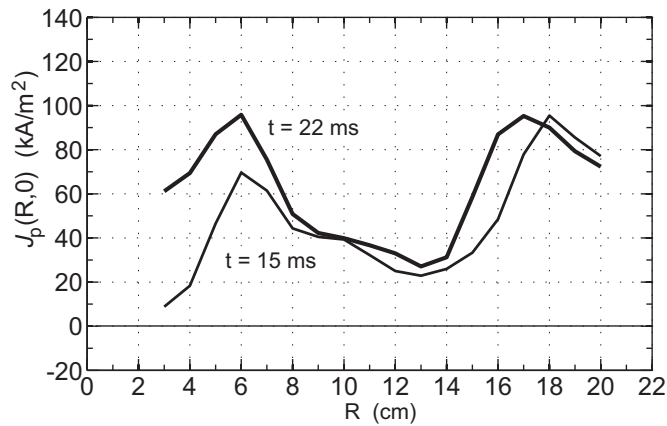


FIG. 14. Profiles of the plasma current density along $Z=0$ chord in ST regime, when the current in the middle coil is zero ($t=15$ ms), and when it reaches 263 A ($t=22$ ms).

of the middle coil capacitors are discussed, i.e., 5 and 15 V, which make the total peak current in the middle coil equal to 263 A and 1080 A, respectively.

A. Small current in the middle coil: $I_{m,peak}=263$ A

For the current in the middle coil below the disruptive level, the plasma current responds with a growth larger than in the FRC case, as seen in Fig. 10.

The scans of magnetic field show that in the initial stage of discharge ($t \approx 0-4$ ms) the plasma current is formed within two “ring” channels on both sides from the midplane, as shown in Fig. 11. Comparing to the FRC case, these two current rings are located further away from the chamber wall, and have a deeper penetration into the plasma interior. Similar to the FRC case, the two current rings merge after 5 ms, but during the steady-state stage of discharge ($t \approx 7-19$ ms) the plasma current remains more evenly distributed along magnetic surfaces than in the FRC regime. The improved

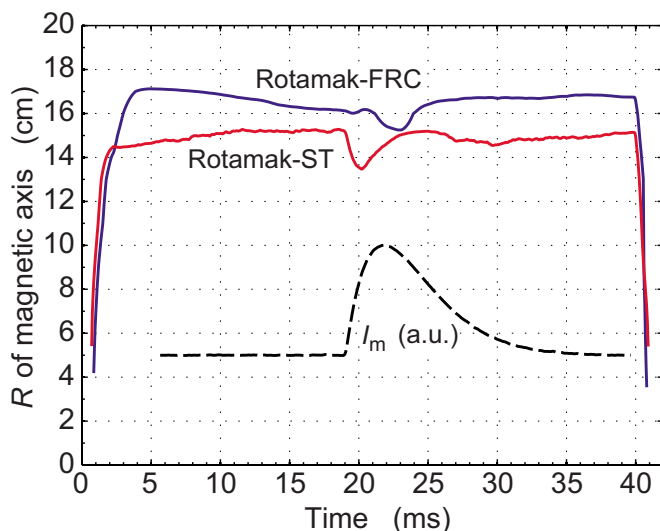


FIG. 15. (Color online) The change of the magnetic axis position during plasma shot. The time history of current in the middle coil is shown by the dashed line.

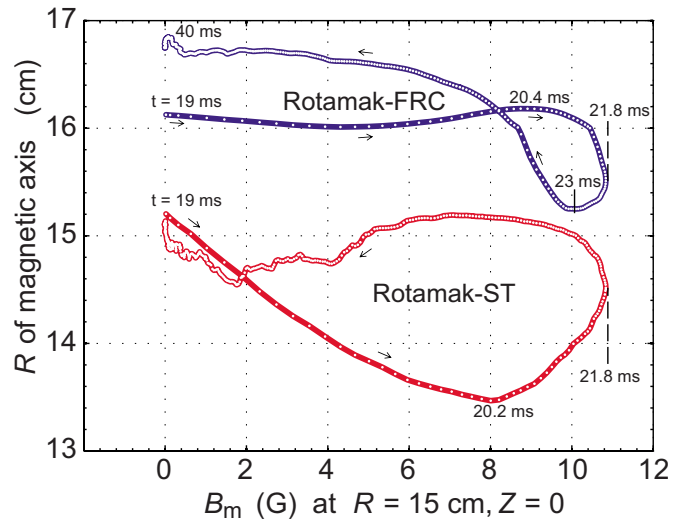


FIG. 16. (Color online) The magnetic axis position as a function of the middle-coil field magnitude.

performance of ST regime, also noted in spherical rotamaks,^{4,18,20-22} is due to better RMF penetration, which is caused by whistler wave excitation.^{18,21}

After the middle coil is energized at $t=19$ ms, the magnetic surfaces become stretched along the axial direction, as shown in Fig. 12. From levels of $\Psi(R,Z)$ (not shown), we estimate $R_0=15.2$ cm, $a=7.2$ cm, $\kappa=2.0$ at $t=15$ ms, and $R_0=14.6$ cm, $a=6.9$ cm, $\kappa=2.6$ at $t=22$ ms. Accordingly, the plasma inductance L_s drops by 19% from 0.085 to 0.069 μ H. However, the plasma current grows from 2.2 to 3.8 kA, so that the flux $L_s I_p$ grows by 39%, much higher number than in the FRC case. Thus, the increase of I_p in the ST case is largely due to enhanced coupling between plasma and RMF antennas.

The axial variation of B_z field and poloidal flux Ψ at the chamber wall are shown in Fig. 13, before and after the middle coil is energized. Similar to the FRC case, the action of the middle coil (for $I_m \leq 263$ A) is mostly to reshape the plasma rather than to push it away from chamber wall.

However, as seen in Fig. 14, the peak of the current density near the edge ($R \approx 18$ cm) is moved away from the wall by 1 cm. The plasma current is boosted in almost all

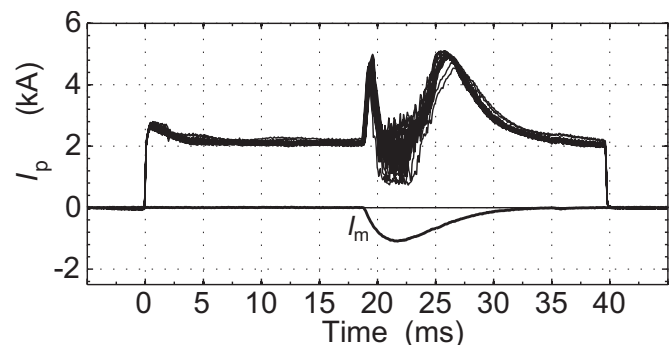


FIG. 17. Time history of plasma current (15 shots are plotted) when a large current is applied to the middle coil during rotamak-ST plasma discharges. Current in the middle coil is also shown; the negative sign corresponds to direction opposite to plasma current.

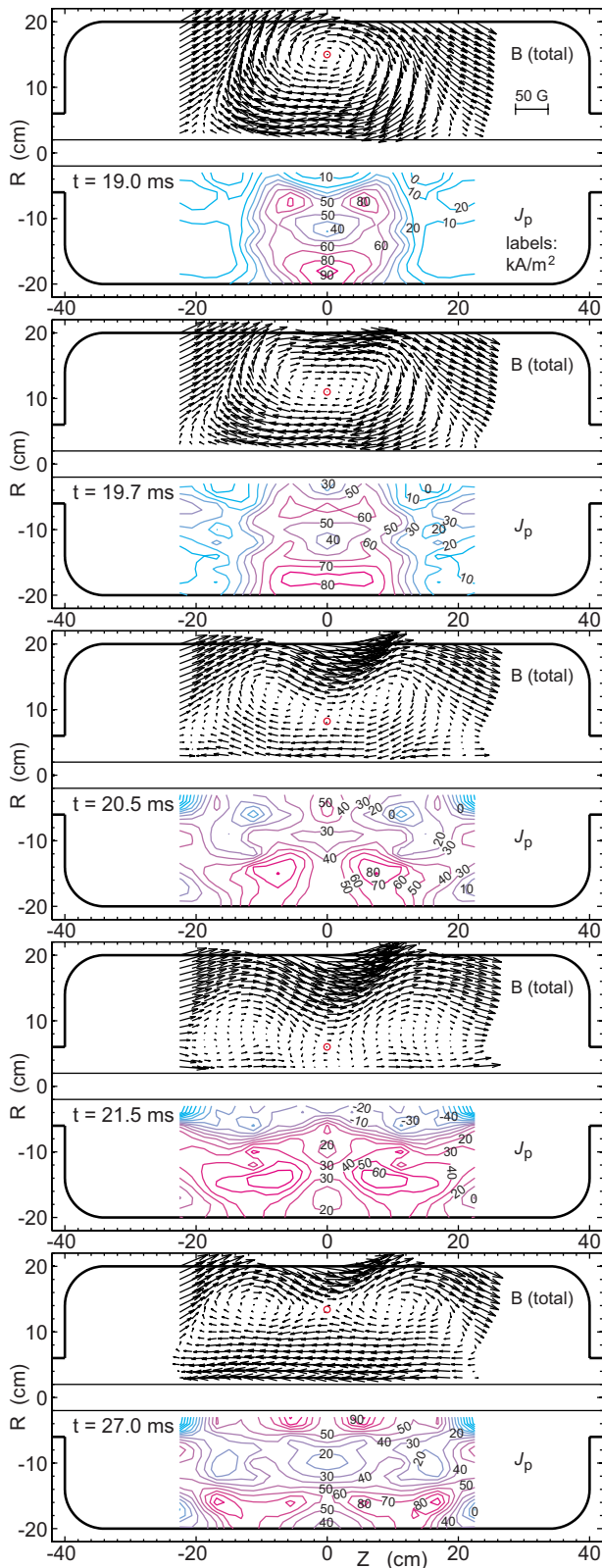


FIG. 18. (Color online) Magnetic field and plasma current density; current in the middle coil is started at $t \approx 19$ ms, and reaches the disruptive level at $t \approx 19.7$ ms; see Fig. 17.

radial range. In contrast to the FRC regime, there is no trend now for the plasma current reversal in the midrange values of R (compare Fig. 14 to Fig. 6).

The effect of the middle coil field on plasma is better

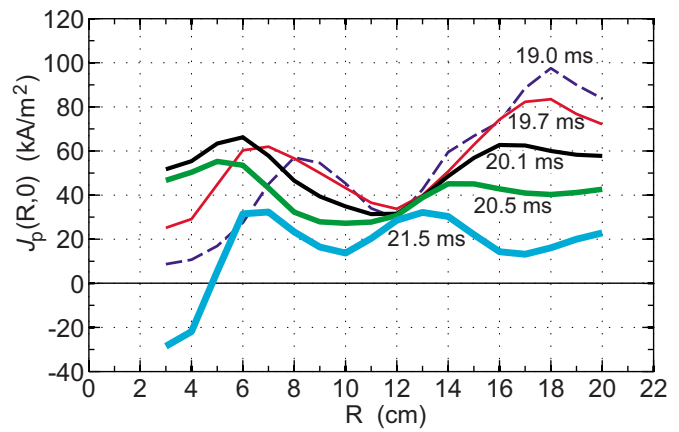


FIG. 19. (Color online) Profiles of the plasma current density along $Z=0$ chord, when the current in the middle coil is just started ($t=19$ ms), reaches disruptive level of 600 A ($t=19.7$ ms), and then further grows to 1080 A ($t=22$ ms) when the total plasma current is reduced.

seen if we consider how the position of the magnetic axis (the point along the $Z=0$ chord, where B_z becomes null) is changed. The time history of the magnetic axis position during whole plasma shot is shown in Fig. 15. It is seen that after the middle coil field is started, the axis is pushed inward by almost 2 cm in the ST regime. In the FRC regime, it is shifted by only 1 cm, with a delay of 3–4 ms.

Figure 16 shows the change of the magnetic axis position as a function of the middle coil field (B_m). The nonlinear relationship between the two values is because the middle coil not only changes the value of $\mathbf{B}$ in the $\mathbf{J}_p \times \mathbf{B}$ force acting on plasma, but it also affects the profile of $\mathbf{J}_p$; besides, as the magnetic configuration is changed, the RMF antenna coupling to plasma is improved, resulting in a larger plasma current. The difference between the FRC and ST regimes seen in Fig. 16 can be attributed to a different change of the plasma current profile under effect of B_m .

B. Large current in the middle coil: $I_{m,peak}=1080$ A

If the peak current in the middle coil is set to 1080 A (capacitor is charged to 15 V), the plasma current is disrupted after a brief initial growth during $t=19$ –20 ms (when $|I_m|$ is still below the disruptive level), as shown in Fig. 17. The most distinctive differences from the FRC regime are that the plasma current doesn't drop to zero during the "disruptive phase," and it always returns to a stable original configuration after $|I_m|$ relaxes to a level below 600 A (compare to Fig. 7).

It is also interesting that in the present case, the application of the middle coil field causes a significant extension of the plasma current profile along the Z axis that results in formation of a stable doublet structure, as displayed in Fig. 18. At $t=20.5$ ms, the value of poloidal flux at the chamber wall grows to 0.4 mWb at $Z=0$, which is 0.1 mWb above the vacuum value; at $Z=\pm 10$ cm it grows to 0.1 mWb.

The profiles of plasma current density shown in Fig. 19 confirm that even for a large value of $|I_m|$, the plasma current does not tend to become reversed, in contrast to the FRC regime (compare to Fig. 9). During $t=19.0$ –20.5 ms

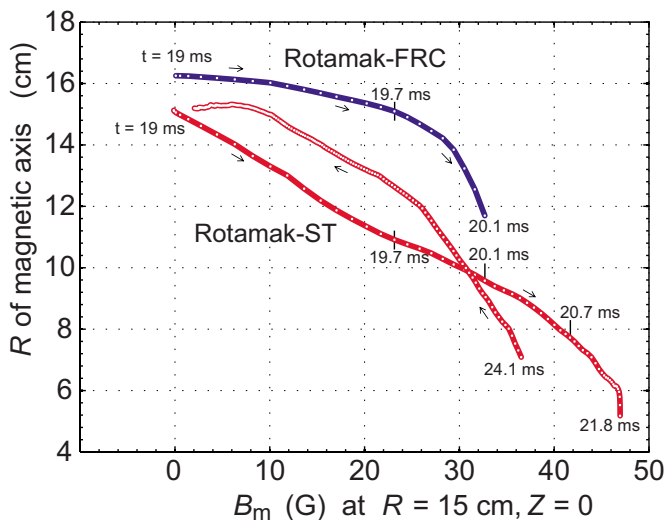


FIG. 20. (Color online) The magnetic axis position as a function of the middle-coil field magnitude.

the plasma is pushed inward. A sudden drop of J_p at $t=21.5$ ms near the chamber center suggests that the plasma is being pushed against the center column, resulting in disruption.

The change of the magnetic axis position under effect of the middle coil field is shown in Fig. 20. At $t \approx 19.7$ ms the current in the middle coil reaches 600 A with plasma current more than doubled from the original level, whereupon the plasma current quickly drops. In the FRC case, at $t \approx 20.1$ ms it drops to such a small value that the reversed magnetic configuration can no longer exist. However, in the ST case the plasma current remains sufficiently large to maintain a field reversed configuration until $t \approx 21.8$ ms; the reversed configuration reappears at $t \approx 24.1$ ms. Remarkably, in spite of significant variation of the plasma current, the change of magnetic axis position remains an almost linear function of B_m for up to 25 G in the FRC case and up to 40 G in the ST case (measured at $R=15$ cm, $Z=0$).

V. SUMMARY

The field of the magnetic coil placed at the midplane of RMF-driven plasma can be used to effectively control the profile of plasma current and the structure of magnetic configuration in the plasma. The response of the plasma to a brief use of the middle coil depends strongly on the initial conditions; in particular, on the presence of the toroidal magnetic field.

If the toroidal field is absent (Rotamak-FRC regime), the plasma appears to be too sensitive to the influence of the middle coil field. If the current in the middle coil reaches a certain threshold level, the plasma current is immediately

disrupted and drops to nearly zero. The cease of FRC configuration is preceded by development of a negative current density in the region $R=6-13$ cm, just before the disruptive level is reached.

On the other hand, if the toroidal field is present (Rotamak-ST regime), the plasma responds to the middle coil field in a more controllable manner. Even when the middle coil field is slightly above the disruptive level, the plasma current does not drop to zero, but only reduces to a smaller value. There is no trend of generating a negative current density in the core region of the plasma. Instead, the plasma current splits into two distinctive current rings. After the current in the middle coil is relaxed to a magnitude below the disruptive level, the plasma current always returns to its original structure and magnitude.

The experimental results imply that an efficient control of the shape and position of RMF-driven plasma is feasible, particularly for the ST regime.

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Comparison of rotamak plasma discharges in cylindrical and spherical devices

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Abstract

A new cylindrical chamber rotamak with one magnetic coil added in the midplane was constructed, and a series of experiments were performed at this device for comparison with the experiments previously conducted at a spherical chamber device. The experiments have been conducted in two regimes: the field-reversed configuration (FRC) and the spherical tokamak (ST) configuration when a steady toroidal magnetic field is added. The results from the new device further confirm that within a certain range of the equilibrium magnetic field B_v , the plasma current driven by a rotating magnetic field (RMF) grows linearly with B_v in both FRC and ST rotamak discharges. The ST configuration is more stable and allows one to achieve higher plasma current I_p with a proper choice of the toroidal field (TF); plasma current is found to reach its maximum at an optimum value of the applied TF. The measured plasma current, electron density and temperature in the cylindrical device are lower than those measured in the spherical device. The quadrupole structure of the self-generated toroidal magnetic field in the FRC regime is observed in both devices but with different magnitudes. In the cylindrical device, when a current is applied in the middle coil, plasma current can be enhanced up to 250% depending on the initial conditions.

(Some figures in this article are in colour only in the electronic version)

1. Introduction

A rotamak is a compact torus configuration having the distinctive feature that the toroidal plasma current is driven in a steady state, noninductive fashion by means of an externally applied rotating magnetic field (RMF) [1, 2]. The RMF is produced with two pairs of coils fed by two 90° -dephased rf generators. With this method, a sufficiently large plasma current can be generated to reverse the external equilibrium magnetic field on the symmetry axis so that a field-reversed configuration (FRC) is formed. Since 1980 extensive RMF driving FRC experiments have been done in the devices with different chamber shapes and volumes, with

different frequencies and output powers of the rf generators [3–10]. The method of driving the noninductive plasma current with a RMF has proven to be successful.

In recent years one of the important modifications in rotamaks is the addition of a steady toroidal magnetic field produced by an axial I_z current [4, 6]. With the steady toroidal field (TF), the rotamak is able to operate as a typical spherical tokamak (ST), with the plasma current sustained by a RMF rather than inductively. The Plasma Physics Laboratory at Prairie View A&M University has conducted a series of experiments on RMF-driven FRC, without and with TF, including the comparison of rotamak plasmas in FRC and ST configurations, the study of a self-generated toroidal magnetic field and high harmonic oscillations of a RMF [11–13].

The information about the effects of chamber shape, RF power density and external magnetic field on the plasma characteristics of RMF-driven FRCs is of crucial importance in the direction of scaling studies for future large devices. However there is no comparison study for experiments conducted in RMF-driven FRC plasmas with different chamber sizes and shapes by using the same rf frequency and input power. In order to study these effects, we built a cylindrical chamber rotamak device. The same RMF generators are used as on the existing spherical chamber rotamak. The distinctive feature of the new device is the additional magnetic coil in the midplane placed in close proximity to the chamber side wall. The coil is designed so that its magnetic field could be briefly turned on during the plasma shot. The advantage of this magnetic field coil is that by proper increase of the current in the coil, the plasma can be reshaped and pushed away from the chamber's wall.

This paper compares the experiments conducted at the new cylindrical chamber device and the existing spherical device for both FRC and ST regimes, with emphasis on the effects of the chamber shape and magnetic field of the middle coil on the plasma characteristics. The paper is organized as follows. Section 2 gives a brief description of the new cylindrical rotamak device in comparison with the existing spherical rotamak. The dependence of driven plasma current on the magnitude of the applied vertical and TFs is discussed in section 3. The measured radial profiles of electron density, temperature and pressure are given and compared in section 4. The results of measurements of the self-generated toroidal magnetic field in FRC discharges are presented in section 5. The effect of the middle magnetic field coil on plasma current in the cylindrical chamber rotamak is discussed in section 6, followed by the summary in section 7.

2. Experimental setup

The main components of our spherical chamber device and the newly built cylindrical chamber device are shown in figures 1(a) and (b), respectively. The spherical chamber is a 60 l (i.d. = 51 cm) Pyrex vessel, having 15 small diagnostic ports spaced along the poloidal direction (z -axis). The newly built cylindrical chamber is a 100 l (i.d. = 40 cm, L = 80 cm) Pyrex vessel, having six 10 cm i.d. ports, two at the top, two at the bottom, and two in the front for diagnostics. The background vacuum at the level of 1.0×10^{-6} Torr can be maintained in both chambers.

The plasma current is driven by a RMF which is produced by two pairs of coils. Each pair is connected through a system of matching/tuning capacitors and inductances to a separate 500 kHz/400 kW rf generator, with a 90° phase shift between the two generators. The waveform is programed in such a way as to maintain steady plasma current for a period of 40 ms.

The plasma current generated by the RMF is kept in equilibrium by a 'vertical' magnetic field that is produced by two coils connected to a 20 kW dc power supply. The coils can provide a field up to 230 G in the cylindrical vessel center; while the coils for the spherical rotamak can provide 130 G in the spherical vessel center.

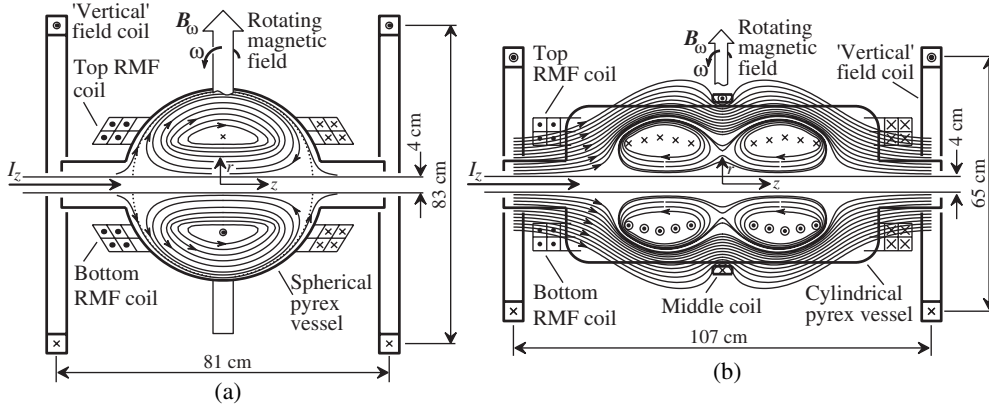


Figure 1. The schematic of the rotamak with (a) the spherical chamber and (b) the cylindrical chamber.

The toroidal magnetic field system in both chambers is made of a set of six wires fed through the central axial stainless steel pipe with o.d. = 4 cm, which is covered with a quartz tube. The steady toroidal magnetic field B_ϕ is produced by discharging a capacitor bank (maximum 152 kJ) through the wires.

In experiments, the ‘vertical’ field B_v is started about 250 ms before applying the RMF, while the TF B_ϕ (for ST configuration) is started about 10 ms before applying the RMF. During the 40 ms period when the RMF is applied, both B_v and B_ϕ remain constant within 5%.

During the experiments, hydrogen gas flows continuously into the vessel at a pressure of p_f around 1.4 mTorr. The plasma is started by using a pair of multi-turn pre-ionization coils, which are wound around side ports of the discharge vessel at $z = \pm 40$ cm and fed by a 650 W/13.6 MHz rf generator. Then the RMF generators are triggered, and the pre-ionization generator is turned off after the 5th ms, so that the plasma is sustained with the RMF alone for the rest of a shot. The plasma discharges in the two devices are very stable and reproducible from shot to shot.

Plasma temperature and density are measured with double floating Langmuir probes. Miniature magnetic pickup coils and Hall probes are used to measure magnetic fields in plasmas in different directions. In the cylindrical chamber rotamak, a set of L-shape and straight Langmuir and magnetic pickup probes are used, which can reach almost any point in the poloidal cross-section. To measure the driven plasma current I_p , a Rogowski coil is pulled through the central steel pipe around a poloidal cross-section. Two other smaller Rogowski coils are used for measuring the axial current I_z and the current through the ‘vertical’ field coils.

3. Dependences of driven plasma current on vertical and toroidal magnetic fields

The initial experiments conducted at the new cylindrical chamber rotamak were focused on finding optimized parameters for stable and repeatable rotamak operation, and comparing these parameters with those for the spherical rotamak. In both devices, the gas filling pressure was $p_f = 1.4$ mTorr; the rf power was 200 kW. The experiments have been performed for two regimes: FRC without external TF, and ST when the TF is added.

The plasma current driven by the RMF depends on the gas filling pressure, the ‘vertical’ field magnitude and the rf power. Besides, in the ST regime, the addition of the TF introduces another parameter.

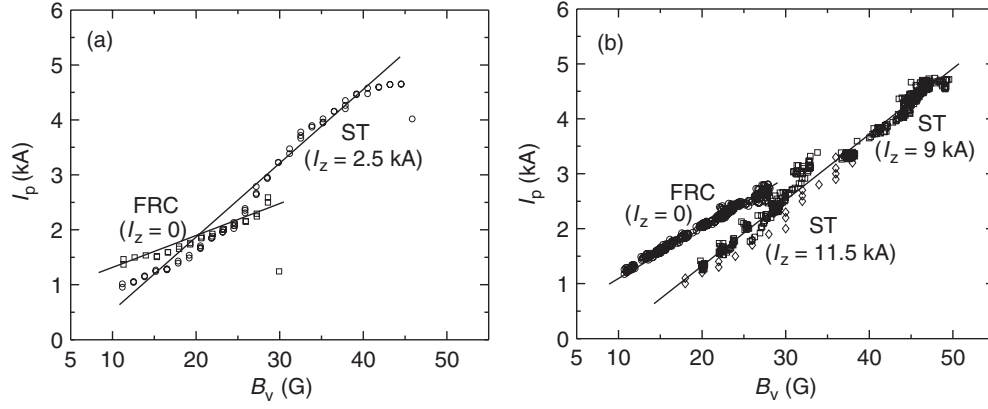


Figure 2. The dependence of the plasma current on the vertical magnetic field for (a) the cylindrical chamber and the (b) the spherical chamber rotamak discharges.

A series of measurements were made for different values of B_v in both devices. Corresponding values of (B_v, I_p) , measured at $t = 20$ ms when the discharge is at a very stable stage, are shown in figure 2. In the cylindrical rotamak FRC, for values of vertical field B_v below 28 G, the plasma current grows linearly with B_v , as shown in figure 2(a). For B_v greater than 28 G, the value of I_p is seen to drop abruptly. This problem of current termination has been discussed in a number of papers [14–16]; it is related to the way the rf power is coupled to the plasma. In the spherical rotamak FRC the dependence of I_p on B_v is the same, as shown in figure 2(b), and the maximum value of B_v before termination of the plasma current is also 28 G. These results are in agreement with earlier experiments [14–16]. Results from both devices confirm that in the FRC configuration, the stable plasma current cannot exceed 3 kA for the specified level of total rf power 200 kW.

The addition of a steady toroidal magnetic field allows one to improve the situation in both cylindrical and spherical devices. As seen in figure 2, when the magnitude of B_v is below a certain value, the presence of the TF (axial current I_z) is disadvantageous for the value of the RMF-driven plasma current. However, when B_v exceeds this critical value, the application of the TF results in the generation of larger plasma current and enables its further growth with the increase in B_v . For both chamber configurations, the value of plasma current grows almost linearly with B_v , and the rate of growth is higher in the ST regime than in the FRC. Using the linear fit to the data, we obtain that the slope in FRC discharges is 86 AG^{-1} for the spherical rotamak and 80 AG^{-1} for the cylindrical rotamak; while in ST discharges it reaches 118 AG^{-1} for the spherical rotamak and 129 AG^{-1} for the cylindrical rotamak. In the earlier Flinders' rotamak-FRC plasmas a value of 92 AG^{-1} has been reported [15]. It is worth mentioning that the Solov'ev equilibrium model [17] predicts this ratio to be: $I_p/B_v = 5a/\mu_0 = 99 \text{ AG}^{-1}$ for the spherical devices and 79 AG^{-1} for the cylindrical device.

It can be concluded from figure 2 that in FRC discharges the driven plasma current has a termination point (28 G for both the cylindrical and spherical chamber rotamaks), above which the plasma becomes unstable; the addition of the TF postpones the occurrence of the plasma current termination to a higher one, 45 G for the cylindrical rotamak and 48 G for the spherical rotamak.

For a deeper understanding of the role of the TF in ST discharges, a series of experiments were conducted as the vertical field was fixed, while the TF, i.e. the axial current I_z , was varied.

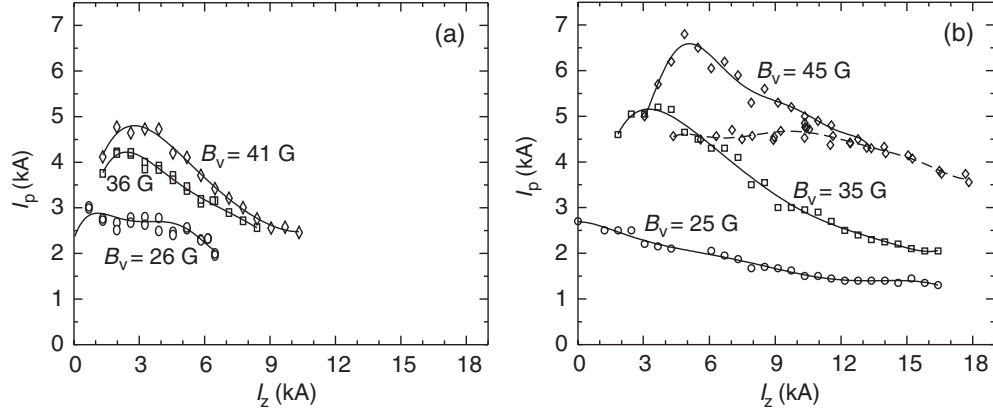


Figure 3. Dependence of plasma current for ST discharge on the axial current for a fixed magnitude of the vertical field in (a) cylindrical chamber and (b) spherical chamber.

The maximum deliverable rf power was unchanged at 200 kW, and gas filling pressure was maintained at 1.4 mTorr. As shown in figure 3, for a given vertical field the stable plasma can be obtained only in a certain range of the axial current values. The different range of I_z between two rotamaks might relate to the chamber configuration and the different power density for a given rf power level; the cylindrical rotamak has almost twice as large a volume than that of the spherical rotamak.

It is also seen in figure 3 that there exists an optimal value of the axial current for the largest plasma current, and this optimal I_z is different for each value of B_v . It occurs that as the vertical field is increased, the peak-shape dependence of the plasma current on the axial current becomes gradually sharper. It is interesting to note that in the vicinity of the peak, the magnitude of the driven current becomes increasingly sensitive to other conditions, such as the gas filling pressure and rf-plasma tuning/matching, which is demonstrated by two branches for $B_v = 45$ G; a slight variation (less than 10%) in the tuning capacitors or filling pressure can result in the change of the driven plasma current by about 30%.

The above general behavior agrees qualitatively with the predictions of the RMF current drive theory in which the presence of a TF has been taken into account [18]. This theory shows that the addition of a TF improves the penetration of the RMF into the plasma, due to excitation of the rotating whistler mode. On the other hand, as the magnetic field is increased, the relative phase between the oscillating $\mathbf{B}_r$ field and $\mathbf{J}_z$ current can change in such a way that there is a decrease in the time averaged force which drives the current, and therefore the plasma current decreases. Quite a similar result has been observed in a smaller (10 L) rotamak-ST [3]. Thus our results suggest that the shown dependence of plasma current on the TF is universal for the rotamak-ST plasmas.

4. Profiles of electron temperature and density in FRC and ST discharges

The electron temperature and density were measured with a double floating Langmuir probe. All the radial profiles discussed in this section were measured at the midplane ($z = 0$) of the chamber using an L-shape probe for the cylindrical rotamak, and a straight probe for the spherical rotamak.

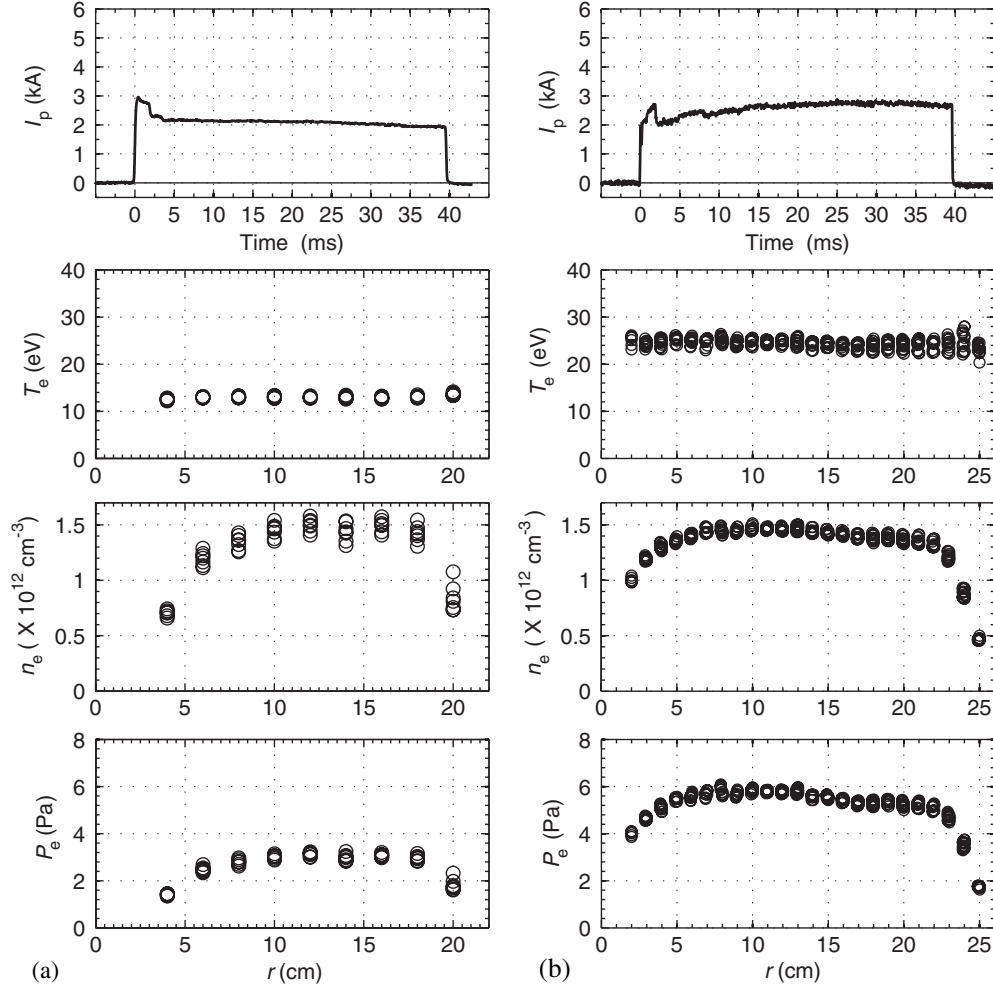


Figure 4. Time history of plasma current and radial profiles of plasma parameters for typical FRC discharges in (a) the cylindrical chamber and (b) the spherical chamber.

4.1. Profiles in FRC plasmas

During each shot, the temperature and density do not show any significant variation in both the spherical and cylindrical chambers (if the middle coil is not used). This is a consequence of a very steady course of plasma current; it is displayed in figure 4. The plasma current is typically 2.2 kA for the cylindrical chamber plasma, and 2.7 kA for the spherical chamber plasma. The electron temperature profile is very flat in both cases, but the value is different, around 12.5 eV for the cylindrical chamber plasma and 25 eV for the spherical chamber plasma. In both devices the electron density profile is not only flat in the greater part of the plasma, but also the value is nearly the same, about $1.5 \times 10^{18} \text{ m}^{-3}$. As seen from figure 4, in the cylindrical rotamak the value of electron pressure (P_e) calculated from the temperature and density is only half of the P_e in the spherical rotamak because of its smaller by half T_e . But the profile of P_e in the cylindrical rotamak is almost uniform in the greater part of the plasma, while the profile of P_e in the spherical rotamak is somewhat higher in the inner area of the

plasma volume. The same observation as in our spherical rotamak has been reported for the cylindrical-shape RMF-driven FRC plasmas [10].

We note that in the FRC regimes, the vertical field, filling gas pressure and rf power are kept the same in both rotamaks, but rf power density in the cylindrical chamber is about half of that in the spherical chamber because of different chamber volumes; this is apparently the reason for lower T_e and P_e in the cylindrical rotamak.

4.2. Profiles in ST plasmas

As discussed in section 3, rotamaks with different chamber shapes have different ranges of I_z and B_v in the ST configuration for stable plasma discharges. So in the ST regime, for profile scanning, only the filling gas pressure and total rf power are kept the same in both rotamaks.

To compare with the ST discharges in the spherical chamber, the operation parameters for repeatable ST discharges in the cylindrical chamber have been chosen to be $B_v = 36$ G and $I_z = 3.1$ kA, resulting in the plasma current typically around 4.1 kA as shown in figure 5(a). A larger plasma current (4.6 kA) can be achieved in the cylindrical chamber rotamak if the vertical field B_v is increased to 45 G, but the discharges frequently result in disruption. In the spherical chamber rotamak, stable discharges with $B_v = 45$ G and $I_z = 9$ kA, resulting in plasma current of 5 kA can be easily achieved, as displayed in figure 5(b).

The measured profiles of electron temperature and density, and the calculated profile of electron pressure are shown in figure 5. The radial profiles of plasma parameters in both rotamaks display almost the same structure: T_e displays almost a flat profile, while the profile of n_e becomes triangular-shaped, with the peak value being close to that in the FRC regime.

A similar modification in the density profiles, i.e. becoming more peaked as the TF is increased, has been observed in a smaller spherically shaped rotamak-ST [7]. The strong peaking is attributed to the diamagnetic poloidal current and the associated $\mathbf{J}_{\text{pol}} \times \mathbf{B}_{\text{tor}}$ force directed towards the magnetic axis. The density at the separatrix radius is about 37% lower than the peak value, which indicates the improvement of particle confinement compared with the FRC case.

5. Observation of a self-generated toroidal magnetic field in FRC discharges

In FRC rotamak experiments with a spherical chamber, a steady toroidal magnetic field has been observed [19, 20], which is generated by poloidal electron flows (no external TF was applied). Such a self-generated TF always has a bi-directional structure—opposite polarity in the two hemispheres, corresponding to two swirls of current in the poloidal cross-section. In the spherical rotamak-FRC experiments, we also observed that the self-generated TF can flip its sign in the radial direction so that its topology corresponds to four swirls of current in the poloidal cross-section [12]. Besides, in some cases we observed a reversal of polarity of the self-generated TF during the shots [12].

In the cylindrical chamber, the measurement of the self-generated TF was performed with magnetic pickup coils; the signal has been filtered and numerically integrated. Measurements have been checked with the Hall probe previously used in the spherical chamber. The conditions of experiments were the same as described in sections 3 and 4.

The main results are shown in figure 6(a), and compared with the measurements performed in the spherical chamber, which are displayed in figure 6(b). The plots show the radial profiles of the quasi-steady self-generated TF measured in two opposite ports (hemispheres). The profiles for other values of z are similar, with almost the same or a little bit smaller magnitude of B_ϕ . The positive value of B_ϕ corresponds to the direction against the plasma current.

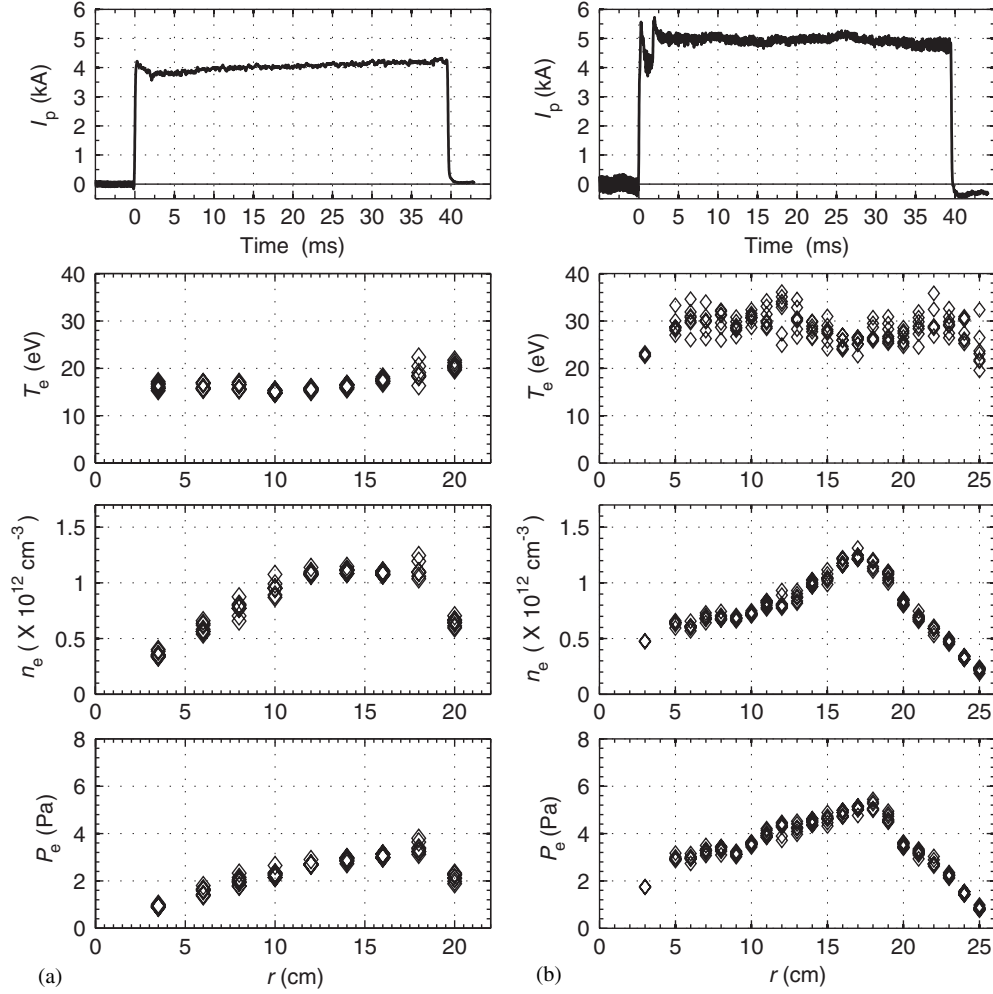


Figure 5. Time history of the plasma current and radial profiles of the plasma parameters for typical ST discharges in (a) the cylindrical chamber and (b) the spherical chamber.

In the spherical chamber device, the structure and magnitude of the TF was almost unchanged during shot (although in another series of experiments it was reversing its sign in two hemispheres [12]). The magnitude was seen to reach 13 G, which is about 60% of the magnitude of the rotating field ($B_{\omega,\phi}(a) \approx 20$ G) at the plasma edge, or 30% of the poloidal field (45 G) generated by the toroidal plasma current.

In the cylindrical chamber, the self-generated TF could reach a large magnitude of 10 G only during the initial 2–3 ms transitional stage; at that time it has a typical dual-vortex bi-direction structure similar to one observed earlier at Flinders University [20]. However this configuration is not sustained during the steady stage of plasma discharge. As seen from figure 6(a) at $t = 10$ ms, the TF, while remaining opposite in the two hemispheres, also becomes reversed in the outer area of the plasma. During this stage of discharge (lasting 37 ms), the TF remains almost unchanged at a low level of 3 G, which is about only 5% of the poloidal field (55 G) generated by the toroidal plasma current. This value is in agreement with

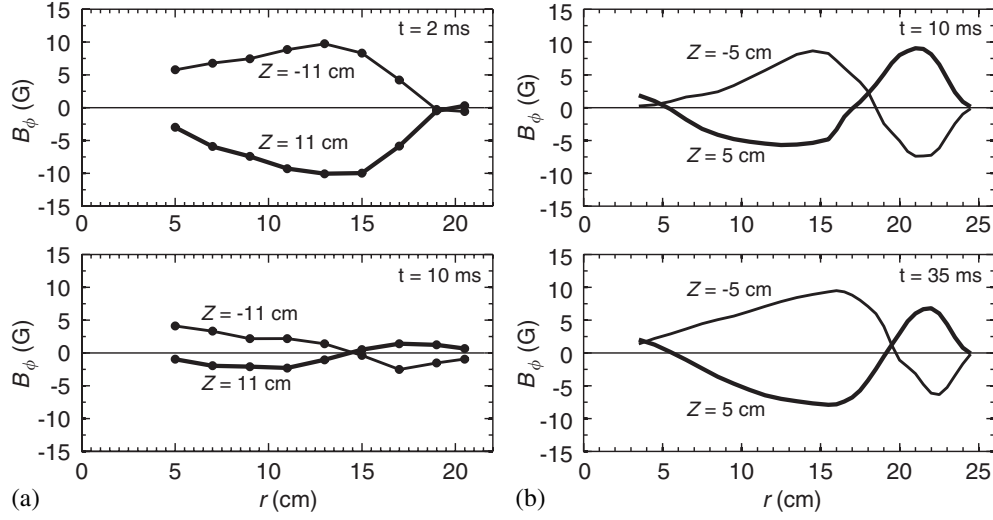


Figure 6. Radial profiles of the self-generated toroidal magnetic field (a) in the cylindrical chamber rotamak and (b) in the spherical chamber rotamak.

earlier experiments in smaller devices, where the self-generated TF was about 8% of the field generated by the toroidal plasma current [20].

6. Effect of the middle coil on plasma profiles in the cylindrical chamber rotamak

The additional middle magnetic coil is made of eight wires (two layers by four turns) wound close to the chamber side surface. The current in the coil is produced by discharging a capacitor. In the experiments discussed in this section, the electronic switch that connects the capacitor to the coil is triggered at $t \approx 19$ ms of the discharge. The current reaches its maximum at approximately 21st–22nd ms, and then gradually reduces to zero within 10 ms. The value of peak current in the coil is adjusted by the capacitor charging voltage. For both FRC and ST experiments discussed in this section, the vertical field was 25 G at the chamber center, and the rf power was about 200 kW; in the ST case, the axial current that produces the TF was 2.5 kA.

The results for three different charging voltages, 5, 15 and 30 V, corresponding to the total ($I_m = i_{\text{feed}} \times 8$) peak current in the middle coil equal to 260, 1080 and 2260 A, are displayed in figure 7. When the current in the middle coil reaches its peak value of 263 A, as seen from figure 7, top row, the plasma current responds with a growth to the level of 3.6 kA in the FRC case and 3.8 kA in the ST regime; the increase being a few times larger than the magnitude of the current in the middle coil. However, when the current in the middle coil exceeds a threshold level of 600 A (we call it ‘disruptive’ level), the plasma current is disrupted and suddenly drops, as shown in the middle row of figure 7. When $|I_{m,\text{peak}}| = 1080$ A (charging voltage = 15 V), the plasma current suddenly drops after a brief initial growth during $t = 19$ – 20 ms (when the current in the middle coil is still below the disruptive level). It is interesting to notice that in the FRC regime the plasma current is completely disintegrated (in fact, a slightly negative current is somehow generated); there are no closed magnetic surfaces in this case; but in the ST regime the plasma current drops to a ‘normal’ level of 2 kA. Moreover, after the current in the middle coil relaxes below the disruptive level (after $t \approx 25$ ms), the plasma current in the ST regime always returns to a higher level, as if there were no disruption. In contrast, in the

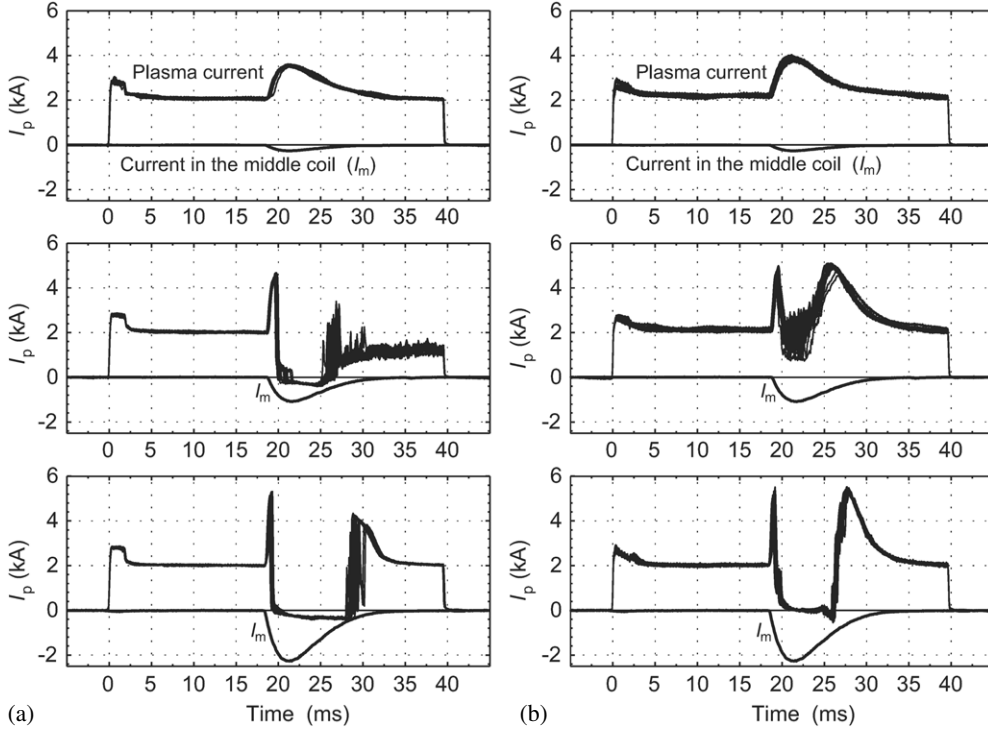


Figure 7. Time history of the plasma current when different currents of the middle coil are applied to (a) FRC and (b) ST rotamak plasma discharges. The data from 15 shots are plotted in each part. The current in the middle coil is also shown; the negative sign corresponds to the direction opposite to the plasma current.

FRC regime the plasma almost never returns to the steady-state configuration of $t = 3\text{--}18$ ms that existed before the current in the middle coil was applied.

It is worth mentioning that we could never find such a value of peak current in the middle coil that the plasma current in the FRC regime is only slightly ‘disturbed’, like it was in the case of ST shown in figure 7(b), middle row. In the FRC regime, as soon as the magnitude of I_m reaches the disruptive level, the plasma current drops abruptly to a negative value. This fact is another indication for the enhanced stability of the ST configuration.

For larger values of current in the middle coil, a repeatable discharge condition can be achieved for the ST regime when the plasma current is disrupted and drops to zero, as shown in the bottom plot of figure 7(b). Still, when the current in the middle coil is relaxed to a magnitude below the disruptive level, the plasma current emerges again at a high value; noteworthy, there is no pre-ionization at this stage (it is turned off at $t = 5$ ms). Such stability is very persistent for the ST regime, in contrast to the FRC regime, where only in some shots does the plasma current returns to the 2 kA level.

From the analysis of the magnetic field structure we conclude that in the ST case the plasma splits into two parts and at the same time is pushed against the center column that eventually results in disruption. In the FRC case, only a small 1–2 cm inward movement of the magnetic axis is observed when suddenly the plasma current drops to zero. Presumably, instabilities develop due to the generation of negative current in the plasma core.

When the current in the middle coil is below the ‘disruptive’ level, the response of the plasma current to the peak value of applied current in the middle coil is summarized in figure 8.

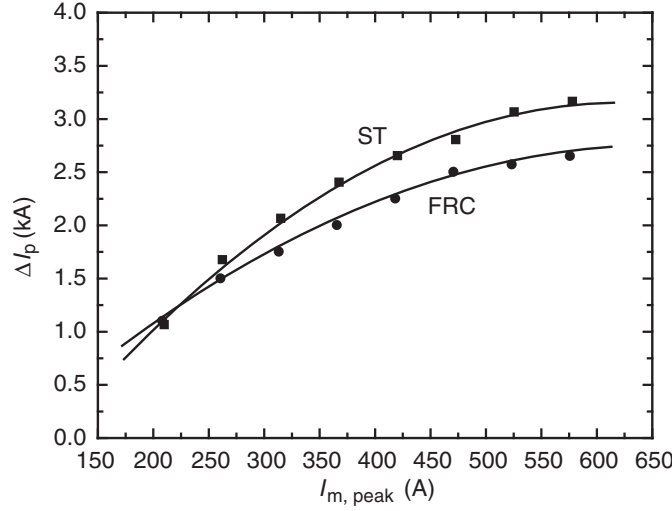


Figure 8. The increase in plasma current as a function of the applied peak current in the middle coil.

For both FRC and ST regimes, $\Delta I_p (= I_{p, \text{peak}} - I_p(15 \text{ ms}))$ grows up monotonously with the increase in peak current in the middle coil; the increase in plasma current is usually higher in the ST regime than in the FRC regime. Depending on the initial conditions, plasma current can be enhanced by up to 3.1 kA; the peak plasma current is about 250% of the original plasma current. When the applied peak current in the middle coil reaches the value of 600 A, the plasma current starts to be disrupted.

Based on experimental data for the FRC regime, we can estimate geometrical parameters for plasma shape before and after the middle coil was energized. At $t = 15 \text{ ms}$ the magnetic axis position is $R_0 = 16.3 \text{ cm}$, the minor radius is $a = 6.6 \text{ cm}$ and the elongation is $\kappa = 1.9$. After the middle coil is energized, at $t = 22 \text{ ms}$, these parameters change to $R_0 = 15.4 \text{ cm}$, $a = 6.9 \text{ cm}$, $\kappa = 2.5$. Using model equations for the plasma self-inductance [21] we can estimate: $L_s(t = 15 \text{ ms}) = 0.123 \mu\text{H}$ and $L_s(t = 22 \text{ ms}) = 0.083 \mu\text{H}$, which is a 33% drop. On the other hand, the plasma current has increased from 2.07 to 3.55 kA. Then, the relative change in $(L_s \times I_p)$ flux value is positive 15%.

Similarly, in the ST case, we have $R_0 = 15.2 \text{ cm}$, $a = 7.2 \text{ cm}$, $\kappa = 2.0$ at $t = 15 \text{ ms}$, and $R_0 = 14.6 \text{ cm}$, $a = 6.9 \text{ cm}$, $\kappa = 2.6$ at $t = 22 \text{ ms}$. Accordingly the plasma inductance has changed from 0.085 to $0.069 \mu\text{H}$, i.e. decreased by 19%. However, the flux $(L_s \times I_p)$ has increased by 39%.

Therefore, the increase in plasma current is partially due to compression of plasma, during which the $L_s \times I_p$ flux is conserved. Besides, an ‘extra’ current is generated because as the plasma is extended in the z direction, the coupling between RMF antenna coils (which stretches along the whole length of the chamber) and plasma is improved. This is confirmed by a 15–20% increase in the current flowing through the RMF coils when the middle coil is energized.

A further insight into the stability of the ST configuration comes from measurements of magnetic field components in the poloidal cross-section of the plasma. A two-component magnetic pickup coil was used to perform the radial scans at axial positions $|z| = 0, 50, 110, 170$ and 225 mm . The coil could be retracted through the sealed L-shape glass tubing, which itself could be moved in the radial direction. After the 2D scan is completed, the field

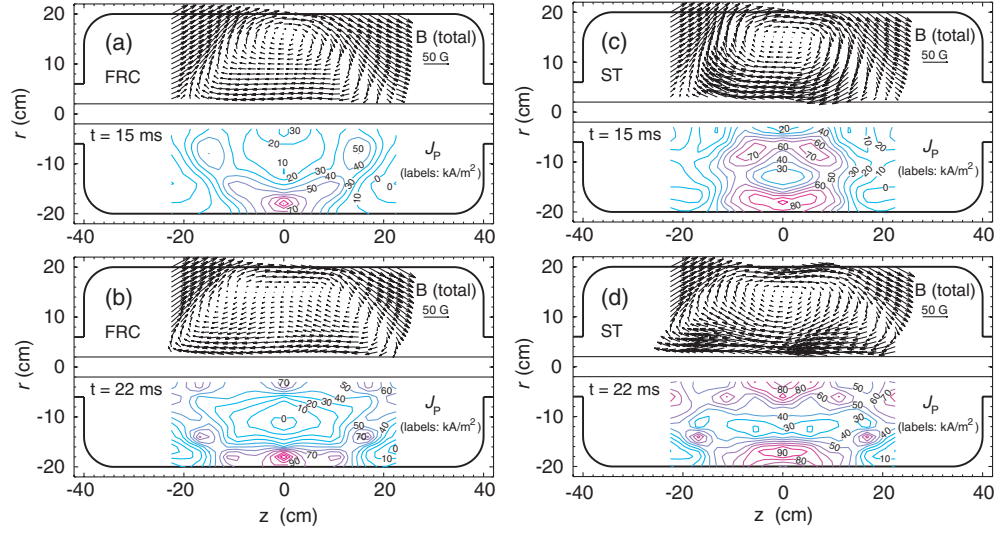


Figure 9. Measured magnetic field and calculated plasma current density in (a), (b) the FRC regime, and (c), (d) the ST regime, when the current in the middle coil is zero, and when it reaches 260 A. The magnetic field includes the field from plasma current, equilibrium field and the field from the middle coil (when it is present at $t = 22$ ms).

components are interpolated to intermediate radial and axial points, and the current density profiles are calculated from $\mathbf{J} = \nabla \times \mathbf{B} / \mu_0$. The results are presented in figure 9.

The first observation from the figure is that the plasma current in the ST regime is more evenly distributed along magnetic surfaces. In the FRC regime it is mostly localized at the midplane ($z = 0$), close to the chamber's wall. When the current in the middle coil is applied and reaches its peak at $t \approx 22$ ms, the profile of the plasma current becomes extended in the axial direction. In the ST regime, the plasma current is also pushed away from the chamber's walls, while in the FRC regime it is not.

The effect of the middle coil is also shown in the plots of plasma current radial profiles for the $z = 0$ chord given in figure 10. It is seen from figure 10 that in the FRC regime when the current is applied in the middle coil, the plasma current is reduced in the region $r = 7$ –16 cm and enhanced at $r < 6$ cm and $r > 17$ cm; in fact, the current density near the wall becomes larger by about 30%, which probably causes a release of impurities and hydrogen. In the ST regime the application of the middle coil field results in moving the plasma current away from the chamber's wall and in an increase in current density at $r < 18$ cm.

Another interesting observation from figure 9 is that the closed magnetic surfaces are quite short along the z -axis, in spite of the fact that the RMF coil extends to the end of the chamber. This result is confirmed by an electron density scan with a Langmuir probe for ST discharge shown in figure 11 (the temperature profile is not shown; it is almost uniform within the scanned area). It is seen that the plasma is located mainly within the area of $|z| < 15$ cm, the position of maximum density being at $z = 0$ cm/ $r = 15$ cm. The nearly spherical shape of the plasma might be the consequence of two factors: the high mirror ratio of the equilibrium 'vertical' field, and the pondermotive force originating from the inhomogeneous oscillating electromagnetic field that pushes plasma towards the midplane [22]. As a result, the cylindrical chamber rotamak is less efficient for RMF current drive than the spherical device, but the situation can be improved if a multi-coil equilibrium field system is added.

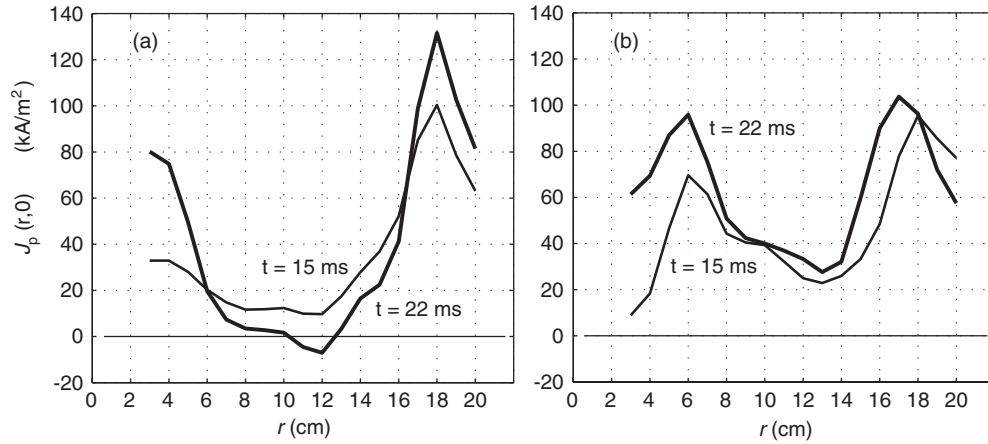


Figure 10. Plasma current profiles along the $z = 0$ chord in (a) the FRC regime and (b) the ST regime, when the current in the middle coil is zero, and when it reaches 260 A (at 22 ms).

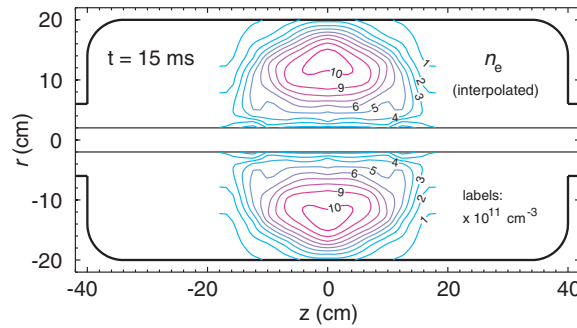


Figure 11. 2D distribution of electron density in the cylindrical chamber rotamak-ST discharge.

7. Summary

Steady plasmas have been produced and sustained in both cylindrical and spherical chamber rotamaks operated in the FRC and ST regimes. The comparison of experiments in two chamber configurations further confirms that the addition of the toroidal magnetic field can result in an essential improvement in the magnitude of the driven current; in the ST regime the RMF-driven plasma current can be doubled comparing with the FRC regime, for the same rf power level.

The scaling dependence of the plasma current on the magnitude of the equilibrium ‘vertical’ field is also advantageous for the ST regime. In the ST case, the growth rate of the driven current with the ‘vertical’ field is 129 AG^{-1} for the cylindrical rotamak and 118 AG^{-1} for the spherical rotamak compared, respectively, with 79 AG^{-1} and 99 AG^{-1} predicted by the Solov’ev model, while in the FRC case it is only 80 AG^{-1} for the cylindrical rotamak and 86 AG^{-1} for the spherical rotamak. For a given rf power level and gas filling pressure, both devices confirm that the value of the driven current strongly depends on the magnitude of the applied TF, with its optimum value different for each value of the vertical field.

Both devices have the same radial profile structures of electron temperature, density and pressure in the bulk plasma, but the magnitudes of temperature and pressure are lower in the cylindrical rotamak because of its half value of the rf power density for the same rf input

power. In ST cylindrical rotamak discharges, the triangle-shaped profiles of electron density and pressure become less sharp because of larger volume, smaller vertical and TFs compared with those values applied in the spherical rotamak.

The self-generated TF in the FRC regime has been measured in both devices. The quadrupole structure of the TF is observed in both devices; the TFs are not only opposite in polarity in the two hemispheres, but they also reverse their sign from the inner areas of the plasma to the edge. During the steady stage of discharge, the measured TF in the cylindrical rotamak is 3 G, which is about 5% of the poloidal field generated by the toroidal plasma current; while in the spherical rotamak, the measured TF is 13 G, which is about 30% of the poloidal field.

When current is applied to the middle coil in the cylindrical chamber rotamak, the plasma current can be enhanced from 2.1 up to 5.2 kA, depending on the initial conditions. If the current in the middle coil exceeds a level of 600 A, the plasma current is disrupted, but in the ST regime it always reappears at a higher value after the current in the middle coil is reduced. The detailed measurements of the magnetic field in the plasma show that the field of the middle coil can be effectively used to reshape the plasma and to push it away from the chamber walls, especially in the ST regime. This result suggests that an active feedback system can be used to control the equilibrium in RMF-driven plasmas.

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The effect of toroidal field on the rotating magnetic field current drive in rotamak plasmas*

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A rotamak is one kind of compact spherically shaped magnetic-confinement device. In a rotamak the plasma current is driven by means of rotating magnetic field (RMF). The driven current can reverse the original equilibrium field and generate a field-reversed-configuration. In a conventional rotamak, a toroidal field (TF) is not necessary for the RMF to drive plasma current, but it was found that the present of an additional TF can influence the RMF current drive. In this paper the effect of TF on the RMF current drive in a rotamak are investigated in some detail. The experimental results show that addition of TF increases the RMF driven current greatly and enhances the RMF penetration dramatically. Without TF, the RMF can only penetrate into plasma in the edge region. When a TF is added, the RMF can reach almost the whole plasma region. This is an optimal strength of toroidal magnetic field for getting maximum plasma current when B_v and radio frequency generator power are fixed. Besides driving current, the RMF generates high harmonic fields in rotamak plasma. The effect of TF on the harmonic field spectra are also reported.

Keywords: rotamak, rotating magnetic field, toroidal field

PACC: 5200, 5220, 5225

1. Introduction

A rotamak is one kind of unique device that can realize field-reversed-configuration (FRC) through current drive by rotating magnetic field (RMF).^[1] In a rotamak, the electrons are trapped and rotated synchronously with external applied RMF, the plasma current is produced while ions are regarded as stationary background. The principle of current drive by RMF in a rotamak has been extensively studied.^[2–7] For driving current effectively, RMF is chosen to satisfy $\omega_{ci} \ll \omega \ll \omega_{ce}$ and $\omega_{ce}/\gamma_{ei} > a/\delta$, where ω_{ci} and ω_{ce} are ion and electron cyclotron frequency around the RMF B_ω respectively, γ_{ei} is the electron-ion collision frequency and a is the plasma radius, $\delta = (2\eta/\mu_0\omega)^{1/2}$ is the classical penetration depth at the RMF frequency ω . Except its compact configuration, the current in rotamak is driven and maintained by RMF which is usually powered by radio frequency (RF) generators, it has another advantage of the possibility of operation in continuous mode. Owing to its unique feature, the rotamak has stimulated more and more interest in fusion plasma recently.

The current drive in rotamak and realizing FRC have been studied in experiments and theories since

the early 1980s and a lot of progress has been made.^[2–13] It is reported recently that if an additional toroidal field (TF) is added, the current drive is improved and the rotamak can run as a low aspect ratio spherical tokamak (ST).^[14] Up to now, what causes the TF to improve current drive is unknown, there are only several ambiguous theoretical papers.^[15,16] So further study about the influence of TF on the RMF current drive is necessary for understanding the reason of TF improving the RMF current drive. In this paper the RMF current drive with and without TF and the change of plasma behaviour are presented. Our experimental results show that the TF will improve RMF penetration and increase the plasma current greatly. TF also affects the harmonics fields stimulated by RMF. This paper is organized as follows. In Section 2 a brief introduction of the experimental device is given. In Section 3 the experimental results are presented. In the end there is a summary in Section 4.

2. Experimental setup

Figure 1 is the structure of the rotamak device. The rotamak chamber is a spherical vessel made from

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Pyrex with 50 cm diameter. The chamber is pumped by a 5001/S turbo pump and the background vacuum is around 1.333×10^{-4} Pa. The RMF is produced by passing RF currents, from two 400 kW/500 kHz RF generators, dephased by 90° , through two pairs of orthogonal Helmholtz coils located on the outside of the chamber. The plasma current generated by RMF is kept in equilibrium by vertical magnetic field which is produced by two vertical field coils with 83 cm diameter. The coils are connected in parallel and powered by a 20 kW DC power supply. The coils can produce a field up to 130 G in the vessel centre. In addition, by using an axial current (I_z in Fig.1), a TF can be added to the rotamak plasma. The system that produces TF consists of a set of six wires fed through the central axial stainless steel pipe with an external diameter of 4 cm. The steady toroidal magnetic field B_ϕ is produced by discharging a capacitor bank (maximum 152 kJ) through the wires. The discharge current threads the central axis six times, resulting in a total I_z current up to 18 kA, which can produce a TF $B_\phi = 288$ G at the half-radius of the chamber, $r = 12.5$ cm.

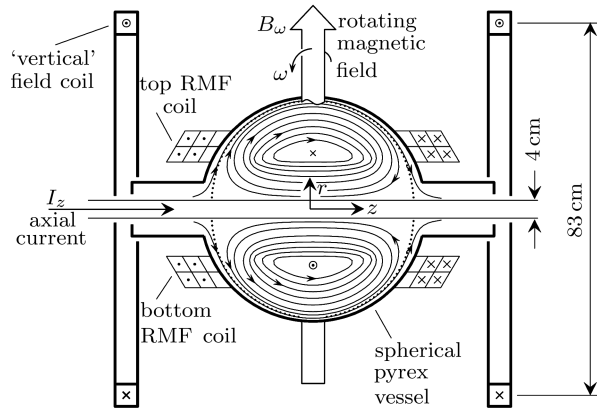


Fig.1. Schematic diagram of rotamak.

A Rogowski coil through the central tube and around a poloidal cross section is used to measure the driven plasma current. Align with the meridian plane of the chamber, there are 15 diagnostic ports with 6 mm inner diameter, which allow inserting probe into chamber for measurement. Miniature magnetic pickup coils and hall probe, which retractable through quartz guide tube sealed in diagnostic port, are used to measure the oscillating magnetic field and the steady magnetic field in the plasma respectively. A double floating Langmuir probe is used to measure the elec-

tron density and temperature of plasmas. Based on the shot to shot repeatable discharges the radial profiles of the parameters are got.

It is important to match the generator and plasma load for coupling the energy from the RF generators to plasma as large as possible. For this purpose there is a matching and tuning circuit in each RMF coil pair and there are RF voltage meters and current sensors for measurement. In air-core transformer model,^[17] each matching and tuning circuit can be simplified as Fig.2. Here L_c and R_c are inductance and resistance of RMF coils, ΔL is the change of inductance caused by plasma, R_p is the reflection resistance of plasma. V_{in} and I_{in} , V_{ld} and I_{ld} are the voltage meters and current sensors for monitoring the voltage and current input from generator and load on the RMF coils respectively. During experiments, by adjusting the value of matching inductance L_m , the capacitance of matching and tuning capacitors C_M , C_T , let the phase between the V_{in} and I_{in} , V_{LD} and I_{LD} become zero and then the maximum plasma current will be produced.

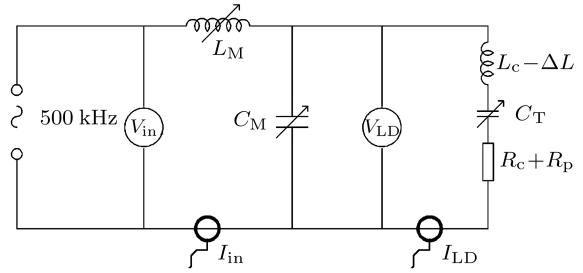


Fig.2. The matching and tuning circuit.

During experiments, hydrogen continuously flows through the chamber. The filling pressure is controlled by a mass flow controller. The discharges are very stable and reproducible from shot to shot when conditions keep same.

3. Experimental results

3.1. Effect of TF on the RMF current drive

Figure 3 is a typical time evolution waveform of plasma currents and the axial magnetic field B_z in $r = 3.5$ cm in the cases with and without TF. Table 1 is the typical parameters of the two cases. For comparison, in both cases the power from RF generators P_{rf} and the gas filling pressure P_f are kept unchanged.

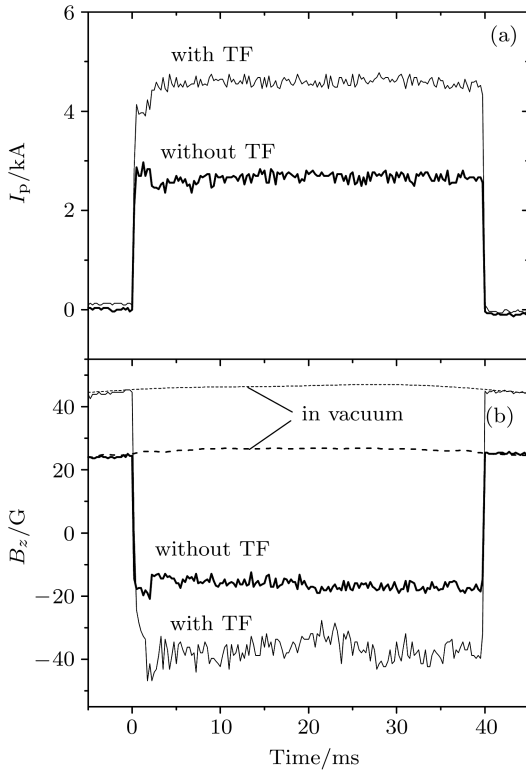


Fig.3. The typical time evolution of plasma currents and the axial magnetic field B_z in $r = 3.5$ cm in the cases with and without TF.

Table 1. The typical parameter for shot of Fig.3.

	without TF	with TF
P_f/Pa	0.18	0.18
P_{rf}/kW	2×180	2×180
B_v/G	25	45
I_z/kA	0	9.0
I_p/kA	2.7	4.7
n_e/cm^{-3}	1.4×10^{12}	1.2×10^{12}
T_e/eV	22	30
I_{RMF}/A	88	51
R_p/Ω	26	87

The power from each RF generator is 180 kW and the gas filling pressure is 0.18 Pa. In the case without TF, the biased vacuum vertical field B_v is 25 GS at the centre of chamber. In the case with TF, a 9 kA axial current is added and the vertical field B_v needs to be increased to 45 GS for keeping the increasing plasma current in equilibrium. Figure 3 shows clearly that the addition of TF dramatically increases the RMF driven current, the plasma current increased from 2.7 kA without TF to 4.7 kA with TF, almost doubled. It also indicates that after launching the RMF, the magnetic field produced by RMF driven plasma current reverses the vacuum vertical field (marked by dash line).

In the case without TF, this is a compact FRC^[1,10] and in the case with TF a spherical tokamak (ST) configuration is generated, as reported in Refs.[1,14].

Besides improving the total RMF driving current, the addition of the TF can also change the plasma current density profile. The plasma current density can be deduced from $J_\theta = \frac{1}{\mu_0} \left(\frac{\partial B_r}{\partial z} - \frac{\partial B_z}{\partial r} \right)$. In experiments only the magnetic fields along the $z = 0$ central channel have been scanned, and the first term in J_θ expression is omitted, which implies that the infinite cylindrical configuration approximation is used, which is not quite accurate in our case. By comparing the profiles of our J_θ with that obtained in the previous rotamak operation,^[14] where the complete (r, z) -scan of the magnetic field was performed, we find that neglecting the first term in J_θ results in underestimation of the J_θ value, around 20%. But current density profiles are not much different. So $J_\theta = -\frac{1}{\mu_0} \frac{\partial B_z}{\partial r}$ is used to characterize the current density profile in the cases with and without TF.

Figure 4 is the profile of B_z in the $z = 0$ central channel at the middle of the shots in the cases with and without TF in the same discharge condition as Fig.3. Figure 5 is the current density profiles calculated by using $J_\theta = -\frac{1}{\mu_0} \frac{\partial B_z}{\partial r}$ from Fig.4. Before calculation, the experimental measurement B_z data

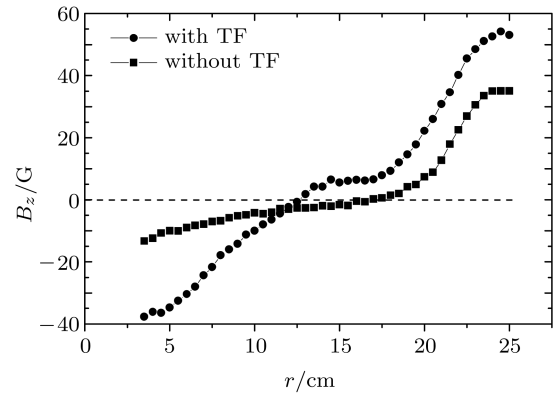


Fig.4. The profile of B_z in $z = 0$ channel with and without TF.

are fitted by a proper polynomial curve. From Fig.5, it is found that the current density in the rotamak has a distinct character that consists of two rings. The profiles are hollow with two peaks located on both sides of the magnetic axis, forming the inner and outer rings. The peak value of the outer ring is larger than that of the inner ring. In the TF case, however, the peak current density of the inner ring becomes larger

than that in the case without TF, roughly three times larger. And at the same time the peaks of two rings are closer to the magnetic axis in ST.

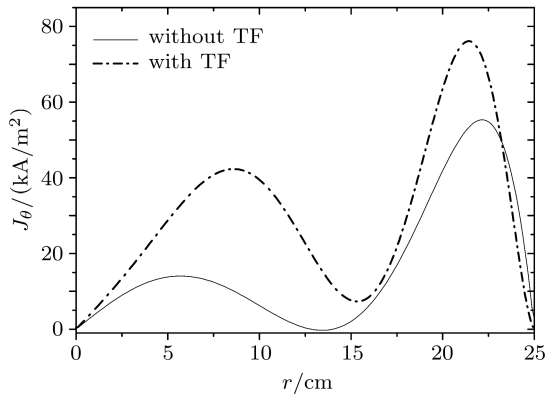


Fig.5. The current density profile with and without TF.

The RMF current drive in the rotamak is strongly dependent on the biased vertical magnetic field. Figure 6 shows the relation of the plasma current with the vertical magnetic field B_v in the cases with and without TF when the RF power and gas filling pressure are fixed at the same values as the Fig.3. It is indicated that in both cases, the plasma current I_p increases almost linearly with B_v , but in the discharges with TF the I_p grows faster with B_v than in the case absent from TF. Using the linear fit to the data, we obtain that the slopes are 86 A/G and 118 A/G in the cases with and without TF respectively. It is worth mentioned that the Solov'ev equilibrium^[18] predicts this ratio to be $\frac{I_p}{B_v} = \frac{5a}{\mu_0} = 99 \text{ A/G}$, where a is the radius of plasmas. The experimental results are very close to the predicted results. This implies that the rotamak plasma current should satisfy equilibrium magnetic configuration, and the rotamak has the common properties as other magnetic confinement devices.

Another important result from Fig.6 is that the addition of TF not always enhances the RMF current drive. Actually in low vertical field the presence of TF will decrease the plasma current. It was reported that in cases without TF, for a certain RF power, the current can not increase with B_v infinitely, there is a critical B_v (28 G as shown in Fig.6) that above this value current termination will occur.^[17,19] From Fig.6, it seems that below this critical B_v , the presence of the TF is disadvantageous for the RMF-driven plasma current. When the vertical field exceeds the critical value, the addition of TF increases the RMF current. However, similar to the case without TF, there is a

current termination point also in the case with TF ($B_v \approx 48 \text{ G}$ in the present case), when $B_v > 48 \text{ G}$ a stable plasma current can not be achieved. So the presence of TF postpones the occurrence of the current termination from low B_v region to high B_v value (from 28 G in the case without TF to 48 G in the case with TF).

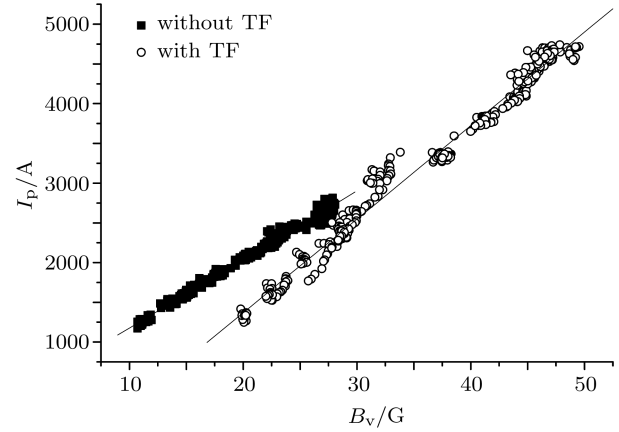


Fig.6. The dependence of plasma current I_p on the biased vertical field.

The I_p - B_v relation in the case with TF in Fig.6 is found with the condition that the axial current is fixed at 9 kA. In order to deep study the effect of the strength of TF on the RMF current drive in case with TF, we keep the other parameters unchanged; only varying the TF, i.e. the axial current I_z . The power from two RF generators is $P_{rf} = 2 \times 180 \text{ kW}$, the gas filling pressure is 0.18 Pa, as those in the previous case. The results are shown in Fig.7 for the two cases of the vertical fields 35 and 45 GS. It is shown that in a certain vertical field, RF power, and filling pressure, there is a

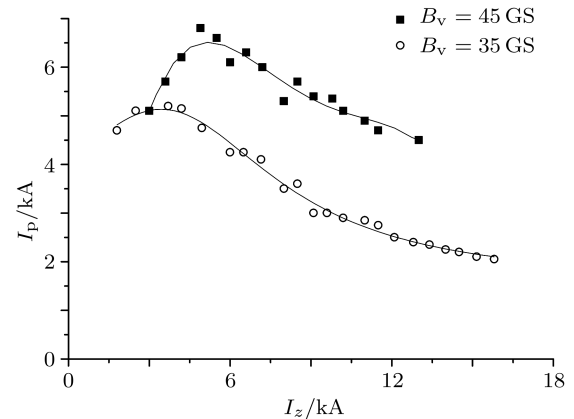


Fig.7. The relation of plasma current and the toroidal field current I_z .

range of the axial current values that the stable plasma can be obtained. For $B_v = 45$ G, for example, when the axial current is smaller than 3 kA or larger than 18 kA, it is difficult to get or maintain stable plasma current. The second conclusion from Fig.7 is the existence of the optimal value of the axial current for the largest plasma current, and this optimum is different for each value of B_v , for instance, 5 kA for $B_v = 45$ GS.

3.2. Effect of TF on harmonic fields oscillation and RMF penetration

It is important to know the extent that the RMF can penetrate into the plasma and the relation between the RMF penetration and the current drive in the rotamak. For studying the RMF penetration during the current drive in the rotamak plasma, the oscillating magnetic fields have been measured by pickup coils. And it is found that besides the fundamental fields, there are high order harmonics generated by

RMF and their frequency spectra are different in the shots with and without TF. Figure 8 gives the spectra of time derivative signals of $\dot{B}_r$, $\dot{B}_\theta$ in the same condition as that in Fig.6. It indicates clearly that significant harmonics are generated in the rotamak plasma. In the case without TF, the harmonics are dominated by 1, 3, 5, 7 *et al* odd orders, the amplitude of even orders are small in comparison with the odd ones. For instance the amplitude of third order is larger than second order. From Figs.8(c) and 8(d) when TF is added, amplitudes of both even and odd orders become large. Opposite to the case without TF, the amplitude of the second order now is larger than the third ones. The ratio between the maximum magnetic amplitude of harmonics and the fundamental one is $B_{3\omega}/B_{1\omega} \sim 26\%$ in the case without TF, and $B_{r2\omega}/B_{r1\omega} \sim 45\%$ in the case with TF. The reason for generating the harmonics is the Hall term $\mathbf{J} \times \mathbf{B}$ which was reported elsewhere.^[20]

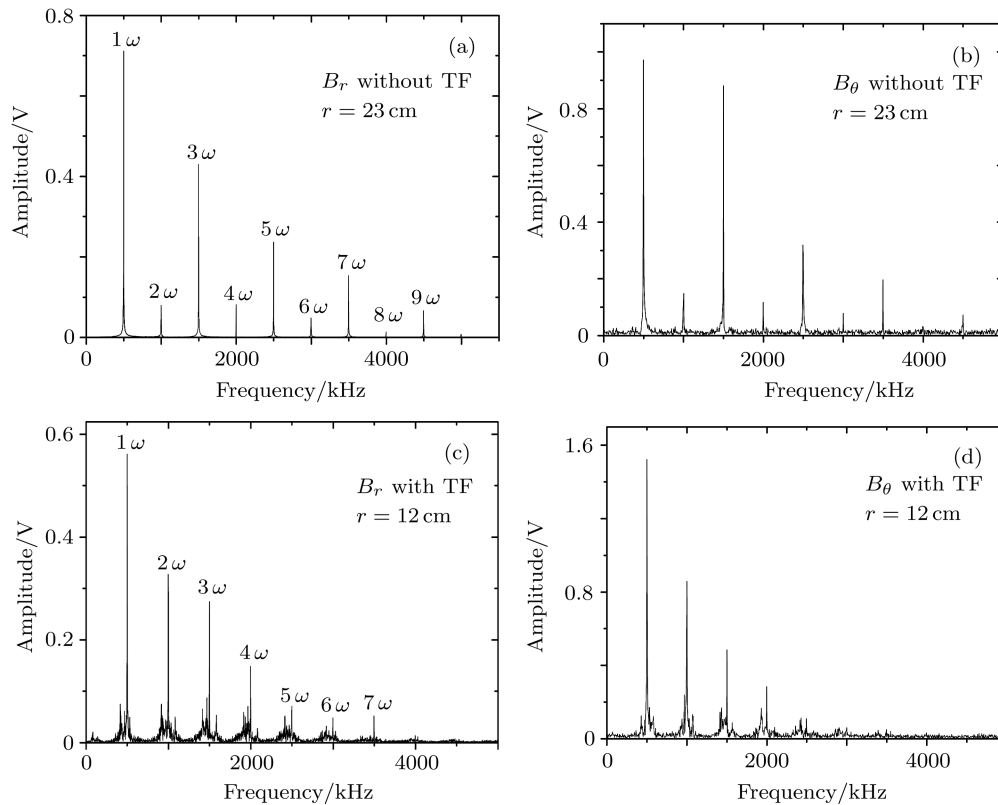


Fig.8. Frequency spectra of $\dot{B}_r$ and $\dot{B}_\theta$ without TF (a,b) and with TF (c,d).

Besides the change of the harmonic field frequency spectra, the addition of TF improves the RMF penetration greatly. Figure 9 is the radial profile of 1ω fun-

damental field. For comparison, the RMF magnetic profile in vacuum (without plasma) is also plotted. It is found that the penetration of RMF is very different

in the cases with and without TF. In the case absent from TF the RMF can only penetrate into plasma in the edge region, in the region from edge to 15 cm of the chamber. When TF is applied, penetration of RMF is improved dramatically, it now can be pene-

trated almost whole volume of the plasmas and the maximum field positions are shifted to inner region. Owing to the penetration of RMF and the generation of J_z in plasma, the amplitude of B_θ in both cases can be larger than its vacuum value.^[6,13]

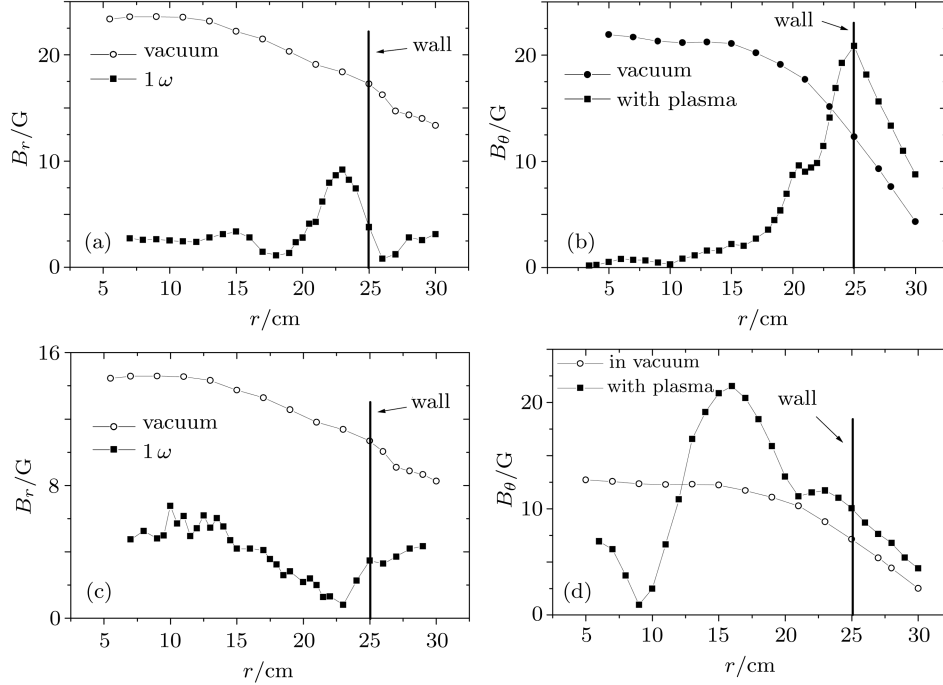


Fig.9. Radial profile of B_r and B_θ without TF (a,b) and with TF (c,d).

Another evidence of improving RMF penetration by TF is the change of plasma reflection resistance. In these shots, the powers coupled into the load are almost the same, which are monitored by $V_{ld} \times I_{ld}$ in Fig.2. But the RF current through RMF coils decreases from 88A in the case without TF to 51A with TF. The plasma reflection resistance calculated from $R_p = V_{ld}/I_{ld} - R_c$ ($R_c = 4\Omega$) increases from 26Ω to 87Ω . According to the air-core transformer model, the plasma reflection resistance can be written as^[17]

$$R_p = \frac{k^2 L_c r_p}{L_s}$$

where k is the coupling coefficient between the RMF coils and plasma current through the mutual inductance $M = k\sqrt{L_c L_s}$, L_s is the inductance of plasma, L_c is the inductance of RMF coil in Fig.2, r_p is the plasma resistance. In the discharges with and without TF, the L_s is not much different, which is evidenced by the little change of the matching and tuning capacitor when matching the discharges. And the r_p will be smaller in the case with TF owing to higher plasma

temperature (reported in Section 3.3). The increase of R_p is owing to the increase of the coupling constant k . It implies that the penetration of RMF is improved in the cases with TF.

3.3. Effect of TF on plasma density and temperature

The electron density and temperature in the discharges with and without TF are measured by double floating Langmuir probe. Figure 10 is the electron density and the temperature profiles in the $z = 0$ cm central channel in the same discharge conditions in Fig.3. It is shown that the addition of TF increases the electron temperature. The electron temperature increases from 20 eV in the shots without TF to 35 eV when the TF is added. In both cases, the electron temperature is almost uniform in the whole plasma region. But the maximum electron density in the case with TF is a little lower than that in the case without TF.

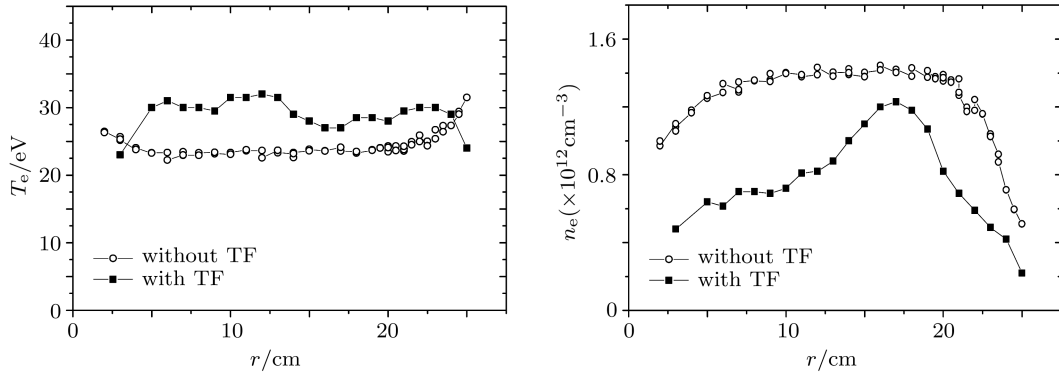


Fig.10. The electron temperature and density in cases with and without TF.

4. Conclusions and discussions

The effect of toroidal magnetic field on the RMF current drive in a rotamak is studied. The experimental results show that addition of an external TF increases the RMF driven current greatly and enhances the RMF penetration dramatically. Without TF, the RMF can only penetrate into plasma in the edge region. When a TF is added, the RMF can reach almost the whole plasma region. This is an optimal strength of toroidal magnetic field for getting maximum plasma current when B_v and RF generator power are fixed. The electron temperature increases also in the discharge with TF. The driven current is linearly proportional to the vertical field in the both cases with and without TF. But the addition of the TF postpones the ‘current termination’ which occurs in the small vertical field B_v in the discharges with TF and stable discharge can be obtained in higher B_v with larger current. Besides driving current, the RMF can generate high harmonic fields in a rotamak plasma. In the case without TF the harmonics are dominated by odd orders. The presence of TF enhances the even harmonics greatly.

From Fig.9, it is seen that the RMF magnitude is almost zero in the inner region of the plasma in the case without TF. So the inner current ring appearing in Fig.5 in this case is not driven by RMF. There should be other reasons to produce this current. One possible mechanism for generating the inner current ring is the convective flux of the particles in the rotamak plasma. As reported in Ref.[7] there is a poloidal swirling flow in RMF driven plasma, which provides a torque for the inner current ring, through the Hall term $u_r B_z$, where u_r is the radial component of the

swirling flow.

The value of the radial flow speed u_r can be estimated from $n_e u_r = -D_\perp \frac{\partial n_e}{\partial r} \approx -D_\perp \frac{n_{e0}}{a/2}$, so that $u_r \approx -\frac{2D_\perp}{a}$. Using the formula for diffusion rate, $D_\perp = n_e \eta_\perp k T_e / B^2$ we get $J_\theta = \frac{u_r B_z}{\eta_\perp} \approx -\frac{2D_\perp}{a \eta_\perp} B_z \approx -\left(\frac{2}{a}\right) \frac{n_e k T_e}{B_z} \approx -18 \text{ kA/m}^{-2}$, which is close to the experimental value.

The reason for improvement of the RMF driven current by TF is not clear as yet. Bertram^[15] and Watterson^[16] investigated the effect of the TF on RMF driven current respectively in the cylindrical configuration and obtained ambiguous results. Bertram’s numerical and analytical results indicate that, for a plasma of low resistivity, imposing a strong TF reduces the amount of current that can be driven, for a resistive plasma the effect is complicated. When the magnetic field strength is near that of RMF, $B_{\text{tor}} \sim B_\omega$, the present of TF would increase the driven current a little. When the TF increases, the driven current is reduced. Whereas from Watterson’s analytical results, the addition of the TF is always harmful to the penetration of RMF and reduces the current drive, no matter the plasma resistivity is low or high. As shown in our experiments, however, the TF always increases the RMF driven current, only if the biased vertical value is larger than that of the termination value. One reason that the present theories can not explain the experimental phenomena properly may be that the theoretical analysis is conducted in cylindrical configuration, and the real shape of the rotamak is a spherical torus. Another reason for the disagreement between the theory and experiments maybe that, from the increase of the electron temperature in

the case with TF, the addition of TF may improve the plasma confinement. So in the future theoretical study, the analysis should be conducted in spherical

configuration, the real shape of the rotamak device and the energy transportation process should be included also.

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Abstract Submitted
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Magnetic reconnection of two FRCs driven by rotating magnetic field¹ TIAN-SEN HUANG, YURI PETROV, XIAOKANG YANG, Prairie View A&M University — In order to study the magnetic reconnection of RMF-driven FRCs, a new device is built at Prairie View Plasma Physics Lab. The main feature of this device is one magnetic coil being added in the middle plane of a long cylindrical chamber. The magnetic field of the additional coil is applied to sever the FRC driven by rotating magnetic field into two FRCs. When that magnetic field is removed, magnetic reconnection occurs as the two FRCs merge into one. The advantage of this magnetic reconnection experiment is the enhanced stability of RMF driven FRCs in comparison with that created by pinch. The device uses our existing 2×400 kW rf power system, and the reconnection experiment is conducted in the condition without a toroidal magnetic field.

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Comparison of rotamak plasmas in FRC and ST configurations

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Abstract

The results of experiments in a newly rebuilt rotamak are presented. A comparison is made for two types of operations: the common field-reversed configuration (FRC) produced by driving plasma current with a rotating magnetic field (RMF), and the spherical tokamak (ST) configuration when a steady toroidal magnetic field is added. In both cases the driven plasma current develops two current rings, but in the ST configuration the inner ring current density is about three times larger than that in the FRC case. The addition of the toroidal field enables the ST to overcome the current limit for a given radio-frequency power. The total plasma current is found to have a peak at an optimum value of the applied toroidal field; the optimum toroidal field depends on the value of the equilibrium magnetic field. With proper choice of the toroidal field magnitude, the plasma current can be enhanced two- or three-fold in comparison with the limits in the cases of too a small or too large toroidal field. In the rotamak–ST, the temperature of electrons is increased by at least 50%. The density profile is triangular-shaped in the ST case and almost uniform in the FRC. The measurements of the oscillating fields indicate that the penetration of RMF into plasma is greatly enhanced in the ST case. The analysis of RMF profiles in the plasma supports the hypothesis that the improved penetration is due to the excitation of a whistler wave mode.

1. Introduction

Driving steady plasma current with an externally applied rotating magnetic field (RMF) was pioneered by Blevin and Thonemann [1] and later extensively studied by Jones with his team at Flinders University of Australia [2, 3] and by another group of scientists [4] in a series of devices called rotamak. Recently, the concept of RMF driving plasma current has gained considerable interest in the US [5–9].

The main components of the rotamak are displayed in figure 1. The RMF is produced with two pairs of coils fed by two 90°-dephased radio-frequency (rf) generators. In the figure, only one pair of coils is shown while the other pair is parallel to the figure plane, with the coils

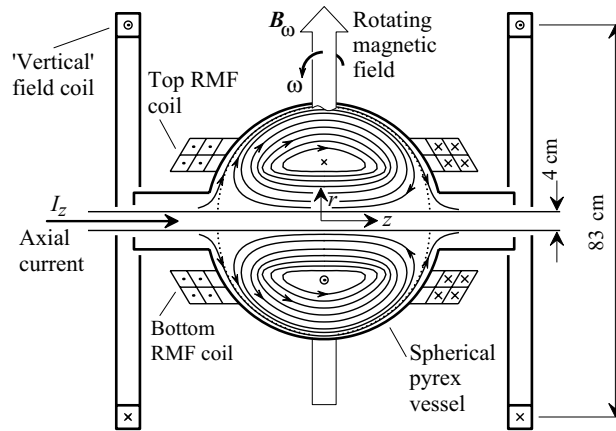


Figure 1. The schematic of the rotamak.

behind and in front of the chamber. The frequency f of the generators is chosen to satisfy the condition $\omega_{ci} < 2\pi f \ll \omega_{ce}$ where ω_{ci} and ω_{ce} are the ion and electron cyclotron frequencies with respect to the RMF magnitude, B_ω . Under this condition, the electrons are effectively tied to the rotating field lines while the ions remain almost unaffected; thus, the net plasma current is generated [1]. Another requirement is that the magnitude of B_ω is large enough to provide a good penetration of the RMF into the plasma volume, which is expressed by $\omega_{ce}/\nu_{ei} > a/\delta$, where ν_{ei} is the electron–ion collision frequency, a is the plasma radius, and $\delta = (2\eta/\mu_0\omega_{rmf})^{1/2}$ is the classical skin depth for the RMF frequency [10]. The produced plasma current is kept in equilibrium by an externally applied ‘vertical’ field, which is along the horizontal z -axis in the figure. When the plasma current is large enough, the ‘vertical’ field is reversed in the central area, resulting in a typical field-reversed configuration (FRC).

The most important modification of rotamaks in recent years is the addition of a steady toroidal magnetic field produced by an axial I_z current. With the steady toroidal field, the rotamak is able to operate as a typical spherical tokamak (ST), with the plasma current sustained by RMF rather than inductively. The earlier experiments had shown that the ST configuration is more favourable for the penetration of RMF than the FRC [11, 12]. Later, a systematic study of rotamak–ST plasmas was done by Flinders’ group at their 50 cm spherical chamber device [13].

In 2001, the Flinders’ rotamak equipment was rebuilt at Prairie View A&M University, where it recently started operating. This paper summarizes the results of initial experiments conducted at the rebuilt rotamak with emphasis on a further comparative study of rotamak operations as the FRC and the ST. This paper is organized as follows. Section 2 gives a brief description of the rotamak. The measured profiles of the steady axial magnetic field are analysed in section 3. In section 4, a discussion is given on the effect of the applied toroidal magnetic field on the magnitude of driven plasma current and the problem of current limitations. The features of the density and temperature profiles are discussed in section 5. The details of RMF penetration into plasma for the two cases are given in section 6, followed by the summary in section 7.

2. Experimental set-up

The chamber of the rotamak is a 60 litre (i.d. = 51 cm) spherical Pyrex vessel, having fifteen diagnostic ports spaced along the poloidal direction (z -axis). The chamber is pumped by a 500 litre s^{-1} turbo pump, which makes a background vacuum at the level of 1.0×10^{-6} Torr.

The RMF is produced by two orthogonal Helmholtz pairs fed by two 400 kW/0.5 MHz rf generators dephased by 90°. The power for the generators is provided by a capacitor bank of 192 kJ that maintains a steady rf wave-form for a period of 40 ms.

The plasma current generated by the RMF is kept in equilibrium by a ‘vertical’ magnetic field that is made by two coils with a 20 kW dc power supply. The coils can produce a field of up to 130 G in the vessel centre.

The system that produces toroidal magnetic field consists of a set of six wires fed through the central axial stainless steel pipe with an external diameter of 4 cm. The steady toroidal magnetic field B_ϕ is produced by discharging a capacitor bank (maximum 152 kJ) through the wires. The discharge current threads the central axis six times, resulting in a total I_z current up to 18 kA. (In most of the ST-experiments discussed here, however, we used $I_z = 9$ kA which produces a toroidal field $B_\phi = 144$ G at the half-radius of the chamber, $r = 12.5$ cm.)

In the experiments, the ‘vertical’ field B_v is started at about 250 ms before applying the RMF, while the toroidal field B_ϕ (for ST configuration) is started at about 10 ms before applying the RMF. During the 40 ms period when the RMF is applied, both B_v and B_ϕ remain constant within 5%.

During the experiments, hydrogen gas flows continuously into the vessel at a pressure of $p_f = 1\text{--}3$ mTorr. The plasma is started by using a pair of pre-ionization coils fed by a 650 W/13.6 MHz rf generator. Then the RMF generators are triggered, and the pre-ionization generator is turned off after 5 ms, so that the plasma is sustained by the RMF alone for the rest of the shot. The plasma discharges are very stable and reproducible from shot to shot.

To measure the driven plasma current I_p a Rogowski coil is pulled through the central steel pipe around a poloidal cross section. Two other smaller Rogowski coils are used for measuring the axial current I_z and the current through the ‘vertical’ field coils. The plasma temperature and density are measured with double floating Langmuir probes, while the ion rotation speed is measured with Mach probes. Miniature magnetic pickup coils and Hall probes are used to measure the oscillations of the magnetic fields in plasmas and their magnitude in different directions.

3. Profiles of reversed field and driven current

3.1. Measurements of plasma current and axial magnetic field

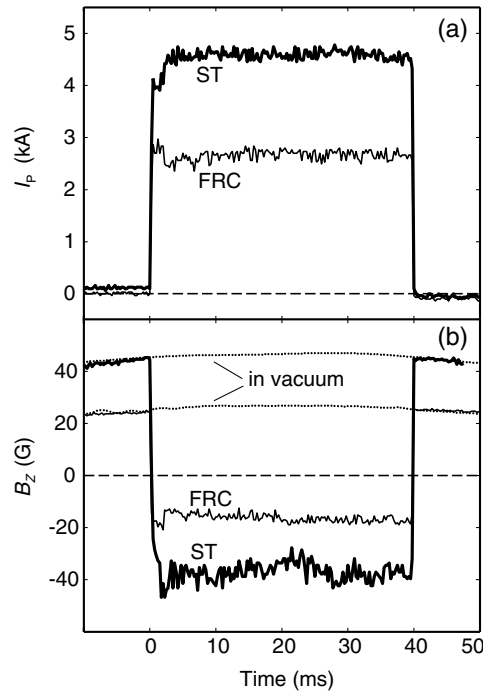
In the first round of experiments in the newly rebuilt rotamak, we focused on the comparison of the FRC and the ST discharges in regard to the magnitude of driven plasma current and the efficiency of RMF penetration into the plasma. The typical parameters for the two types of rotamak operation are given in table 1. In both cases, the total rf generator power (P_{rf}) and the gas filling pressure (p_f) are kept unchanged. In the FRC case, the bias ‘vertical’ field B_v at the chamber centre is 25 G and the plasma current I_p is 2.7 kA. In the ST case, an $I_z = 9$ kA axial current is added; the plasma current reaches 4.7 kA and the vertical field needs to be increased to 45 G to maintain the equilibrium. It should be noted that the plasma in both cases is weakly ionized: the filling pressure $p_f = 1.4$ mTorr corresponds to the neutral density $n_n = 4.5 \times 10^{13} \text{ cm}^{-3}$, 32 times larger than the electron density.

Table 1 also shows the value of the rf current through the RMF coils, I_{rf} , which is seen to decrease from 88 A in the FRC case to 51 A in the ST case corresponding to the increase of the plasma reflected resistance R_p from 26 to 64 Ω , which indicates a better rf-plasma coupling in the ST discharges.

In figure 2(a), the time courses of plasma currents are shown for the two types of operations. In the FRC case, the current has never been observed to exceed 3 kA for the specified level of

Table 1. The parameters of the two main types of rotamak operation.

	FRC	ST
p_f	1.4 mTorr	1.4 mTorr
P_{rf}	200 kW	200 kW
B_v	25 G	45 G
I_z	0	9.0 kA
I_p	2.7 kA	4.7 kA
n_e	$1.4 \times 10^{12} \text{ cm}^{-3}$	$1.4 \times 10^{12} \text{ cm}^{-3}$
T_e	25 eV	30–70 eV
I_{rf}	88 A	51 A
R_p	26 Ω	64 Ω

**Figure 2.** The time history of (a) the plasma current, and (b) the axial magnetic field at $r = 3.5$ cm from the chamber centre for the FRC and ST discharges. The dotted lines correspond to the fields in the absence of the plasma.

total rf power 200 kW, while in the ST configuration the largest achieved current is $I_p = 10.5$ kA (for the values of I_z and B_v different from the table 1), although the current in such a case tends to be less stable than that shown in the figure. As seen from figure 2(b), the axial magnetic field B_z produced by the plasma current reverses the direction of the initial ‘vertical’ field (shown with a dotted line) in the central area of the plasma (signal from $r = 3.5$ cm is shown). It should be noted that the external vertical field is rather flat during a shot; in this respect the present rotamak operation is different from the earlier operation [3, 13].

For both FRC and ST cases, the profiles of B_z during the discharges, measured with a Hall probe through the central $z = 0$ port, are shown in figure 3(a). It takes roughly 2 ms (1000 revolutions of RMF) for the field to become reversed in the central area of the plasma and to

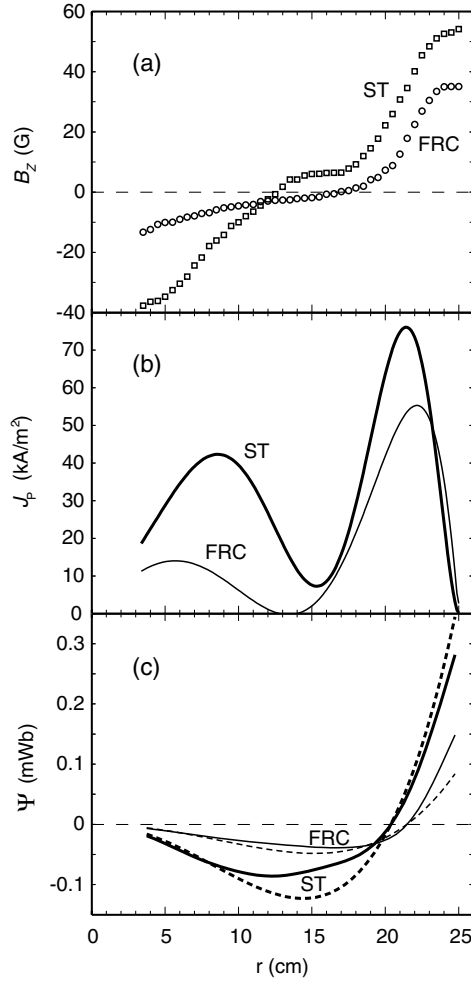


Figure 3. The radial profiles of (a) the axial magnetic field, (b) the plasma current density and (c) the poloidal flux function for FRC and ST discharges. The value of $r = 0$ corresponds to the chamber centre, and $r = 25$ cm to the chamber wall. The dashed lines are calculated from the Solov'ev model.

be stabilized. The profiles in figure 3(a) are shown for the middle of the discharge. It is seen that in the ST case, the magnetic axis is positioned further away from the walls compared with the FRC case, indicating a better RMF penetration in the ST discharges.

It is interesting to note a flattening of the B_z profile in the central region of the plasma in the ST case. A similar flattening has also been observed in the previous rotamak plasmas [13–15] and in cylinder-shaped RMF-driven FRCs [6, 8]. An unusual feature of our profile is that the flat region (corresponding to the minimum of the plasma current density) occurs on the right side of the magnetic axis, while in others the flattening happens to the left from the axis (inside the ‘field null’, in the case of the cylindrical FRC). However, in some shots with gradually growing plasma current we have also observed a sudden jump of the flattening region from one side of the $B_z = 0$ layer to another side. In fact, the position of the magnetic axis was moving away from the chamber centre, while the flat region remained approximately at the same location.

The profile of plasma current density at the midplane ($z = 0$) can be inferred from

$$J_p = \frac{1}{\mu_0} \left(\frac{\partial B_r}{\partial z} - \frac{\partial B_z}{\partial r} \right). \quad (1)$$

Using the poloidal flux function,

$$\Psi(r, z) = 2\pi \int_0^r r' B_z(r', z) dr', \quad (2)$$

and $\nabla \cdot \mathbf{B} = 0$, the equation can be rewritten as

$$J_p = -\frac{1}{\mu_0} \left(\frac{1}{2\pi r} \frac{\partial^2 \Psi}{\partial z^2} + \frac{\partial B_z}{\partial r} \right). \quad (3)$$

Because we scanned the magnetic field along the equatorial chord only, the first term in the bracket is unknown; omitting this term implies switching to an approximation of the infinite cylinder plasma, which is not quite accurate in our case. However, we notice that both terms in equation (3) are of the same sign, so that neglecting the first term would reduce the actual value of J_p . Indeed, comparing the profiles of our J_p with that obtained in the previous rotamak operation [13], where the complete (r, z) -scan of the magnetic field was performed, we find that neglecting the first term in equation (3) results in underestimation of J_p by approximately 20%.

Before calculating the value of $\partial B_z / \partial r$, a statistical scattering present in the experimental data for B_z (see figure 3(a)) is eliminated by a proper polynomial fit to the data. The profiles of J_p for the two types of operation are compared in figure 3(b). In both the ST and the FRC cases, the profiles are hollow with two peaks located on both sides of the magnetic axis, forming the inner and outer current rings. In the ST discharges, however, the peaks are closer to the magnetic axis, and the peak current density of the inner current ring is roughly three times larger than in the FRC case.

3.2. Formation of the inner current ring

In the FRC case, the inner current ring is located in the region where the RMF magnitude is practically zero. Similar double-peak profiles of current density have been observed in the early rotamak plasmas [11, 13, 15, 16] and in the present devices [7, 8]. As discussed in [8, 9, 17], a poloidally swirling flow of electrons could possibly provide a torque for the inner current ring, through the Hall term $u_r B_z$, where u_r is the radial component of the swirling flow. The evidence for such swirling currents has been indicated in earlier studies [18, 19] where a self-generated toroidal magnetic field has been detected in FRC-type rotamak plasmas, oppositely directed in the two hemispheres. A peculiar quadrupole structure of such self-generated toroidal field is observed in our recent experiments; the detailed results will be reported in another paper.

The value of the radial flow speed u_r can be estimated from

$$n_e u_r = \frac{-D_\perp \partial n_e}{\partial r} \approx \frac{-D_\perp n_{e0}}{a/2}, \quad (4)$$

so that

$$u_r \approx \frac{-2D_\perp}{a}. \quad (5)$$

Using the formula for diffusion rate, $D_\perp = n_e \eta_\perp k T_e / B^2$, we can find

$$J_\phi = -\frac{u_r B_z}{\eta_\perp} \approx \left(\frac{2}{a} \right) \left(\frac{D_\perp}{\eta_\perp} \right) B_z \approx -\left(\frac{2}{a} \right) \frac{n_e k T_e}{B} \approx -18 \text{ kA m}^{-2}, \quad (6)$$

which is close to the experimental value (the electron toroidal flow direction is positive and same as in the outer current ring). The magnitude of the radial flow speed, and therefore the

value of J_ϕ , can be reduced if the gradient of density is smaller than the assumed $n_{e0}/(a/2)$, above.

Another cause for the inner current might be the inertia of electrons. The electrons that gain toroidal momentum from RMF at the edge of the plasma and then travel to the inner part, tend to rotate at a faster toroidal angular velocity, by the ratio of radii squared. Because of conservation of angular momentum, the toroidal spin-up acceleration is given by

$$\frac{du_\phi}{dt} = -\omega_e u_r, \quad (7)$$

where $\omega_e \equiv \omega_e(r)$ is the angular velocity of electron flow, and $u_\phi = \omega_e r$. Balancing with the frictional force,

$$m_e(-\omega_e u_r) = -e\eta_\perp J_\phi, \quad (8)$$

we obtain

$$J_\phi = \frac{m_e \omega_e u_r}{e\eta_\perp} \approx -\frac{2}{a} \frac{m_e \omega_e D_\perp}{e\eta_\perp} = -\frac{2}{a} \frac{m_e \omega_e}{e} \frac{n_e k T_e}{B^2}. \quad (9)$$

Assuming $\omega_e \approx \omega_{\text{rmf}}$, the value of the inner ring current density is estimated to be $J_\phi \approx -0.1 \text{ kA m}^{-2}$. This value is much smaller than that given by equation (6). However in the ST case the RMF-driven current, besides the toroidal component, would also gain a poloidal one because of the helical structure of the magnetic field lines. Then, the value of u_r estimated earlier by equation (5) should be rather given by a fraction of $\omega_e a$ (depending on the inclination angle of the field lines). Therefore, the magnitude of J_ϕ , according to this mechanism, should be two or three orders of magnitude larger (depending on the value of $\eta_\perp$) than that calculated from equation (9).

3.3. Buildup of the poloidal flux

The profiles of axial magnetic field shown in figure 3(a) are used as an input into equation (2) to calculate the poloidal flux functions for the two types of operation; they are compared in figure 3(c). The location of separatrix (where $\Psi = 0$) is almost the same in both cases, $r_s = 21.5 \text{ cm}$ in the FRC, and $r_s = 20.4 \text{ cm}$ in the ST. However, the positioning of the nested magnetic surfaces ($\Psi = \text{constant}$) is quite different: in the FRC case the inner (smaller) surfaces tend to group towards the outside edge of the plasma, while in the ST case the centres of all surfaces remain close to the magnetic axis.

The dashed lines in figure 3(c) correspond to the analytic fit using the Solov'ev equilibrium model [20],

$$\frac{\Psi(r, z)}{2\pi} = -\left(\frac{\beta}{2}\right) r^2 + \left(\frac{\zeta}{2}\right) r^2 z^2 + \left(\frac{\vartheta}{12}\right) r^4, \quad (10)$$

where $\beta = -B_z(0)$, and two other coefficients can be found by knowing the two points of the separatrix, $(r_s, 0)$ and $(0, z_s)$,

$$\vartheta = \frac{6\beta}{r_s^2}, \quad \zeta = \frac{\beta}{z_s^2}.$$

Further, we will assume $z_s \approx r_s$. Although the fit with experimental data looks acceptable, the Solov'ev model cannot provide a good description of the double-peak shape of current profiles shown in figure 3(b). Using equation (10), the model current density can be calculated to be

$$J_\phi(r, z) = \frac{-r(\zeta + 2\vartheta/3)}{\mu_0} \approx \frac{5r B_z(0)}{\mu_0 r_s^2}, \quad (11)$$

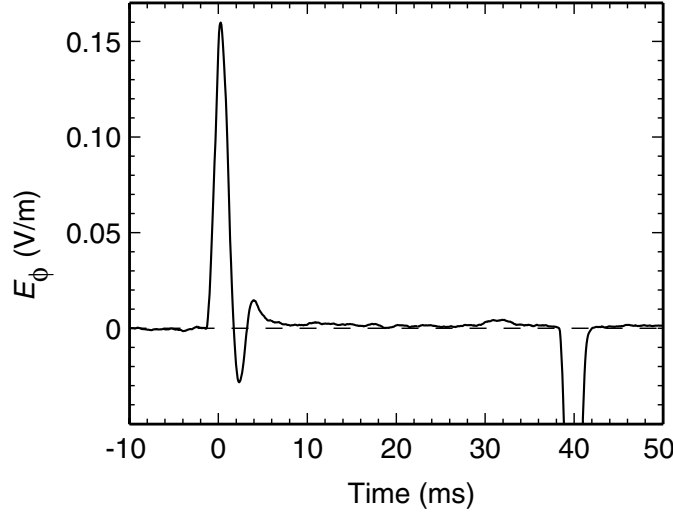


Figure 4. The time history of the toroidal electric field.

which is simply a linear function of the radius. Moreover, if the total plasma current is calculated using this model, giving

$$I_p = \frac{10\beta a^3}{3\mu_0 r_s^2},$$

it can account for only 44% of the plasma current measured with the Rogowski coil in the FRC case, and 80% of that in the ST case. The unaccounted part of the current can be attributed to the observed change of its profile at $z > 0$ and $z < 0$ cuts, where the hollow part of the profile gradually vanishes, while the peak value remains almost unchanged [13].

The axial magnetic field was measured during the whole time range of the shots that allows analysing the process of the flux build-up. Figure 4 shows the time behaviour of the toroidal electric field,

$$E_\phi = -\frac{1}{2\pi r} \frac{\partial \Psi}{\partial t}. \quad (12)$$

The trace of E_ϕ is shown for $r = 20$ cm only; the signals for the other radii are similar. It is seen that after a few oscillations, the electric field relaxes to nearly zero, and so the steady state is reached (in fact, E_ϕ continues to randomly oscillate within the range ± 2 mV m⁻¹). From the plot, the penetration time can be seen to be 2 ms. This value is larger than that given by the formula for the lower limit of transient time [10],

$$\tau_{\text{pen}} = \frac{\mu_0 a^2}{8\eta} \approx 0.1 \text{ ms},$$

where resistivity $\eta = m_e v_{\text{eff}} / (n_e e^2) \approx 80 \mu\Omega \text{ m}$ is adopted in correspondence with $v_{\text{eff}} = (v_{\text{ei}} + v_{\text{en}}) \approx (0.45 + 2.7) \times 10^6 \text{ s}^{-1} \approx 3 \times 10^6 \text{ s}^{-1}$. In the calculation of the collision rate of electrons with neutrals we used $v_{\text{en}} = n_n \sigma_{\text{en}} V_{T_e}$, with the density of neutrals $n_n = 4.5 \times 10^{19} \text{ m}^{-3}$, cross section $\sigma_{\text{en}} = 2 \times 10^{-20} \text{ m}^2$ and electron thermal speed $V_{T_e} = 3 \times 10^6 \text{ m s}^{-1}$ (for $T_e = 25 \text{ eV}$). Alternatively, the penetration time can be estimated using [21]

$$\tau_{\text{pen}} = \frac{\lambda^2}{2\omega_{\text{rnf}} \sqrt{(\gamma - \gamma_c)/\gamma_c}}. \quad (13)$$

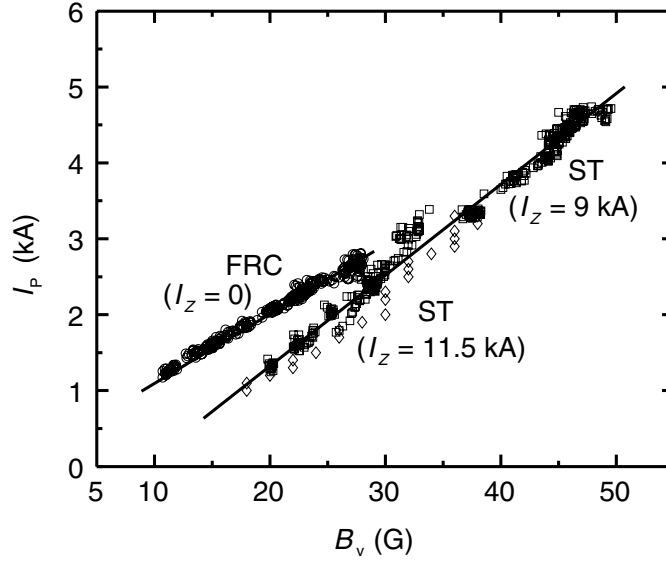


Figure 5. The dependence of the plasma current on the ‘vertical’ magnetic field for the FRC ($\circ$) and the ST ($\square$) for $I_z = 9$ kA, ($\diamond$) $I_z = 11.5$ kA discharges.

Here, $\lambda = a/\delta$, $\gamma = \omega_{ce}/\nu_{eff}$ and γ_c , the critical value for γ , is equal to $1.12\lambda(1 + 0.12(\lambda - 6.5)^{0.4})$. For our plasmas, assuming $\nu_{eff} \approx 3 \times 10^6 \text{ s}^{-1}$ in calculations of λ and γ , we find $\lambda \approx 40$, $\gamma \approx 100$ and $\gamma_c \approx 70$, that yields $\tau_{pen} \approx 0.3$ ms. If, however, a larger effective collision rate is assumed, $\nu_{eff} \approx 11 \times 10^6 \text{ s}^{-1}$ that corresponds to anomalous resistivity $\eta \approx 280 \mu\Omega \text{ m}$, the value of γ approaches $\gamma_c (\approx 32)$, and the penetration time is increased to $\tau_{pen} \approx 3$ ms, which is close to the observed value. The anomalous resistivity might be caused by micro-instabilities induced by the RMF [8, 22, 23].

4. Effect of the toroidal field on current drive

The plasma current driven by the RMF depends on the gas filling pressure, the ‘vertical’ field magnitude and the rf power. In the ST case, the addition of the toroidal field introduces another parameter.

For the fixed maximum deliverable power $P_{rf} = 200$ kW (which is determined by the capacitor bank voltage for generators) and the gas filling pressure $p_f = 1.4$ mTorr, the dependence of the plasma current on the value of B_v is given in figure 5. For the ST configuration, the two axial currents, $I_z = 9$ and 11.5 kA, were applied. As expected, the value of the plasma current grows almost linearly with B_v , but in the ST discharges the value grows faster with B_v than in the FRC. Using the linear fit to the data, we obtain the slope in the FRC discharges as 86 A G^{-1} , while in the ST discharges it reaches 118 A G^{-1} (and approximately equal for the two cases of I_z). In the earlier rotamak-FRC plasmas a value of 92 A G^{-1} was measured [24]. It is worth mentioning that the Solov’ev equilibrium [20] predicts this ratio to be

$$\frac{I_p}{B_v} = \frac{5a}{\mu_0} = 99 \text{ A G}^{-1}.$$

In our experiments, no plasma current could be produced for the ‘vertical’ field larger than 28 G in the FRC discharges with the specified level of $P_{rf} = 200$ kW; this result is in agreement

with the earlier experiments [15, 24, 25]. As seen in figure 5, when the magnitude of B_v is below this critical value, the presence of the toroidal field (axial current I_z) is disadvantageous for the RMF-driven plasma current. However, when B_v exceeds the critical value, the application of the toroidal field enables the generation of the plasma current and makes its further growth with the increase of B_v possible. Similar to the FRC case, the driven current also has a termination point (for $B_v \approx 48$ G in the present case), above which the plasma becomes unstable. It can be concluded that the addition of the toroidal field postpones the occurrence of the plasma current termination from a low B_v region to a higher one.

The problem of the current termination has been discussed in a number of papers [15, 24, 25]. It is related to the way the rf power is coupled to the plasma. From a simple circuit theory, the total power dissipated by the two rf channels in the plasma is given by [24]

$$P_p = I_{\text{rf}}^2 R_p = \frac{V_0^2 R_p}{(R_p + R_g + R_c)^2}, \quad (14)$$

where V_0 is the open circuit voltage of each generator (proportional to the capacitor bank voltage), $I_{\text{rf}} = V_0/(R_p + R_g + R_c)$ is the amplitude of the rf current flowing in the coupling circuit (RMF coils), R_p is the plasma resistance reflected into the circuit by the presence of the plasma, R_g is the resistance of one generator ($\approx 18 \Omega$) and R_c is the resistance of each pair of RMF coils ($\approx 8 \Omega$); the inductive impedance of the circuit is tuned out. According to the FRC-type experiments [15, 24, 25], the plasma discharges are stable when $R_p < R_g + R_c$. As the vertical field and plasma current are increased, the value of R_p is observed to increase too (the exact analytical formula relating R_p to I_p or B_v is unknown). For the FRC operation, the termination of plasma current usually occurs when $R_p \approx R_g + R_c$. For smaller levels of filling pressure it happens when R_p is somewhat smaller than $R_g + R_c$, and for larger filling pressure, R_p can slightly exceed the value of $R_g + R_c$ before the termination point is reached. The value of the plasma resistance $R_p = R_g + R_c$ corresponds to the maximum of rf power that can be transferred to the plasma,

$$P_{p,\text{max}} = \frac{V_0^2}{4(R_g + R_c)}; \quad (15)$$

this is identified with the maximum deliverable power P_{rf} . If the value of R_p is progressively higher than $R_g + R_c$, the power delivered to the plasma continues to decline with R_p , until it drops below the level required to maintain a steady ionization rate, or until the amplitude of the rotating field drops below the critical value when the RMF penetration into the plasma becomes impossible.

In the rotamak-ST operation, the presence of the toroidal field allows overcoming of the threshold $R_p = R_g + R_c$ significantly. As follows from the measurements using RF voltage and current probes, the plasma reflected resistance in the ST discharges that are close to the current termination point is 64Ω , in comparison with $R_p = 26 \Omega$ in the FRC discharges. Because of this increase in R_p , the amplitude of the current in the RMF coils is smaller by a factor of $(64 + 26)/(26 + 26) \approx 1.73$. Indeed, the measurements of the amplitude of the RMF at the plasma edge in the ST case show a drop of the field by two times as compared with the FRC case. However, in the central region of the plasma ($r \approx 13$ – 16 cm) the amplitude of RMF in the ST discharges grows more than twice relative to the edge, while in the FRC case it drops almost tenfold. It appears that in the ST case the partitioning of the rf power delivered to the plasma becomes more efficient—a considerable portion of the power that was channelled into driving the screening currents in the FRC is now used for driving the plasma current.

For a deeper understanding of the role of the toroidal field in ST discharges, a series of experiments were conducted where the vertical field was fixed, while the toroidal field, i.e. the

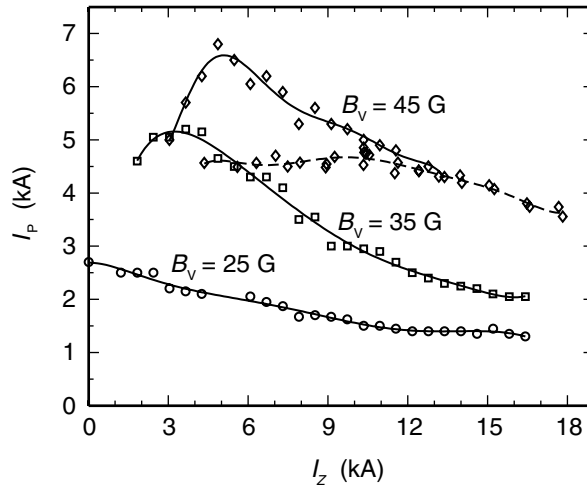


Figure 6. The dependence of the plasma current on the axial current for a fixed magnitude of the ‘vertical’ field.

axial current I_z , was varied. The maximum deliverable rf power was unchanged at 200 kW, and the gas filling pressure was maintained at 1.4 mTorr. The results are shown in figure 6 for the three cases of the vertical field, 25, 35 and 45 G. The first conclusion from our experiments is that for a given vertical field, the stable plasma can be obtained only in a certain range of the axial current values. For $B_v = 45$ G, for example, the minimum value of I_z for which the plasma can be produced is 3 kA, which corresponds to $B_\phi = 48$ G at $r = 12.5$ cm. The second conclusion is the existence of the optimal value of the axial current for the largest plasma current, and this optimum is different for each value of B_v . Quite a similar result has been observed in a smaller (10 litre) rotamak-ST [11]. Thus, our results suggest that the shown dependence of the plasma current on the toroidal field is universal for the rotamak-ST plasmas.

It occurs that as the vertical field is increased, the peak-shape dependence of the plasma current on the axial current (i.e. on the toroidal field) becomes gradually sharper. It is interesting to note that in the vicinity of the peak, the magnitude of the driven current becomes increasingly sensitive to other conditions like the gas filling pressure and the rf-plasma tuning/matching. For $B_v = 45$ G and $I_z \approx 5$ kA, for example, a slight variation in the tuning capacitors can result in the change of the driven plasma current by about 30%. This effect is shown with the two branches of $B_v = 45$ G in the figure. As also seen from the figure, the peak value of I_p/I_z ratio achieved for the ST configuration with $B_v = 45$ G is close to 1.4, which is about 26% higher than that in the previous rotamak-ST operation [13].

A number of analytical and numerical studies have been performed on the role of externally applied toroidal field in rotamaks [26–28] using the infinite cylinder approximation. They confirm qualitatively the view that the correctly chosen value of the toroidal field can increase the efficiency of the current drive. A numerical treatment of actual rotamak-ST plasmas with a spherical geometry still remains a big challenge [29].

5. Temperature and density in FRC and ST discharges

The temperature and density of the plasma have been measured with a double floating Langmuir probe. The voltage generator of the probe produces seven ramps during a shot, so that the

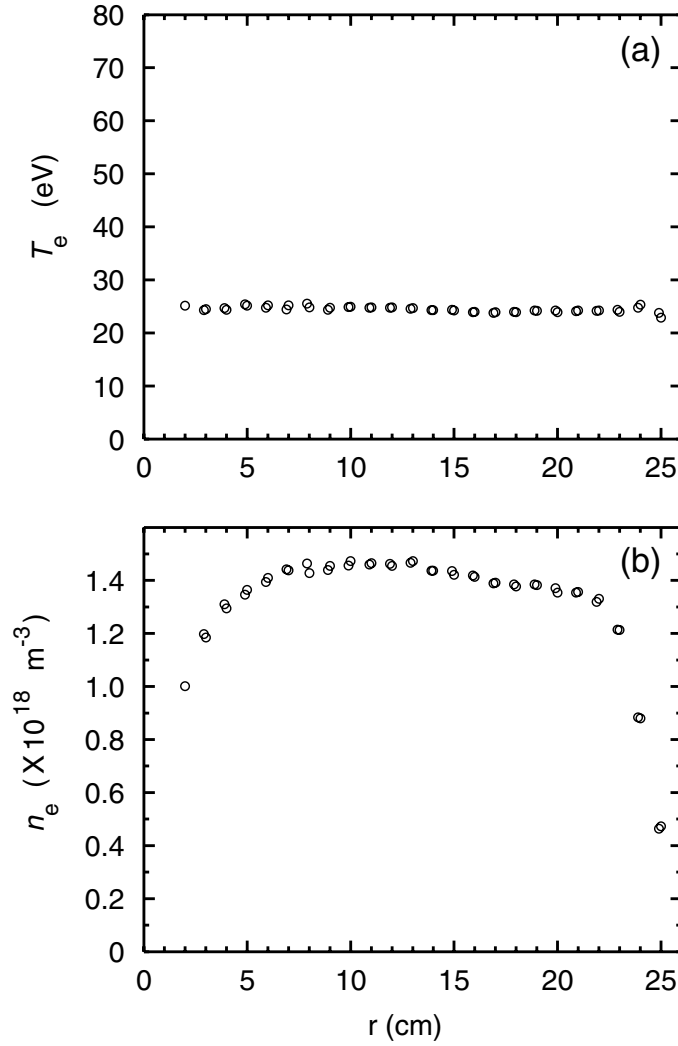


Figure 7. The radial profiles of (a) electron temperature, and (b) electron number density for the FRC discharges.

changes of the electron temperature and density could be observed for each discharge. All the radial profile scans were done through the middle diagnostic port ($z = 0$) of the chamber.

Typical profiles for the FRC discharges with parameters specified in table 1 are shown in figure 7. (The data from two rounds of scans are displayed; each point in the figure corresponds to the value averaged over a shot.) The temperature profile is very flat at the level of 25 eV, with only a small swing at the plasma edge. The density profile is also flat in most part of the plasma. The sharp bend of the density profile corresponds approximately to the position of the separatrix radius ($r_s = 21.5$ cm) where the density equals $1.34 \times 10^{18} \text{ m}^{-3}$, only 9% smaller than the peak value. Beyond $r = 22$ cm, the density steeply decreases. As seen from figure 8(a), the electron pressure calculated from the temperature and density is somewhat higher in the inner areas of the plasma ($r < 12.5$ cm). The same observation has been reported for the cylindrical-shaped, RMF-driven FRC plasma [8]; it was attributed to possibly existing

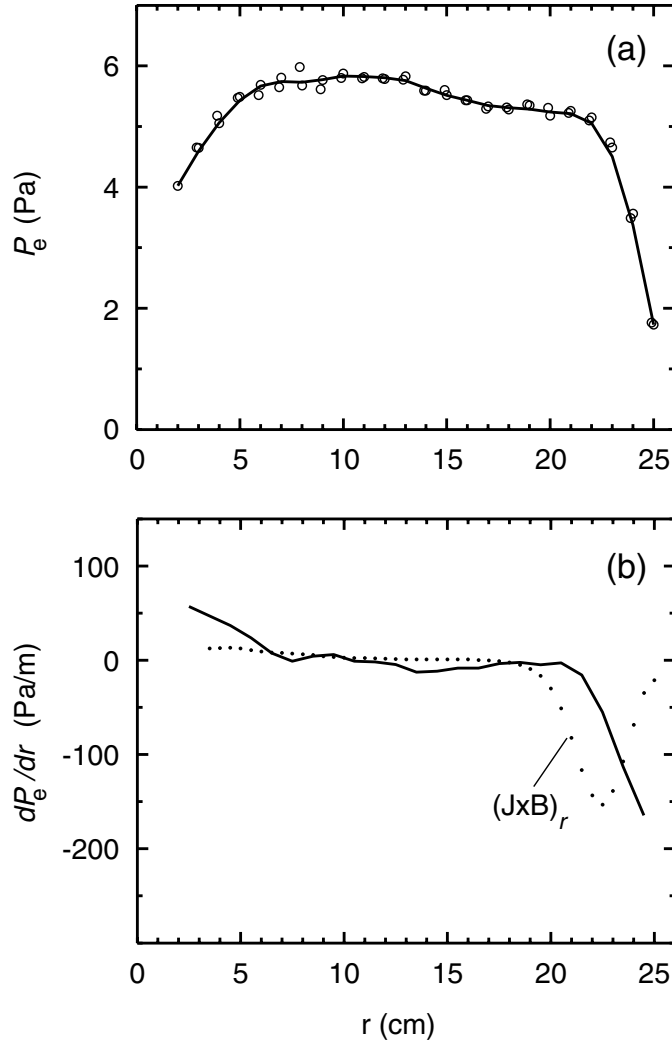


Figure 8. The radial profiles of (a) the electron pressure and (b) the gradient of pressure for the FRC discharges. The profile of $(\mathbf{J} \times \mathbf{B})_r$ is also shown for comparison ($\cdots$).

inwardly driven flow. Figure 8(b) shows that the profile of dP_e/dr closely follows the profile of $(\mathbf{J} \times \mathbf{B})_r = -J_p B_z$, derived from the data displayed in figure 3.

In the ST discharges with the parameters listed in table 1, the profiles become significantly different from those of the FRC case. Moreover, the temperature profile is changed during a shot. The dynamical change of the electron temperature is displayed in figure 9. It is seen that while the temperature remains almost unchanged in the outer areas of the plasma, in the region near the steel pipe it steadily grows from 45 to 65 eV. This effect was observed in many rounds of experiments. However, since a symmetric double probe samples only the high-energy part of the electron energy distribution, the measurements at the edge might be affected by deviations from a Maxwellian distribution, as was suggested in the case of the smaller rotamak-ST [11]. We should note, though, that the very same probe was used in both of our FRC and ST cases, and in the former, no non-Maxwellian tail seems to develop. The cause of

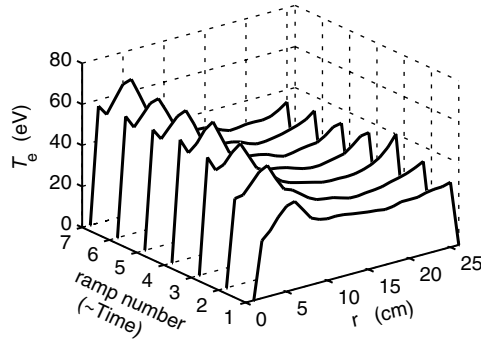


Figure 9. The temporal change of the electron temperature profile in the ST discharges. The ramp number corresponds to the voltage sweep at the Langmuir probe (in total seven ramps during a 40 ms shot).

the slow temperature increase (or more likely, formation of a group of high-energy electrons) near the pipe is unknown. The steel pipe bears numerous traces of particles; some of them are in the shape of long scratches, and others look like a ‘broccoli’ cascade pattern. Another piece of evidence of the presence of high-energy particles in the ST discharges is the quick (within one day) destruction of the Pyrex glass probes. The glass surface gets matted and fractured on one side of the probes (electron upstream flow side). The measurements discussed in this section have been done with an improved all-ceramic (alumina) probe that was able to sustain the particles’ impact.

To further investigate the effect of the central steel pipe on an apparent temperature rise, we isolated the pipe from the plasma by covering it with a quartz tube. The temperature profiles before and after the modification are compared in figure 10(a). With the quartz tube installed, the temperature profile became almost uniform (shown by the solid line). The profiles correspond to the time-average data; in the case of the quartz tube, the temperature fluctuates within ± 5 eV range during a shot, similar to the variations observed in the FRC discharges. We conclude that exposure of the steel pipe to the plasma presumably results in the formation of the sheath layer near the pipe surface with a subsequent acceleration of electrons in a radial electric field. The presence of this effect in ST discharges (but not in FRC) can be related to the $\sim 90^\circ$ -turn of the magnetic field lines near the pipe.

As seen from figure 10(b), the density profile in ST discharges becomes triangular-shaped, with the peak value being close to that in the FRC case. A similar modification in the density profiles, i.e. becoming more peaked as the toroidal field is increased, has been observed in smaller rotamak-ST [11]. A strong peaking is attributed to the diamagnetic poloidal current and the associated $\mathbf{J}_{\text{pol}} \times \mathbf{B}_{\text{tor}}$ force directed towards the magnetic axis. The density at the separatrix radius ($r_s = 20.4$ cm) in our experiments equals $0.88 \times 10^{18} \text{ m}^{-3}$, about 33% lower than the peak value, which indicates the improvement of particle confinement comparing to the FRC case. Another piece of evidence of the improved particle confinement in ST discharges is obtained by monitoring the H_α signal. The quantity n_e/H_α , which is a measure of electron confinement time, is seen to steadily grow with the increase of the applied toroidal field [3, 11].

In the ST discharges the change of the density profile during a shot is not large, only about 10–15%. In the FRC case, the change during the shots is even smaller, 7–9%.

The profiles of the electron pressure and its gradient (in the case with the quartz tube cover) are shown in figure 11, together with the profile of $(\mathbf{J} \times \mathbf{B})_r$, which is assumed to be equal to $(-\mathbf{J}_p \mathbf{B}_z)$. The noticeable difference between the profiles of dP_e/dr and $(\mathbf{J} \times \mathbf{B})_r$ can

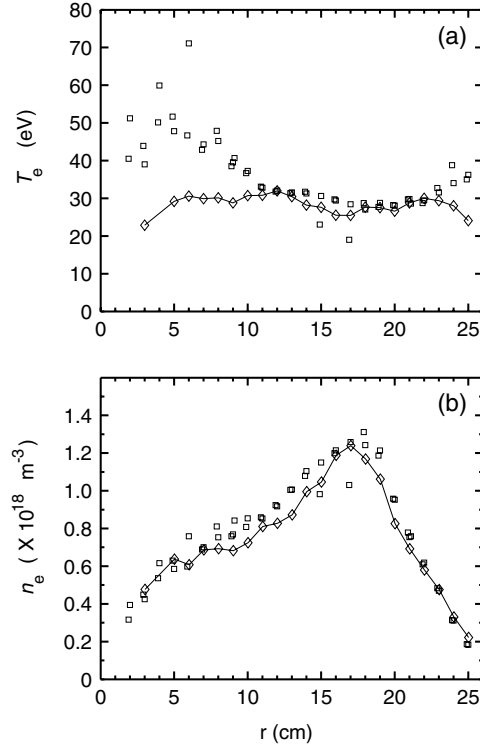


Figure 10. The radial profiles of (a) the electron temperature and (b) the electron number density in the ST discharges. The squares correspond to experiments in which the central steel pipe was exposed to the plasma, and the lines with diamonds describe the experiments with the quartz tube placed over the pipe.

be attributed to the unaccounted for $J_z B_\phi$ term, where J_z is the plasma current component in the z -direction (unknown), and B_ϕ is the external toroidal field.

It is seen that the volume-average electron pressure in the ST discharges is somewhat smaller than in the FRC discharges. Therefore, the plasma energy content $W_p = (3/2) \int P_e dV$ is lower in the ST shots (we assume that the ion temperature does not exceed 5 eV, so P_i can be neglected). However, the power delivered to the plasma in the ST discharges is also smaller, $P_p = I_{rf}^2 R_p \approx 166 \text{ kW}$ compared with $P_p \approx 200 \text{ kW}$ in the FRC discharges. Thus, the energy confinement time, which in the steady state regime can be estimated as $\tau_E = W_p / P_p$, is approximately the same in both the cases, $\tau_E \approx 2\text{--}3 \mu\text{s}$.

The measured electron density and plasma current profiles can be used to restore the profile of the toroidal angular velocity of electrons, ω_e , using

$$J_p(r) = en_e r \omega_e. \quad (16)$$

In the rigid rotor approximation, the value of ω_e has no dependence on the radius. However, as it follows from figure 12, the toroidal flow of electrons can by no means be regarded as the rigid rotor motion. A more striking result from the plot is that in the ST case (square markers) the ratio $\omega_e / \omega_{\text{rmf}}$ for the inner current ring considerably exceeds unity. A similar result can be deduced from the previous rotamak-ST operation [13]: for $r = 3.6 \text{ cm}$, for example, the ratio is estimated to be $\omega_e / \omega_{\text{rmf}} \approx 1.8$. The result could be attributed to errors in the Langmuir probe data, implying that the density is considerably undervalued. However, using the same probe in the FRC case yields reasonable values of $\omega_e / \omega_{\text{rmf}}$, 0.34 and 0.38, for the inner and outer peaks

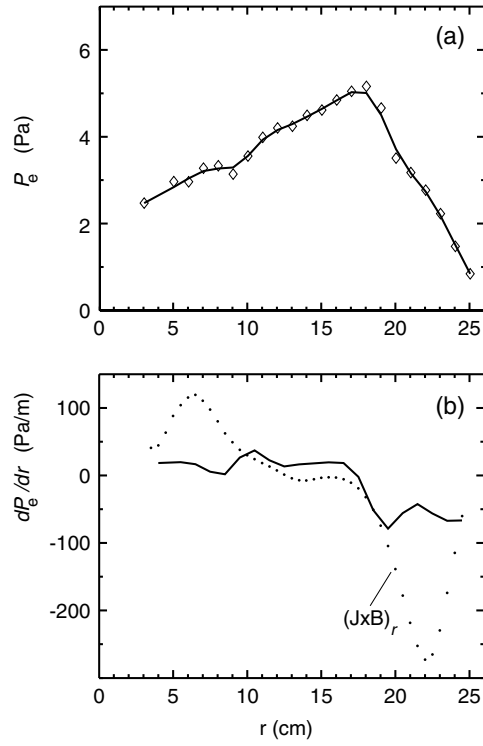


Figure 11. The radial profiles of (a) the electron pressure and (b) the gradient of pressure for the ST discharges. For comparison, the profile of $(\mathbf{J} \times \mathbf{B})_r$ is also shown.

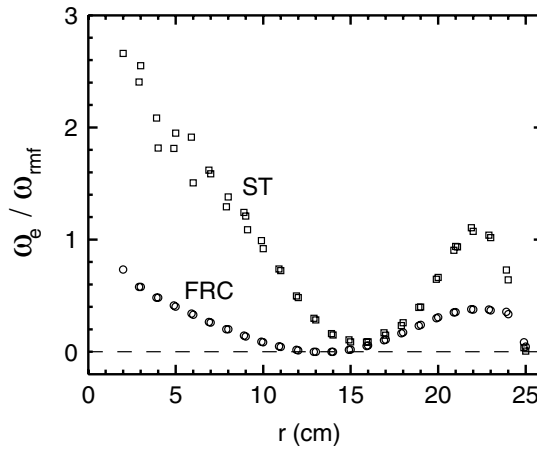


Figure 12. The ratio of the toroidal angular speed of the electrons to the RMF angular frequency in the FRC ($\circ$) and ST ($\square$) discharges.

in the current density profile. This can be compared with $\omega_e / \omega_{\text{rmf}} \approx 0.46$ and 0.36 , deduced from data given in the translation, confinement and sustainment (TCS) experiment [8]. It is a possibility that the driving mechanism involving the poloidal electron flows, discussed above, may result in a toroidal acceleration of the electrons in the inner current ring.

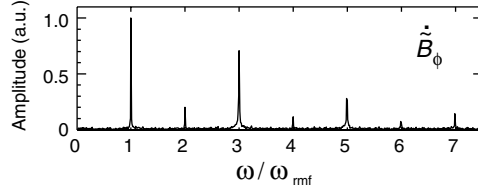


Figure 13. The frequency spectrum of $\dot{\tilde{B}}_\phi$ in the FRC case for the edge region of plasma $r = 22$ cm (where the RMF field is the largest).

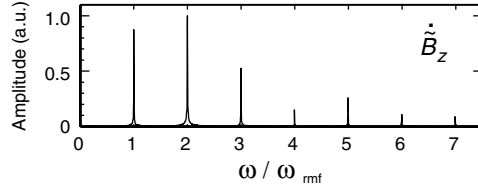


Figure 14. The frequency spectrum of $\dot{\tilde{B}}_z$ in the FRC case for $r = 23$ cm.

6. Oscillating fields in plasma

An effective penetration of the rotating field into the plasma is one of the main concerns in RMF-driven-current experiments, as the oscillating magnetic fields and currents are expected to supply a torque for the electrons [30]. The early theoretical considerations [10, 31] have predicted that the full penetration of the RMF into the plasma is possible provided $\gamma > \lambda$ (see explanation of designations following equation (13)). A recent study suggests a more accurate condition for full penetration, $\gamma > \gamma_c$ [21]. Although this condition is satisfied in the recent experiments with the rotating field, the observed penetration of the RMF remains only partial [7, 9]. The lack of deeper penetration is explained by the anomalous resistivity at the plasma edge [23].

The study of oscillating fields in our device has been performed with the miniature magnetic pickup coils. For all the experiments discussed here, we have used the middle diagnostic port ($z = 0$) located at the equatorial plane of the chamber. It is common to measure the radial profile of the magnetic field oscillations at the fundamental frequency, $\omega = \omega_{rmf}$, only. In our experiments, however, we also inspected the higher frequencies.

For the FRC case, the typical frequency spectrum of the probe signal is shown in figure 13. Similarly to the spectrum of $\dot{\tilde{B}}_r$ shown previously in [32], the spectrum of $\dot{\tilde{B}}_\phi$ in the FRC case is dominated by the odd harmonics, while the even harmonics are strongly suppressed. The spectrum of $\dot{\tilde{B}}_z$ component is shown in figure 14. As seen from the figure, the second harmonic prevails; in theory, the spectrum of $\dot{\tilde{B}}_z$ should possess only even harmonics.

When the axial current I_z is added to form the ST configuration, the frequency spectra are changed, as seen from figure 15. The amplitudes of the even-order harmonics in the spectra of $\dot{\tilde{B}}_\phi$ become larger and comparable with the neighbouring odd harmonics. In contrast to the case of the FRC, the amplitude of the second harmonic now is larger than the third one. A close inspection also reveals the splitting of the frequency spectral peaks. The ratio of the sub-peak to the main peak amplitude changes with the radius. The frequency gap between the main peak and each of the sub-peaks is about 80 kHz. However, the spectra of $\dot{\tilde{B}}_z$ displayed in figure 16 does not show such splitting. The figure also shows that in the ST case, the first harmonic in the

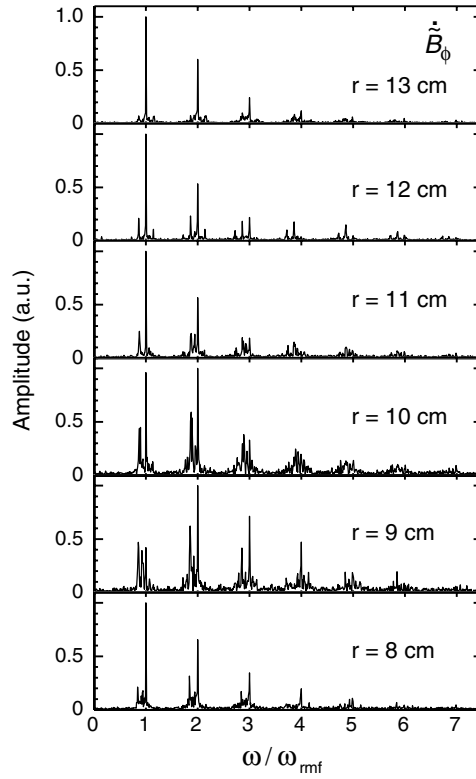


Figure 15. The frequency spectra of $\dot{\vec{B}}_\phi$ in the ST case for the middle region of the plasma (where the RMF field is the largest) showing the splitting of harmonics.

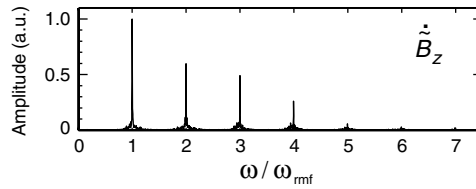


Figure 16. The frequency spectrum of $\dot{\vec{B}}_z$ in the ST case for $r = 12$ cm.

spectra of $\dot{\vec{B}}_z$ prevails. Unfortunately, the magnetic pickup coils could not provide accurate data for the slowly varying component (zero harmonic) of the magnetic field, which could be used to determine whether the ST configuration is para or diamagnetic in relation to the applied toroidal field. On the other hand, our measurements of B_ϕ with a Hall probe show that the applied toroidal field drops by 5–10% in the centre of the plasma during a discharge as compared with the vacuum value. Thus, the rotamak plasma is diamagnetic in regard to B_ϕ , confirming the earlier evidence that was obtained from the analysis of the oscillating currents' distribution in the plasma [12].

The numerical simulations using the simplified cylindrical geometry confirm the main features of the frequency spectra [32]. In the FRC case, the Hall term in the Ohm's law produces a cascade of odd-order harmonics for the radial and toroidal directions of the oscillating magnetic field, and even-order harmonics for the axial (z) direction of the field. For the ST

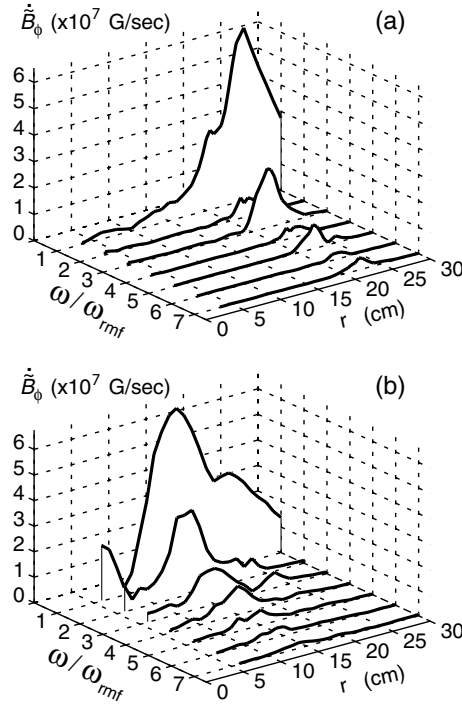


Figure 17. Radial profiles of harmonics for $\dot{B}_\phi$, (a) in the FRC case and (b) in the ST case.

configuration, introducing the toroidal field into the simulations results in the appearance of both even and odd order harmonics in the spectra of all components of the magnetic field.

The role of high harmonic oscillating fields in rotamak plasma is not fully understood. The harmonics are also present in the signal of the plasma current, which suggests that they might be harmful for stability. However, in recent TCS experiments [9] when the RMF antennas were moved closer to the plasma, that should have resulted in accentuating higher order harmonics, no deleterious effects were found. On the other hand, the observation of the peaks' splitting could be a useful diagnostic tool provided a connection between the degree of splitting and the plasma parameters can be found.

The radial profiles of the harmonics of $\dot{B}_\phi$ in the FRC and the ST cases are compared in figure 17. It is seen that in the FRC case, the field is localized at the plasma edge, although the conditions for full penetration are satisfied (however, if the anomalous resistivity, $\eta = 300 \mu\Omega \text{ m}$, or higher is adopted, the condition for full penetration is no longer satisfied, as discussed at the end of section 3.3). In the ST discharges, the presence of toroidal field results in a much better penetration of RMF, with its maximum being close to the magnetic axis. The improved RMF penetration has also been observed in the Flinders' rotamak-ST [13] and in a smaller rotamak ($a = 14 \text{ cm}$), which was the first one to have the steady toroidal field [12]. As suggested by the authors [12], the penetration is improved owing to the excitation of a rotating whistler wave mode, which is possible once there are oscillating currents perpendicular to the steady magnetic field. The main evidence for such a wave is a rapid 180° jump in the phase of the oscillating fields, at a certain radial position, correlated with a sharp bend in the amplitude of the oscillating fields. (In the FRC case, the phase of the fields was a slowly varying function of the radius.)

According to a simplified model (infinite cylinder, uniform density, zero resistivity and ignoring the field of the plasma current), the oscillating magnetic field components of the rotating whistler are given by [12]

$$\tilde{B}_r = \frac{iA}{\alpha r} J_{1/3}(\zeta), \quad (17a)$$

$$\tilde{B}_\phi = -Ar^2 J'_{1/3}(\zeta), \quad (17b)$$

$$\tilde{B}_z = Ar^2 J_{1/3}(\zeta), \quad (17c)$$

where the argument of the Bessel function is $\zeta = \alpha r^3/3$ with $\alpha = \mu_0 \omega n e / B_a a$ (B_a is the value of the applied toroidal field at the plasma edge), and A is an arbitrary constant. For our conditions $B_a = 70 \times 10^{-4}$ T and $\omega = \omega_{\text{rmf}} = 3.14 \times 10^6$ rad s⁻¹. For $n = 2.5 \times 10^{18}$ m⁻³ (somewhat larger than what was measured), the absolute values of the field components are plotted in figure 18, together with the experimental data. The coefficient A is set at 10^3 G m⁻². The sharp bend of the curves actually corresponds to the change of the sign (180° phase shift) of the field components given by equations (17a)–(17c). The position of the ‘bend’ is determined by the value of α , which is adjusted by the increase of n , to obtain the best resemblance between the curves and the experimental data. Taking into account the roughness of the model, the similarity between the data and the calculated curves can be regarded as supportive for the whistler wave excitation model in the rotamak–ST plasmas.

7. Summary

Steady plasmas have been produced and sustained in the newly rebuilt rotamak operated for FRS and ST discharges. Up to 10.5 kA of plasma current was driven with the RMF at 200 kW power level. The comparison of the two types of operations demonstrates that the addition of the toroidal magnetic field can result in an essential improvement of plasma parameters and the magnitude of the driven current.

The driven current consists of an inner and an outer current ring in both configurations, similar to the majority of the RMF-driven currents observed in other devices. In the ST case, however, the inner current ring is enhanced at least threefold compared with the FRC case, for the same rf power. Adding the toroidal field also postponed the current termination point from 3 to 5 kA under our selected conditions. In the ST case, the growth rate of the driven current with the ‘vertical’ field is 118 A G⁻¹ compared with 99 A G⁻¹ predicted by the Solov’ev model, while in the FRC case it is only 86 A G⁻¹. For the fixed rf power and gas filling pressure, the value of the driven current strongly depends on the magnitude of the added toroidal field, with its optimum position different for each value of the ‘vertical’ field. Such behaviour is similar to that observed earlier in a smaller 10 litre rotamak, therefore it appears to be universal, at least in the spherically-shaped rotamaks.

The electron temperature in the bulk of the plasma is 30–40 eV in the ST discharges, compared with 25 eV in the FRC discharges. The profile of electron density becomes much sharper in the ST case implying a better particle confinement.

The oscillating fields in the plasma bear quite different features in the two types of operations. The analysis of the frequency spectra not only confirms the early predictions in general, but it also reveals some details, such as the splitting of the spectral peaks in the ST case, that are not yet described theoretically. The penetration of the RMF is significantly improved in the ST case with the peak of the oscillating field close to the magnetic axis; in the FRC case, the field exponentially decays from the edge into the interior of the plasma. Evidence is found to support the theory that the improved RMF penetration might be due to

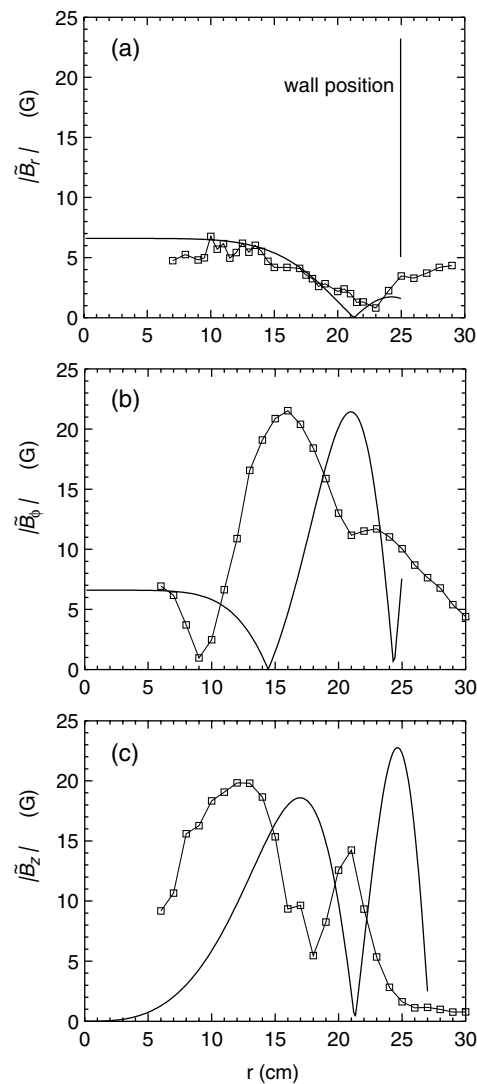


Figure 18. The radial profiles of oscillating fields in ST plasmas given by the whistler wave excitation model (—) and by measurements (—□—).

the excitation of the whistler wave mode. In our plasmas, the first radial mode appears to be present. A further study is needed to understand whether the excitation of the first radial mode corresponds to the best conditions for efficient current drive. The excitation of the whistler wave mode, whether using the toroidal field or other means, might be a key to an efficient current drive scheme as it allows minimizing the rf power losses related to the screening currents.

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High harmonic fields in a rotamak plasma

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High harmonic oscillations of the magnetic fields in a rotamak plasma are measured for the first time in experiments. Experimental results indicate that besides the fundamental rotating magnetic field, harmonic fields are stimulated simultaneously. The features of the harmonics in the discharges with and without toroidal field are compared. In the case without toroidal field the harmonics are dominated by odd orders, as expected by theory. The presence of toroidal field enhances the even harmonics greatly. The ratio of amplitudes of second harmonic and the fundamental one can be as high as 45% in this case. Numerical simulation confirms that the Hall term $\mathbf{J} \times \mathbf{B}$ is the cause of the harmonic generation. © 2004 American Institute of Physics. [DOI: 10.1063/1.1649990]

A rotamak is a compact torus configuration having the distinctive feature that the toroidal plasma current is driven in a steady-state, noninductive fashion by means of a rotating magnetic field (RMF).^{1–3} The plasma is kept in equilibrium by an externally applied magnetic field (referred to as the “vertical field”); the RMF driven current reverses the equilibrium field in the central area generating a field reversed configuration (FRC). Owing to its special features, the rotamak has attracted much attention and significant progress has been made in experimental and theoretical studies.^{4–13}

The principle of current driven by a RMF in the rotamak is the $\langle \mathbf{J} \times \mathbf{B} \rangle$ scheme which has been extensively studied.^{1,2,10–17} In the rotamak a rotating magnetic field is applied in the form of $b_r = B_\omega \cos(\omega t - \theta)$ and $b_\theta = B_\omega \sin(\omega t - \theta)$, where B_ω is the amplitude of RMF, and ω is the rotating angular frequency. This field induces an electric field $E_z = \omega r b_r$ and thus an oscillating current j_z is generated. Here r , θ , z correspond to the radial, azimuthal, and axial directions in cylindrical geometry. The Hall term ($j_z b_r$) in the generalized Ohm’s law,

$$\mathbf{E} - \frac{1}{n_e e} (\mathbf{J} \times \mathbf{B}) = \eta \mathbf{J} \quad (1)$$

has two parts, a dc part and an ac part. The dc part drives the steady azimuthal current J_θ , the plasma current in a rotamak, while the ac part oscillates at 2ω . The Hall term ($j_z b_\theta$) also has a 2ω component. These 2ω components directly contribute to currents and fields, which may interact with 1ω currents and fields resulting in 3ω harmonic fields. The higher 4ω , 5ω , etc., harmonic fields can also be generated in the plasma. From Eq. (1) and Maxwell equations, it can be shown that, if there is no external azimuthal field, the total B_r , B_θ , and J_z in plasma have odd order harmonics while B_z , J_θ , and J_r have zero and even order harmonics.¹⁴

In previous theoretical and experimental studies, most attention has been focused on the RMF current drive and the properties of rotamak plasmas. In theoretical studies, the high harmonics are often ignored to simplify the derivations, while in experimental investigations the high harmonic fields

have been neglected. This Letter reports the first measurement of high harmonic oscillations in a rotamak plasma. The results indicate that besides the fundamental oscillation, high order harmonic fields with large amplitude are also present in a rotamak plasma.

Figure 1 is the structure of the rotamak device, which has been rebuilt at Prairie View A&M University based on the parts of the Flinders University’s rotamak. The rotamak chamber is a 50 cm diameter spherical vessel made of pyrex glass. The RMF is generated by passing rf currents, from two 400 kW/500 kHz rf generators, dephased by 90°, through two orthogonal Helmholtz coils. The plasma current driven by a RMF is kept in equilibrium by the vertical magnetic field produced by two 83 cm diameter coils. In addition, by using an axial current, a toroidal field (TF) can be added to the rotamak plasma. The axial current is generated by discharging storage capacitor (RC) through six series connected copper conductors inside a stainless steel tube (o.d.=4 cm) positioned along the major axis of the chamber. A Rogowski coil through the central tube and around a poloidal cross section is used to measure the plasma current. Miniature magnetic pickup coils, which are retractable through quartz guide tubes (o.d.=6 mm) sealed in diagnostic ports, are used to measure the oscillating magnetic fields. A double floating Langmuir probe is used to measure the electron density and temperature of the plasmas.

Discharges with and without TF are carried out in experiments. The rf generator power (2×180 kW) and the gas filling pressure (1.4 mTorr H_2) are kept unchanged in the two cases. In the shots without TF, the vertical field is 25 G and the plasma current is 2.5 kA. In the case with TF, a 9 kA axial current is added; the plasma current is 4.5 kA and the vertical field is increased to 45 G to maintain equilibrium. In experiments, the plasma current duration is 40 ms as determined by the rf pulse length, and the discharges are quite stable and reproducible from shot to shot. The oscillating magnetic fields are measured through the central diagnostic port in the meridian plane of the chamber. The data used for analysis are recorded around the middle of the discharges when the plasma current is in its flat stable stage.

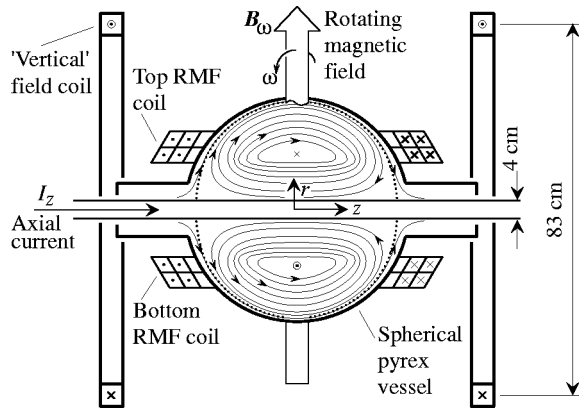


FIG. 1. The schematic of rotamak.

Figure 2 displays the time derivative signals $\dot{B}_r$ from the magnetic pickup coil at $r=23$ cm for the shots without TF and at $r=12$ cm for the discharges with TF, and their frequency spectra. The spectral structure of $\dot{B}_\theta$ is the same as that of $\dot{B}_r$, so it is not shown. In vacuum (without plasma) the $\dot{B}_r$ signal resulting directly from the RMF is a pure sinusoidal oscillation with a 500 kHz fundamental (1ω) component. From Figs. 2(a) and 2(c) it is clearly seen that the $\dot{B}_r$ signals in the shots with plasma are severely distorted from the sinusoidal waveform. From the frequency spectra, it is seen that in addition to the 500 kHz fundamental oscillation, significant high harmonic content is present.

Figure 2(b) shows that the harmonics in the shots without TF are dominated by the odd orders, while the amplitudes of the even orders are small. For instance, the amplitude of the third harmonic is much larger than the second harmonic. As the order of harmonics increases, the amplitudes of both the odd and even harmonics decrease. When TF is added, as seen in Fig. 2(d), the amplitudes of the even order harmonics become larger and comparable with the odd ones. In contrast to the case without TF, the amplitude of the

second harmonic now is larger than the third one. The harmonic amplitudes decrease monotonically as in the case without TF.

Besides the increase of the even order harmonics, another effect of adding TF is the splitting of frequency spectral peaks. To show it more clearly, the $\dot{B}_r$ spectrum around 2ω is enlarged in the inset of Fig. 2(d). There are two side-band peaks around the main peak at 1000 kHz, located on each side of the main peak. The features of splitting vary from one order to another: some orders have one side peak, some have two or more.

Figure 3 displays the radial profiles of the first three harmonics of B_r and B_θ in the cases with and without TF for the same conditions as in Fig. 2. These radial profiles are obtained by shot to shot repeatable discharges. The amplitudes of the magnetic fields are given based on the calibration of the pick up coils. For reference, the profile in vacuum is also plotted. The vacuum field is renormalized for the same current in the RMF coils as in the discharges with plasma since the current in coils is changed when the plasma is present. From Fig. 3, it is seen that the profiles are different in the two cases, and for each case they are different for the r and θ directions and depend on the harmonic order. By comparing the amplitude of the 1ω component with its value in vacuum, it is found that the RMF in the case without TF can only penetrate the edge region of plasmas. From Figs. 3(c) and 3(d), when TF is added, the penetration of RMF increases dramatically. It can now penetrate the whole volume of the plasmas, with the maximum of oscillating field being shifted to the inner region. The appearance of jump in $B_{\theta 1\omega}$ near the center and in $B_{r 1\omega}$ near the wall is similar to the previous observations.^{18,19} Because of the penetration of the RMF and the generation of J_z in the plasma,^{8,13} the amplitude of B_θ in both cases can be larger than its vacuum value as is seen in Figs. 3(b) and 3(d). In the case with TF, the B_θ is even larger than its vacuum value in most regions of the plasma with the maximum value almost double the

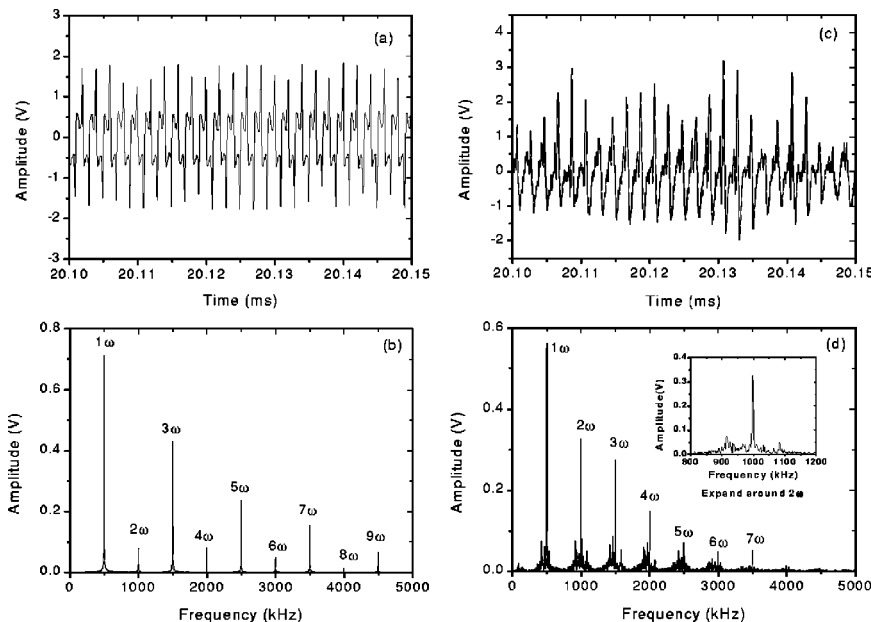


FIG. 2. The time evolution and frequency spectra of $\dot{B}_r$ without and with TF. (a) The $\dot{B}_r$ signal and (b) its frequency spectrum at $r=23$ cm without TF. (c) The $\dot{B}_r$ signal and (d) its frequency spectrum at $r=12$ cm with TF.

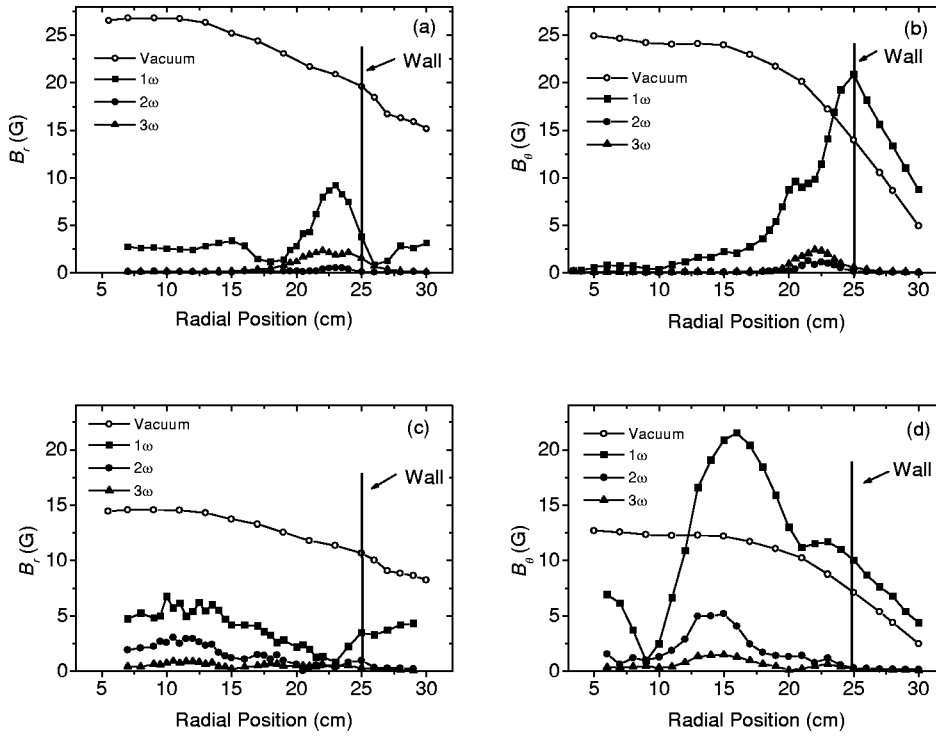


FIG. 3. Radial profiles of harmonics: (a) B_r without TF, (b) B_θ without TF, (c) B_r with TF, (d) B_θ with TF.

value in vacuum. The amplitude profiles of the 2ω and 3ω are not similar to the profile of the 1ω , but their maxima are around the peak position of the 1ω . From Fig. 3, the ratios between the maximum amplitude of harmonics and the fundamental one are $B_{r3\omega}/B_{r1\omega} \sim 26\%$, $B_{\theta3\omega}/B_{\theta1\omega} \sim 26\%$ in the case without TF, and $B_{r2\omega}/B_{r1\omega} \sim 45\%$, $B_{\theta2\omega}/B_{\theta1\omega} \sim 25\%$ in the case with TF. During a discharge, the amplitude of every harmonic is steady as long as the total plasma current is flat.

As mentioned above, the Hall term $\mathbf{J} \times \mathbf{B}$ leads to the high harmonics in the currents and fields. In the absence of toroidal field, B_r , B_θ , and J_z have odd order harmonics, while B_z and J_θ , J_r have even order harmonics. When an external toroidal field $B_{\theta0}$ is added, the even harmonics of J_r interact with $B_{\theta0}$ ($J_{r2\omega} \times B_{\theta0} \Rightarrow J_{z2\omega}$, for example) and produce even harmonics in J_z , B_r , and B_θ , as observed in the experiment, while odd harmonics of J_z interact with $B_{\theta0}$ generating odd order harmonics in J_r , J_θ , B_z . Our measurements show that in the case without TF, the amplitude of 2ω in the B_z spectrum is indeed larger than other harmonics. With TF, a large 1ω component is stimulated in B_z as predicted by the above picture and the spectral structure of B_z becomes the same as that for B_r and B_θ .

The Hall term $\mathbf{J} \times \mathbf{B}$ is a nonlinear term in Eq. (1). For further understanding of the harmonic generation in this nonlinear process, a numerical simulation has been carried out. The model used in Ref. 11 is adopted. The main equations used in our simulation are

$$\frac{\partial \mathbf{A}}{\partial t} = -\eta \mathbf{J} - \frac{1}{en} [\mathbf{J} \times \mathbf{B}],$$

$$\mathbf{B} = \nabla \times \mathbf{A},$$

$$\mathbf{J} = \frac{1}{\mu_0} \nabla \times \mathbf{B}.$$

In the model, the ions are regarded as stationary, and the pressure gradient is neglected. The equations are solved numerically in cylindrical geometry using a finite difference

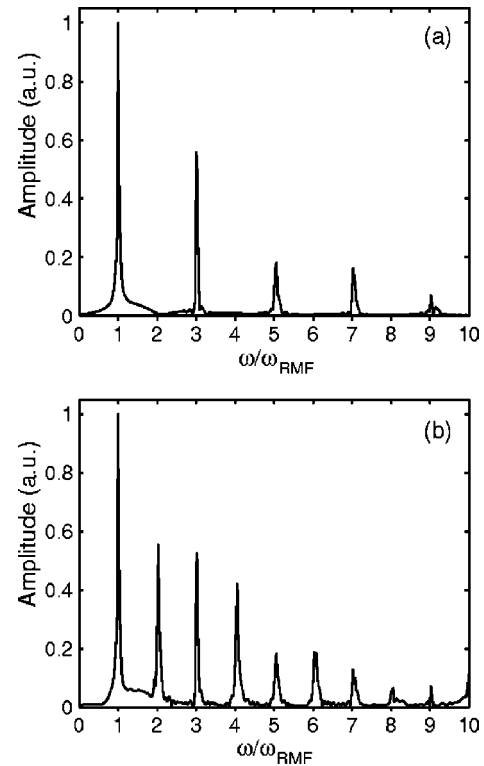


FIG. 4. The spectra of $\hat{B}_r$ from simulation: (a) without TF, (b) with TF.

mesh in the r direction and a Fourier expansion in the θ direction, assuming no axial variation ($\partial/\partial z=0$). A predictor–corrector algorithm is used in temporal iterations. Figure 4 is the spectra of $\tilde{B}_r$ in the cases with and without TF from simulations with the parameters in the experimental range. The spectral features observed in experiments are well reproduced. In the simulations, it is seen that the presence of high harmonic fields causes small oscillating ripples on the otherwise rigid-rotor profile of $J_{\theta 0}$, as also reported in Ref. 11.

In summary, the high harmonics of the rotating magnetic field in a rotamak plasma are measured and analyzed. Experimental results indicate that besides driving the steady plasma current, the RMF stimulates high harmonic fields. In the discharges without toroidal field, the odd harmonics are dominant in the B_r and B_θ frequency spectra. The largest is the third harmonic with the ratio of its amplitude to the fundamental one $B_{3\omega}/B_{1\omega} \sim 30\%$. The RMF in this case can only penetrate the edge layer of the plasma. Adding TF enhances the even order harmonics of B_r and B_θ greatly and makes the amplitude of the second order larger than that of the third with $B_{2\omega}/B_{1\omega}$ up to 45%. The TF also improves the penetration of the RMF dramatically which can penetrate the whole plasma volume.

According to the qualitative explanation and the simulation, without TF there should be only odd order harmonics in B_θ and B_r and even order in B_z . But our experiments show the presence of small even order harmonics in B_θ and B_r and odd orders in B_z . The possible reason for this is that a small toroidal field already exists in the rotamak plasma, even if

there is no external toroidal field, owing to the circulation of particles along the closed magnetic field lines.

ACKNOWLEDGMENTS

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**Appendix-D: All Publications
2004-2015
(Listing)**

APPENDIX-D: Listing of All Publications

Listing of All Publications: Plasma Research at Prairie View A&M University with Support from DE-FG02-04ER54777 (2004-2015)

- 1. Microwave experiments on Prairie View Rotamak**
R. Zhou, M. Xu, and T.S. Huang
Physics of Plasmas 22, 05450, 2015
- 2. Microwave Heating Experimental at Prairie View Rotamak**
R. Zhou, M. Xu, D. Vasquez, and T.S. Huang
American Physical Society Division of Plasma Physics Meeting 2014, abstract #UP8.023
- 3. Improved Magnetic Reconnection Experiment at FRC Device**
M. Xu, R. Zhou, D. Vasquez, and T.S. Huang
Abstract Submitted for the DPP14 Meeting of the American Physical Society DPP14-2014-001073, 2014
- 4. Improved Magnetic Reconnection Experiment at FRC Device**
M. Xu, R. Zhou, D. Vasquez, T.S. Huang
American Physical Society Division of Plasma Physics Meeting 2014, abstract #UP8.022
- 5. Discontinuity in Euler Potential in the Presence of Field Line-Aligned Currents**
T.S. Huang, D.J. Wu
Chinese Science Bulletin, 2012
- 6. High sensitivity, low noise Mirnov coil array on Prairie View Rotamak**
X. Yang, S. Houshmandyar, O. Dada, E. Reddic, and T. S. Huang
Review of Scientific Instruments, 83, 10E304, 2012
- 7. Overview of recent experimental results from the Rotamak-ST discharges**
X. Yang, E. Reddic, O. Dada, T.S. Huang
American Physical Society, 54th Annual Meeting of the APS division of Plasma Physics, 2012, abstract #GO6.014
- 8. High resolution ion Doppler spectroscopy at Prairie View Rotamak**
S. Houshmandyar, X. Yang, and R. Magee
Review of Scientific Instruments, 83, 10D506, 2012

9. Euler potentials' discontinuity in the presence of field line-aligned currents

T.S. Huang and D. Wu

SPECIAL ISSUE: Basic Plasma Processes in Solar-Terrestrial Activities April 2012
Vol.57 No.12: 1436 – 1437, 2012

10. Boost of plasma current with active magnetic field shaping coils in Rotamak discharges

X. Yang, J. Goss, D. Kalaria, and T.S. Huang

Physics of Plasmas, 18, 080703, 2011

11. Effect of toroidal magnetic field on $n = 1$ mode stability in Rotamak plasmas

X. Yang, J. Goss, D. Kalaria, and T.S. Huang

Physics of Plasmas, 18, 080708, 2011

12. Recent Development of Magnetic Field Measurements on Rotamak

X. Yang, J. Goss, D. Kalaria, and T.S. Huang

American Physical Society, 53rd Annual Meeting of the APS Division of Plasma Physics, 2011, abstract #CP9.111

13. The study of mode instability in Rotamak plasmas driven by rotating magnetic field

X. Yang, J. Goss, D. Kalaria, S. Houshmandyar and T.S. Huang

American Physical Society, 53rd Annual Meeting of the APS Division of Plasma Physics, 2011, abstract #PO5.010

14. A New time-dependent ionosphere-magnetosphere coupling model: Comparison of field aligned currents against ST5 observations

T.S. Huang, E. Romashets, G. Le, Y. Wang, and J. A. Slavin

Journal of Atmo and Solar-Ter. Phys., 72, 369, 2010

15. Experiments on Rotamak plasma equilibrium and shape control

T.S. Huang, X. Yang, Y. Wang, Y. Petrov

Phys of Plasmas, 17, 012506, 2010

16. Study of Plasma Shape Control and Current Drive

X. Yang, J. Goss, E. Sidney, A. Razzaq and T.S. Huang

American Physical Society, 52nd Annual Meeting of the APS Division of Plasma Physics, 2010, abstract #GP9.128

17. Effects of External Magnetic Field on the coupling between RF Antenna and FRC Plasma

X. Yang, J. Goss, D. Kalaria, E. Sidney and T.S. Huang
American Physical Society, 52nd Annual Meeting of the APS Division of Plasma Physics, 2010, abstract #GP9.115

18. Suppression of $n = 1$ tilt instability by magnetic shaping coils in Rotamak plasmas

T.S. Huang X. Yang, Y. Petrov
Physical Review Letters, 2009, Lett 102, 255004

19. Analysis of the active equilibrium control experiments in Rotamak

T.S. Huang X. Yang, Y. Petrov
American Physical Society, 51st Annual Meeting of the APS Division of Plasma Physics, 2009, abstract #UP8.009

20. Effect of Magnetic field configuration on $n=1$ Instabilities in Rotamak

T.S. Huang X. Yang, Y. Petrov
American Physical Society, 51st Annual Meeting of the APS Division of Plasma Physics, 2009, abstract #UP8.008

21. Poloidal magnetic flux control experiments in Rotamak

T.S. Huang X. Yang, Y. Petrov
American Physical Society, 50th Annual Meeting of the APS Division of Plasma Physics, 2008, abstract #TP6.111

22. Magnetic Field structure evolution in rotating magnetic field plasmas

T.S. Huang X. Yang, Y. Petrov
Physics of Plasmas 15, 072509, 2008

23. Comparison of Rotamak plasma discharges in spherical and cylindrical devices

T.S. Huang X. Yang, Y. Petrov
Plasmas Physics and Controlled Fusion, 50, 085020, 2008

24. Effects of Plasma Shaping Magnetic Field Coil on RMF Plasmas

T.S. Huang X. Yang, Y. Petrov
American Physical Society, 50th Annual Meeting of the APS Division of Plasma Physics, 2008, abstract #TP6.109

25. Comparison of Rotamak plasma discharges in cylindrical and spherical devices

X. Yang, Y. Petrov, T.S. Huang
Plasma Physics and Controlled Fusion, Volume 50, Number 8, 2008

26. Plasma flows around magnetic obstacles in the solar wind

E. Romashets, and S. Poedts
A&A, 475, 1093-1100, 2007

27. Modelling of Three-Dimensional Motion of Toroidal Magnetic Clouds in the Inner Heliosphere

E. Romashets, and S. Poedts, M. Vandas
A&A, 466, 357-565, 2007

28. The effect of toroidal field on the rotating magnetic field current drive in Rotamak plasmas

Z Fang-Chuan, T.S. Huang, Y. Petrov
Chinese Physics, Volume 16, Number 11, 2007

29. Use of Euler potentials for describing magnetosphere-ionosphere coupling

T.S. Huang, P. Le Sager, F.R. Toffoletto, S. Sazykin, R.W. Spiro, R.A. Wolf
J. Geophys. Res., 111, A07315, DOI: 10.1029/2005JA011558, 2006

30. Discrete compressional Alfvén eigenmode spectrum in tokamaks

N.N. Gorelenkov, E.D. Fredrickson, W.W. Heidbrink
Nuclear Fusion, Volume 46, Number 10, 2006

31. Phase space gradient driven discrete compressional Alfvén eigenmodes in tokamaks: simulations and observations

N.N. Gorelenkov, C.Z. Cheng, N.A. Crocker, E. D. Fredrickson, W. W. Heidbrink, S. Kubota, C. K. Phillips, W. A. Peebles, G. Wang. 2006

32. Principal Physics of rotating magnetic field current drive of field reversed configurations

A.L. Hoffman, H.Y. Guo, K.E. Miller, R.D. Milroy
Physics of Plasmas, 13, 012507(2006)

33. Magnetic reconnection of two FRCs driven by rotating magnetic field

Y. Petrov, X. Yang, T.S. Huang
Abstract Submitted for the DPP06 Meeting of the American Physical Society, 2006

34. Gas puffing experiments in Rotamak

Y. Petrov, X. Yang, T.S. Huang

American Physical Society, 48th Annual Meeting of the Division of Plasma Physics, October 30-November 3, 2006, abstract #CP1.079

35. Comparison of Rotamak plasmas in FRC and ST configurations

T.S. Huang, Y. Petrov, F.C. Zhong

Plasma Physics and Controlled Fusion, 47, 1517, 2005

36. Observation of the self-generated toroidal magnetic field in Rotamak

T.S. Huang, Y. Petrov, F.C. Zhong

Physics of Plasmas, 12, 082514-082514-6, DOI 10.1063/1.2007719, 2005

37. Sudden disappearances of filaments and their relation with Coronal Mass Ejections

T.S. Huang, F.R. Crawford, S. Pojoga

Bulletin of the American Astronomical Society, 36, 713, 2004

38. No increase of the interplanetary electric field since 1926

P. Lesager, L. Svalgaard

J. Geophys. Res., 109, DOI: 10.1029/2004JA010411, 2004

39. High harmonic fields in a Rotamak plasma

T.S. Huang, Y. Petrov, F. Zhong

Physics of Plasmas 11, L1 (2004)

40. Electron temperature and ion rotation features in Rotamak

T.S. Huang, Y. Petrov, Y. Luo, F. Zhong

American Physical Society, 46th Annual Meeting of the Division of Plasma Physics, 15-19 November, 2004, Savannah, GA. MEETING ID: DPP04., abstract #HP1.041

41. Effect of the toroidal magnetic field on the plasma current in Rotamak

T.S. Huang, Y. Petrov, Y. Luo, F. Zhong

American Physical Society, 46th Annual Meeting of the Division of Plasma Physics, 15-19 November, 2004, Savannah, GA. MEETING ID: DPP04., abstract #HP1.040