Overview of the Latest HT-7 Experiments

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Abstract. An overview of the HT-7 experimental progress during 2003–2004 is presented. The operational scenarios of H-mode, negative reversed shear (RS) and high \( l_i \) were investigated for quasi-steady-state high performance plasma discharges. Stationary internal transport barriers (ITBs) with normalized performance \( \beta_n H_{99} > 1–3 \) have been obtained with combined injection of lower hybrid (LH) and ion Bernstein (IB) waves for a duration of several hundred energy confinement times in weak negative reversed shear. The maximum fraction of non-inductive current was \( >99% I_p \). The increase of the total injected power up to 1 MW did not degrade the plasma confinement significantly in the RS operational scenario. Plasma performance and duration were mainly limited by two kinds of MHD instabilities and recycling. The high \( l_i \) plasma was created by fast plasma current ramp-down and sustained by central LHCD and IBW heating for a duration of \( >1 \) s with a strongly peaked electron temperature profile. The highest central electron temperature obtained was as much as
4.5 keV. Stationary improved confinement has been observed in the high \textit{l}, plasma. The longest plasma discharge, with a duration of 240 s, $T_e(0) \sim 1$ keV and a central electron density of $>0.8 \times 10^{19}$ m$^{-3}$, was achieved in 2004. A fully LHW current driven plasma without using ohmic current in the central solenoid coils was sustained for 80 s. The main limitation for the pulse length was due to the recycling, which caused an uncontrollable rise in the electron density. The poloidal large-scale $E \times B$ time-varying flows, electrostatic and magnetic Reynolds stress were directly measured in the boundary plasma of the HT-7 tokamak.

1. Introduction

To be attractive as an energy producing system, a tokamak requires high performance plasmas with high confinement, high power density (which means high $\beta$) and high stability under steady-state conditions \cite{1}. HT-7 is a medium sized tokamak with superconducting toroidal coils and water-cooled graphite limiters, which is relevant to these issues. Significant progress in developing the technological and scientific basis for steady-state operation modes of high performance plasmas has been achieved in the last few years \cite{2–4}. Since the last IAEA meeting, experiments in the HT-7 tokamak have been focused on investigating different scenarios for the confinement improved operation modes, long pulse discharges and high performance plasmas under quasi-steady-state conditions. New systems, including an LHCD launcher, cryogenic compressor, poloidal real-time control, water-cooled toroidal belt limiter, ferretic liner and several diagnostics, were installed and operated in order to meet the long pulse operation requirements.

Significant progress towards integrated advanced performance under steady-state conditions has been obtained in most of the present large devices in the last few years \cite{5–11}. High performance, indicated by the product $\beta_N*H_{\text{89}} \geq 10$, has been achieved for several tens of $\tau_E$ or $4\tau_{\text{CR}}$, limited by the neoclassical tearing mode \cite{7}. Stationary tokamak operation at the fusion gain parameter of $\beta_N*H_{\text{89}}/q_{95}^2 \approx 0.4$ has been sustained for 6.5 s, about $36\tau_E$ in DIII-D \cite{6, 7}. A high $\beta_P$ H-mode plasma with full non-inductive current drive has been sustained for 7.4 s with $\beta_N = 2.7$ \cite{9}. All these steady-state high performance plasmas demonstrated the importance of the development of improved confinement scenarios and advanced control tools. Several improved confinement scenarios, including the H-mode, high \textit{l}, mode and RS mode, were developed by IBW, off-axis LHCD or their combination in HT-7. Long pulse plasma discharges with improved confinement were obtained only in the presence of LHCD. Since the current profile in inductive and LHCD tokamak discharges is strongly linked with the electron temperature profile, active control of the localization of LHCD becomes possible by employing the interaction of LHW and IBW. Experiments utilizing the combination of LHCD and IBW heating have been performed in HT-7 to enhance plasma performance under quasi-steady-state conditions \cite{2–4, 12}.

One of the main efforts on the HT-7 superconducting tokamak is directed to long pulse discharges and the relevant physics. The issues involved are non-inductive current drive, plasma control, heat exhaust, particle removal, etc. Significant progress in steady-state physics and technology was achieved on the Tore Supra, Triam-1M and HT-7 tokamaks in the last few years \cite{13–15}. New doped graphite with a SiC gradient coating as the limiter material and ferretic steel were used to meet the requirements for long pulse plasma discharges in HT-7. In the 2003 experimental campaign, long pulse plasma discharges were obtained for a duration of $>60$ s with two water-cooled fully poloidal limiters. The duration of the plasma discharges was mainly limited by the particle recycling, which caused an uncontrollable density rise. The temperature of the limiter measured by an embedded thermal couple in the graphite tiles can be as high as $800^\circ$C, which causes strong outgassing from the limiter surface
and makes a significant contribution to the particle recycling. A new top–bottom symmetric toroidal belt limiter was installed in the recent campaign in 2004 to replace the old poloidal limiters. It has a larger surface area facing the plasma and a much higher heat exhaust capability.

Several important technical modifications have been made in the last two years. A new multijunction grill launcher of LHCD with improved protection allows higher power injection into the plasmas for a longer duration. The launched parallel refractive index $\eta_{\text{peak}}$ of LHW can be varied from 1.8 to 3.5 with FWHM of about 0.5, which provides flexibility for current density profile control. The poloidal feedback control system was upgraded with a real-time operation system, which eliminates the limitation of discharge duration of the old operation system. Several new diagnostics provided measurements of electron temperature profiles and edge plasmas, etc. These modifications allow the discharge duration to be extended up to 4 min, in which time some key issues of steady-state operation can be studied.

The paper is organized as follows. In Sections 2 and 3, the improved confinement scenarios of H-mode and high $l$ plasma are discussed. The high performance discharges with an ITB under steady-state conditions are then presented and the limitations for further increases of the plasma performance and duration are discussed in Section 4. The control and operation of long pulse discharges are discussed in Section 5, followed by a short introduction of edge physics in Section 6. A summary and conclusion are given in Section 7.

2. H-mode

2.1 H-mode by IBW

IBW heating was investigated in the HT-7 superconducting tokamak operating with deuterium plasmas. Direct electron heating via electron Landau damping from IBW has been observed [16, 17]. In the case of IBW heating at 27 MHz with a toroidal field strength of 1.8–2 T, both global and localized electron heating were obtained by locating the resonant layer in the plasma far from the edge region, where only the second harmonic deuterium cyclotron resonant layer is in the plasma [18]. This operation mode demonstrates the possibility of IBW as a tool for controlling the electron pressure profile, which is needed for an advanced tokamak scenario.

In the case of IBW heating at 30 MHz, a significant improvement of the particle confinement was observed. It was found that a minimum IBW power of 120 kW was required for attaining a significant improvement of the particle confinement. The particle confinement was improved by accompanying a more peaked electron density with an increase in the injected IBW power. In this experiment, the IBW mode conversion layer was located in the SOL. The local deposition of RF power in this layer markedly modified the electron temperature profile in the SOL, resulting in the profile of radial electric field shown in Fig. 1. The shearing rate $\omega_{e,B}$ of a turbulent structure in the IBW-heated phase exceeds the ambient turbulence decorrelation rate $\Delta\omega_B$, for the drift-wave-like turbulence. This strong shear decorrelation effect produced a distinct weak turbulence regime in the boundary plasma and can account for the improved particle confinement, as discussed elsewhere [19].
The electron heat diffusivity coefficient, derived from sawtooth heat pulse analysis, decreased with increasing IBW power up to 220 kW for the same target plasma [18]. Recent transport analysis by ONETWO shows that both electron and ion heat diffusivity coefficients are reduced compared with the ohmic target plasma in the outer half of the plasma. Figure 2 shows the ion heat diffusivity coefficients in the ohmic target plasma and the IBW heated phase. This leads to a global improved energy confinement at higher injected IBW power.

2.2 H-mode by LHW

Off-axis LHCD was explored not only for sustaining the plasma current, but also for controlling the profile of plasma current density and enhancing the confinement. In the previous experiments, HT-7 was operated at $I_p > 200$ kA, $\bar{n}_e \sim 1 \times 10^{19} / \text{m}^3$, with a central electron temperature $T_e(0)$ in excess of 1.2 keV for an ohmic target plasma [20, 21]. LHW damped its power by electron Landau damping in a single pass regime and led to the off-axis driven current profile. A weak negative shear in the plasma current profile was formed and an ITB at the footprint of the minimum $q$.

In the present experiment, another scenario to realize off-axis current drive by LHW was operating plasmas by proper optimization of the LHCD launched spectra. Typical results are shown in Fig. 3 for a target plasma of $I_p = 150$ kA, $\bar{n}_e \sim 1 \times 10^{19} / \text{m}^3$ and launched $n_0 \text{peak} = 3.1$. Improvement of the plasma confinement by LHW injection was indicated by a slight increase in $\bar{n}_e$ and a drop in the $D_\alpha$/$H_\alpha$ emission and increase in the energy confinement time during the LHCD phase. The profile of HX radiation (HXR) normalized by electron density $I_{\text{HX}}(r)/n_e(r)$ at energy 40–60 keV given in Fig. 4 shows a clear off-axis LHCD. This result is obtained by the Abel inversion of the line integrated HXR in the selected energy range. Off-axis current drive broadened the global current density profile, as indicated by a drop in plasma inductance. Measurements by a Langmuir probe array at the limiter radius showed a decrease of the turbulent particle flux accompanied by the formation of a positive radial electric field and suppression of the fluctuation levels in the floating potentials and ion saturated current [21]. In the shot shown in Fig. 3, the non-inductive current fraction estimated from the loop voltage was about 65%. The plasma in such scenarios with a
normalized product of $\tilde{H}_89^*\beta_N \sim 1.2$ was sustained for 7 s at $P_{LHW} \sim 300$ kW, $\pi_e \sim 1.5 \times 10^{19}/m^3$ and $I_p = 120$ kA, where the factor of $H_{89}$ was about 1.2 and $\beta_N$ around unity [2]. This is longer than $400\tau_E$ and $40\tau_{CR}$.

3. High $l_i$ Mode

Theories have predicted that a stabilizing effect from magnetic shear on both ideal high n ballooning modes and electrostatic microinstabilities would require high positive shear and low or negative shear over a substantial portion of the plasma, corresponding to the current profile in the high $l_i$ and NCS regime [22]. Although the high $l_i$ scenario is attractive for the advanced tokamak concept, its compatibility with the steady state requirements is still questionable [23]. High positive shear is obtained when the plasma current profile is strongly peaked, which gives a high internal plasma inductance $l_i$. High $l_i$ plasmas were created by negative current ramp in the present experiments. Tore Supra has produced improved confinement in high $l_i$ LHCD steady state plasmas by this method [24]. Fast plasma current ramp-down at a rate of $-(0.5–1.2)$ MA/s has been used to create different high $l_i$ target plasmas in HT-7 [25]. The LHW pulse was applied before or just before the plasma current was ramped down. IBW heating was applied to further increase the plasma beta and improve the plasma confinement. With such a scenario, a steady state value of $l_i > 1.5$ was obtained for a duration of several current diffusion times, which is nearly quasi-steady state.

Information on the current profile indicated by the plasma internal inductance $l_i$ and global energy confinement times was deduced from both magnetic probe and diamagnetic measurements. The measurements of the interferometer, Thomson scattering and SX-PHA provide the kinetic electron energy. A plasma inductance of $l_i > 1.5$ was achieved and sustained by central LHCD and IBW heating with equilibrium density and temperature profiles. In such discharges, global electron heating was observed, but the electron temperature profile was strongly peaked. The highest central electron temperature, up to 4.5 keV, at a line averaged density of $2.2 \times 10^{19}$ m$^{-3}$, was obtained by applying 400 kW LHW at $N|| = 2.3$ and 200 kW IBW just before $I_p$ was ramped down from 180 kA to a 100 kA plateau at a rate of $-0.8$ MA/s, as shown in Fig. 5. The ion temperature in such discharges was 1.5 keV. The non-inductive current fraction was about 80% of $I_p$. A stationary improved confinement has been observed in such high $l_i$ plasmas. No impurity accumulation was observed during the improved confinement phase.

These experiments have been carried out at central line averaged densities between $1.0 \times 10^{19}$ and $3.0 \times 10^{19}$ m$^{-3}$. The total injected power $P_{LHW}$ and $P_{IBW}$ ranged from 250 to 800 kW. The stored energy increases nearly linearly with the injected LHW power at constant density.
and current ramp rate. The global confinement time at lower $P_{LHW}$ and lower density is close to the ITER-89P scaling, but is higher than the ITER-89P scaling at higher $P_{LHW}$ and density. The effects on LHW power and density could be explained assuming that the absorbed LHW power increases with density and higher power improves the particle confinement more efficiently. Indeed, the particle confinement improvement was observed at the higher injected LHW power with slightly increased and peaked density.

The energy confinement time is increased to a level above the ITER-89P scaling when IBW is applied. It is found that IBW heating can significantly increase the plasma beta at the same total injected power if part of the LHW power is replaced by IBW. This effect could be attributed partly to the synergy of LHW and IBW and partly to the particle confinement by the IBW heating. The synergetic effect can improve the power deposition for both LHW and IBW [2, 12]. The particle confinement improvement obtained by applying IBW was indicated by the increase of the density and the drop of the $H_\alpha$ emission.

The current profile effect on the global confinement has been investigated by changing the plasma internal inductance, $l_i$. This was realized at different current ramp rates and launched LHW parallel wave index $N_{||}$. An increase in the energy confinement time with $l_i$ in the range 1.3–1.8 is observed at a constant line averaged electron density of $1.5 \times 10^{19}$ and a total injected power of 450 kW. The result is shown in Fig. 6. IBW heating improved the plasma confinement at the same total injected power if part of the LHW power was replaced by IBW. No MHD activity was detected in the stationary phase, and small sawteeth only existed during the ramp-down phase. The small sawtooth activity, if it existed in the target plasma, could be suppressed after the current ramp-down. This observation is very similar to the results in Tore Supra [24]. A possible reason is that the safety factor $q(r)$ was greater than unity everywhere in the plasmas in the stationary phase, although there was no direct measurement of the current density profile.

4. RS Mode

HT-7 has successfully realized off-axis LHCD by operating plasmas at a high target electron temperature and with higher $n_{||}$, which may cause strong Landau damping of the LHW in a single pass regime and lead to the off-axis driven current profile. A weak negative shear in the plasma current profile with an ITB at the footprint of the minimum $q$ was formed in the plasmas with $I_p > 200$ kA [20, 21]. However, operation at such high current is marginal on HT-7, owing to the limited toroidal magnetic strength. Its confinement could be good, as indicated by the factor $H_{90} \sim 1.6$ in such an off-axis driven mode, but the plasma $\beta$ is low. The normalized $\beta_N$ was never higher than 0.6, owing to the lower electron densities needed for efficient LHCD.

The combination of IBW heating and LHCD provides an alternative way to create an off-axis fast electron current channel in a well-defined region. Theory and simulation suggested that IBW could be used in conjunction with LHW to aid the localization of the non-inductive current generated in the LHCD regime and to help fill the LHCD spectral gap for high values of $n_{||}$ [26, 27]. Experiments on HT-7 confirmed that the properties of IBWs in controlling temperature and density profiles can be integrated into the LHCD plasmas to improve the local current drive efficiency and change the local electron pressure profile [2, 4, 12]. This feature can be used to tailor the current density profile, and it could also be a way to improve the plasma performance through maximizing the volume of high confinement plasma by proper selection of plasma and wave parameters. Experiments in HT-7 demonstrated that the features of off-axis heating by IBW are capable of enhancing the LHCD plasma performance through extension of the high performance volume via LHW and IBW synergy.
FIG. 7. Stored energy versus total injected LHW and IBW powers. The predictions from ITER-89 scaling are plotted as solid lines for two different effective masses: \( m = 2 \) and \( m = 1.7 \) for \( \frac{H}{(H + D)} \sim 30\% \) plasmas.

FIG. 8. Energy confinement time versus total injected LHW and IBW powers. The predictions from ITER-89 scaling are plotted as solid lines for two different effective masses: \( m = 2 \) and \( m = 1.7 \) for \( \frac{H}{(H + D)} \sim 30\% \) plasmas.

FIG. 9. A combined LHW and IBW plasma discharge was sustained at \( \beta_n H_{99} > 2 \) for 4.6 s, which is about 235 energy confinement times.

HT-7 has produced a variety of discharges with the normalized performance \( \beta_n H_{99} > 1–3 \) with weak negative reversed shear by a combination of LHCD and IBW heating [2]. Recent experiments found that the increase of total injected power up to 1 MW did not degrade the plasma confinement significantly in such an operational scenario. For the plasmas (\( I_p \sim 150 \) kA, \( \bar{n}_e \sim 1.5 \times 10^{19} / \text{m}^3 \), \( n_{\text{peak}} \text{(LHW)} = 2.3 \), \( \bar{f}(\text{IBW}) = 27 \) MHz), the incremental energy content increases linearly with additional injected power up to 0.8 MW and saturated at 15 kJ for further increased power, as shown in Fig. 7. At higher injected power, in particular at higher IBW power, the increased impurity levels caused strong radiation and prevented the further increase of the stored energy in the plasma. The ITER-89P scaling is plotted in the same figure for two effective masses. The \( m = 2 \) is for pure deuterium plasmas. The \( m = 1.7 \) was derived from the isotopic ratio \( n_H/(n_H + n_D) \sim 30\% \), deduced from \( H_\alpha \) and \( D_\alpha \) spectroscopic measurements in these discharges.

The global energy confinement time is plotted versus the total power for the discharges performed at \( I_p = 120–150 \) kA and \( \bar{n}_e \sim 1.5 \times 10^{19} / \text{m}^3 \) in Fig. 8. The best plasma performance is close to \( H_{99} \sim 2 \) for the medium additional IBW and LHW powers with formation of a weak negative reversed shear. The fraction of non-inductive plasma current was larger than 80\% in such discharges, to which bootstrap current contributed considerably. In the discharges with lower injected power, the plasma performance was relatively low, although improved confinement was also observed in these plasmas. It is found that neither LHCD nor the synergy between IBW and LHCD was sufficient to create negative or weak positive magnetic shear in these operation regimes, which may be a possible reason for the limited plasma performance. At higher injected IBW power, the increase of the impurity level caused stronger radiation power loss and slightly degraded the plasma confinement, whereas the internal \( m = 1 \) kink mode, which may be correlated with the unfavourable fast electron profile produced by LHCD, limited the performance in high beta plasmas [2, 28].

Through proper optimization of operation, in particular through the choice of a strategy to avoid MHD activity, high performance discharges under quasi-steady state have been realized in HT-7 by applying RF waves in the early phase of discharges and extending the RF pulse as long as possible. Figure 9 shows such an optimized discharge utilizing the synergy between LHW and IBW. In this shot, the plasma current was 120 kA, \( B_T = 1.7 \) T, \( \bar{n}_e(0) = 1.6 \times 10^{19} / \text{m}^3 \), \( P_{\text{LHW}} \sim 400 \) kW and \( P_{\text{IBW}} \sim 180 \) kW. The performance indicated by \( \beta_n H_{99} \sim 2.3 \) was sustained for 4.3 s, about \( 220 \tau_E \) and longer than \( 20 \tau_{\text{CR}} \), in which all plasma parameters and their profile reached a stationary state. The analysis of these discharges by EFIT and
ONETWO shows that the current profile had a negative shear in a stationary state. The current density profile at stationary state (2 s) displays a negative shear configuration, shown in Fig. 10. Clearly an ITB was formed at around the footprint of the minimum q. More than 80% of the plasma current was sustained non-inductively by LHCD and bootstrap current. A transport barrier at the edge of the plasma was also observed from the corresponding Langmuir probe measurements, which means an H-mode edge in this reversed shear operational mode.

5. Long Pulse Plasma Discharges

![Graph](image1)

**FIG. 10.** Profiles of plasma pressure and safety factor (top) and ion heat diffusion coefficient for the shot in Fig. 8 at 2 s. The neoclassical ion heat diffusion coefficient is shown as a dashed curve in the bottom plot.

![Graph](image2)

**FIG. 11.** A fully current driven long pulse plasma discharge was realized by feedback control of the magnetic swing flux by the LHW power. From top to bottom: plasma current, LHW power, magnetic swing flux in V*S, loop voltage, central electron density and temperature.

![Graph](image3)

**FIG. 12.** Four minute long pulse plasma discharge. Left from top to bottom: \(I_p, n_e(0), \) horizontal plasma position, tile temperature in the top limiter, global particle recycling coefficient. Right from top to bottom: magnetic swing flux in V*S, \(T_e(0), \) vertical plasma position, tile temperature in the bottom limiter, \(\Sigma_{Sg} \) integrated gas filling.

Long pulse discharges were performed, using three control loops. The plasma current and position were feedback controlled by the ohmic poloidal system. The central line averaged electron density was controlled by feedback control of deuterium gas injection using a pulsed piezoelectric valve. The magnetic swing flux of the transformer was feedback controlled by the LHW power. This is equivalent to controlling the loop voltage for full current drive. Figure 10 shows such a plasma discharge, in which the magnetic swing flux of the transformer (Fig. 11) was controlled to be constant for most of the discharge and the loop voltage was kept at zero (Fig. 11). A reproducible long pulse discharge with \(T_e(0) \sim 1\) keV and central electron density \(n_e(0) \sim 0.8 \times 10^{19} \text{ m}^{-3} \) has been obtained with a duration of >200 s under this operation mode. The longest plasma discharge was 240 s, as shown in Fig. 12. Figure 13 shows the electron temperature profile at 100 s. Such discharges can qualifiy the new PFCs, power and particle injection and exhaust capabilities, diagnostics and feedback control loops, etc. The main limitation for the pulse length was due to the recycling, which caused an uncontrollable rise in the electron density. This situation is shown by the vertical dash-dot line in Fig. 12 at about 220 s.

The wall saturation and refresh process was firstly observed only in such long pulse discharges with a duration of longer than 200 s. The deuterium gas was supplied by feedback control to keep a constant central line averaged electron density. The curves at the bottom of Fig. 12 show the global recycling coefficient \(R\) (left axis) and the temporal trace of time integration of the gas supply (right axis). The gas feed was automatically stopped during the
discharge by the feedback control of gas supply when the global recycling coefficient $R$ reached unity or above. At about 180 s, $R$ decreased again to below unity and gas was again supplied. In the same period, the electron density was decreased dramatically, in spite of the feedback control of the electron density and gas supply, which means that the wall repeats a process of being saturated and refreshed. This indicates that ultra-long-pulse discharges are required for the investigation of the wall equilibrium, which is one of the key issues for steady state tokamak operation.

Control of the plasma position is another key issue for achieving steady state plasma discharges. The plasma position was preset and then feedback controlled in the present experiments. An incorrect position caused strong plasma–wall interaction and led to a significant rise in the limiter temperature, which accounted for the strong outgassing and uncontrollable density rise. During the discharge shown in Fig. 12, the temperature of one tile on the bottom limiter rose quickly to over 350°C at about 220 s, when the vertical plasma position reached about –0.3 cm. The continuous rise of the limiter temperature caused an uncontrollable density rise and ultimately led to the termination of the discharge at 4 min. While the temperatures at all other measurement locations on both top and bottom limiters were kept below 200°C during the discharge, this indicates the importance of fine alignment for all plasma facing components. It also shows that new feedback loops to alleviate plasma–wall interaction are needed to include information on the temperature of the plasma facing components.

A new operation mode with feedback control of plasma current and density but constant LHW power created over-current drive ($V_p < 0$) and recharged the transformer. The current in the central solenoid was switched off when the transformer was reverse saturated. In this case, the plasma current was fully sustained by LHCD, as shown in Fig. 14. The longest duration of such plasma discharges without the central solenoid current in this operational mode was 80 s. In all these discharges, the measurements of the HXR emission profiles show the central power deposition of the LHW. The electron density/temperature profiles and plasma inductance were not obviously different from those in partially LHCD sustained discharges, which means a negligible contribution of the toroidal electric field to the plasma behavior.

6. Edge Plasma Physics

The radial profiles of electrostatic and magnetic Reynolds stress have been measured in the edge region of low beta plasmas in the HT-7 tokamak using two types of triple-tip-array Langmuir probes and an insertable magnetic probe. A radial gradient of magnetic Reynolds
stress was observed close to the velocity shear layer location; however, its contribution to driving the poloidal flows is small compared with the electrostatic component. A comparison of the profiles between the radial gradient of electrostatic Reynolds stress and the neoclassical damping of poloidal velocity gives experimental evidence that the electrostatic turbulence-induced Reynolds stress might be the dominant mechanism to sustain the poloidal flow shear at the plasma edge in steady state. Details have been discussed in Ref. [29].

7. Summary and Conclusion

In the 2003 and 2004 experimental campaigns, we focused on long pulse plasma discharges and high performance plasmas under quasi-steady-state conditions in the HT-7 tokamak. Different scenarios of the improved confinement of plasmas were investigated. IBW heating at 30 MHz produced the typical edge H-mode plasmas. Transport in both the electron and the ion channel was reduced in the outer half of the plasma. Although IBW heating alone cannot sustain the plasma current in steady state operation, its features in controlling the electron pressure profile can be applied to LHCD plasmas to enhance the plasma performance. H-mode plasmas have been produced by off-axis LHCD, which was realized by selecting a larger launched $n_{\text{peak}}$ of LHW at 3.1. The status of improved confinement in such plasma discharges has been sustained at the normalized performance of $H_{99} > 1.2$ and $\beta_N$ around unity for nearly 8 s, which is longer than 400$\tau_E$. The negative plasma current ramp created a high $\ell_i$ target plasma. This high $\ell_i$ plasma can be sustained by LHCD for a duration much longer than the energy confinement and current diffusion times. A stationary improved confinement has been observed in such high $\ell_i$ plasmas. An increase in the energy confinement time with $\ell_i$ in the range of 1.2–1.7 is observed at constant line averaged electron density and injected power.

A combination of LHCD and IBW heating can produce reversed shear in current density profiles. The features of IBW in controlling the electron pressure profile were integrated into LHCD plasmas to tailor the current density profile and avoid MHD instability and were optimized to achieve high performance plasmas. Significant progress in achieving high performance discharges under quasi-steady-state conditions in the HT-7 superconducting tokamak has been realized in such RS operation scenarios. The normalized performance indicated by the product $\beta_N H_{99} > 2.2$ was achieved for $>220\tau_E$ or $>20\tau_{CR}$. The fraction of non-inductive plasma current was larger than 80% in such discharges with a considerable bootstrap current contribution. The current profile had a steady-state negative shear configuration with a stationary ITB formed at the footprint of the minimum $q$. The increase of the total injected power up to 1 MW did not degrade the plasma confinement significantly in the RS operational scenario. Still higher plasma performance in this operation mode was limited by MHD instabilities, recycling and the increased impurity levels at higher injected power.

After installation of the new top–bottom symmetric belt limiter, the capabilities for heat load exhaust and particle removal were significantly enhanced. The new, modified LHCD launcher allowed long pulse power injection. With these technical improvements, a reproducible long pulse discharge with $T_e(0) \sim 1$ keV and central electron density $n_e(0) \sim 0.8 \times 10^{19} \text{ m}^{-3}$ has been obtained with a duration of $>200$ s. A new operation mode with over-current drive ($V_p < 0$) and without use of central solenoid current has been demonstrated fully sustained by LHCD. The longest discharge in this operational mode was sustained for 80 s.

Detailed measurements of all quantities in the poloidal momentum balance indicate that the damping of poloidal flows is balanced by an accretion due to the radial gradient of electrostatic Reynolds stress, which sustains the equilibrium sheared flow structure in a steady state. It is suggested that the electrostatic turbulence-induced Reynolds stress might be the
dominant mechanism to generate the poloidal flow shear at the plasma edge.

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