Overview of the last HT-7 experiments

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Abstract: An overview of the HT-7 superconducting experimental progress during 2003-2004 is presented. The operational scenarios of H-mode, negative reversed shear and high l_i were investigated for quasi steady-state high performance plasma discharges. Stationary internal transport barriers (ITBs) with the normalized performance beta-N*H_89 > 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 the weak negative reversed shear. The maximum fraction of the non-inductive current was > 90% Ip. Increase of total injection power up to 1 MW did not degrade the plasma confinement significantly in the RS operational scenario. Plasma performance and duration was mainly limited by two kinds of MHD instabilities and the 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 the strongly peaked electron temperature profile. The highest central electron temperature up to 4.5 keV has been obtained. A stationary improved confinement has been observed in the high l_i plasma. The longest plasma discharge, with a duration of 240 seconds, $T_e(0) \sim 1$ keV and central electron density $>0.8\times10^{19}\,m^{-3}$ has been 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 un-controllable rise of the electron density. The poloidal large-scale E×B time-varying flows, electrostatic and magnetic Reynolds stress were directly measured in the boundary plasma of the HT-7 tokamak.

1. **Introduction**

An attractive tokamak as an energy producing system requires high performance plasmas with high confinement, high power density (which means high β) and high stability under steady-state condition [1]. HT-7 is a medium sized tokamak with superconducting toroidal coils and water-cooled graphite limiters, which is positioned at these issues. Significant progress in developing the technologic and scientific basis for steady-state operation modes of high performance plasmas has been achieved in the last few years [2-4]. Since the last IAEA meeting, experiments in the HT-7 tokamak were focused on investigating different scenarios for the confinement improved operation modes, long pulse discharges and high performance plasma under quasi-steady-state condition. New systems including LHCD launcher, cryogenic compressor, poloidal real-time control, water cooled toroidal belt limiter, ferretic liner and several diagnostics were installed and operated in accommodation with long pulse operation requirements.

Significant progress towards integrated advanced performance under steady-state condition was obtained in most of present-day's large devices in last few years [5-11]. High performance indicated by the product of $\beta_N{*}H_{89}\geq10$ has been achieved for several tens τ_E or $4\tau_E$ limited by the neoclassical tearing mode [7]. A stationary tokamak operation at the fusion gain parameter of $\beta_N{*}H_{89}/q_{95}^2\approx0.4$ has been sustained for 6.5s, about $36\tau_E$ in DIII-D[6,7] A high β_P H-mode plasma with full non-inductive current drive has been sustained for 7.4s with β_N=2.7 [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-l_i 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 presence of LHCD. Since the current profile in inductive and LHCD tokamak discharges is strongly linked with the electron temperature profile, an active control of the localization of LHCD becomes possible by employing the interaction of LHW and IBW. Experiments by utilizing the combination of LHCD and IBW heating have been performed in HT-7 to enhance plasma performance under quasi-steady state condition [2-4,12].

One of The main efforts of the HT-7 superconducting tokamak is directed to long pulse discharges and relevant physics. The problems involved are non-inductive current drive, plasma control, heat exhaust, particle removal, etc. Significant progress in steady-state physics and technology were achieved on the Tore-Supra, Triam-1M and HT-7 tokamaks in the last few years [13-15]. New doped graphite with a SiC gradient coating as the limiter material and ferretic steel was used to accommodate the requirements for long pulse plasma discharges in HT-7. In 2003 experimental campaign, long pulse plasma discharges have been obtained for duration > 60 seconds with two water-cooled fully poloidal limiters. The duration of the plasma discharges was mainly limited by the particle recycling, which caused uncontrollable density rise. The temperature of the limiter measured by embedded thermal couple in the graphite tiles can be as high as 800 ^{o}C, which causes the strong out gassing from the limiter surface and significant contribution in 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 larger surface area facing the plasma and much higher heat exhausting capability.

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

The paper is organized as follows. In section 2 and 3, the improved confinement scenarios of H-mode and high l_i plasma are discussed. The high performance discharges with an ITB under steady-state condition 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 given in section 5 followed by short introduction of edge physics in section 6. The end is the summary and conclusion in section 7.

2. H-mode

2.1 H-mode by IBW

IBW heating was investigated in the HT-7 super-conducting tokamak operating with deuterium plasma. Direct electron heating via electron Landau damping from IBW has been observed [16,17]. In the case of IBW heating at 27 MHz with toroidal field strength of 1.8 to2 T, both global and localized electron heating were obtained by locating the resonant layer in 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 in controlling the electron pressure profile, which is needed for an advanced tokamak scenario.

In the case of IBW heating at 30 MHz, 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 increasing 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, consequently, the profile of radial electric field as shown in Fig.1. The shearing rate $\omega_{E\times B}$ of a turbulent structure in the IBW-heated phase exceeds the ambient turbulence decorrelation rate $\Delta\omega_D$, 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].

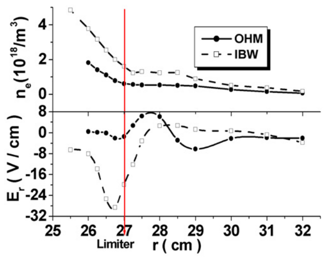

Fig.1 Comparison of the edge n_e (top) and E_r in ohmic and IBW heated plasmas.

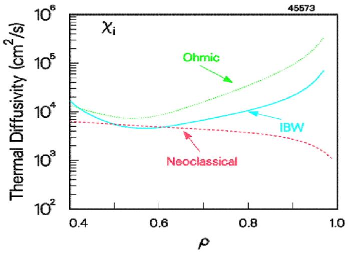

Fig.2 Comparison of the ion heat diffusion coefficients in ohmic and IBW heated plasmas.

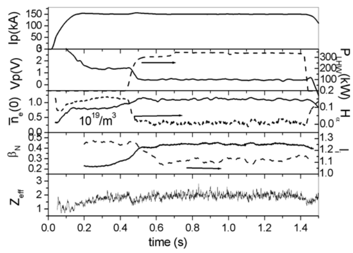

Fig.3 An off-axis LHCD produced typical edge H-mode plasma. Left from top to bottom:I_p, loop voltage, central line averaged n_n, normalized beta, Z_{eff}. Right from top to bottom: P_{LHW}, intensity of H_α/D_α emission and plasma inductance.

The electron heat diffusive coefficient, which derived from sawtooth heat pulse analysis, decreased with increasing IBW power up to 220kW for the same target plasma [18]. Recent transport analysis by Onetwo shows that both electron and ion heat diffusive coefficients are reduced compared with the ohmic target plasma in the half outer plasma region. Fig. 2 shows the ion heat diffusive 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

The off-axis LHCD was explored not only to sustain the plasma current, but also to control the profile of plasma current density and enhance the confinement. In the previous experiments, HT-7 was operated at I_P > 200 kA, $\bar{n}_e \sim 1\times10^{19}/m^3$ with the central electron temperature $T_e(0)$ in excess of 1.2 keV as ohmic target plasma [20,21]. LHW damped its power by electron Landau Damping (ELD) in a single pass regime and led to the off-axis driven current profile. A weak negative shear in plasma current profile was formed and an ITB at the footprint of the minimum q.

Another scenario to realize off-axis current drive by lower hybrid wave was operating plasmas by proper optimization of the LHCD launched spectra in the present experiment. Typical results are shown in Fig. 3 for a target plasma of I_p=150 kA, $\bar{n}_e \sim 1\times10^{19}/m^3$ and the launched $n_{||}^{peak}$ = 3.1. Improvement of the plasma confinement by LHW injection were indicated by a slightly increase of $\bar{n}_e$ and a drop of the D_α/ H_α emission and increase of the energy confinement time during the LHCD phase. The normalized HX radiation (HXR) profiles by electron density $I_{HX}(r)/n_e(r)$ at energy of 40-60 keV in Fig.4 shows a clear off-axis LHCD. This result is obtained by the Abel inversion of the line integrated HX radiations in the selected energy range. Off-axis current drive broadened the global current density profile as indicated by dropped plasma inductance. Measurements by a Langmuir probe array at the limiter radius showed a decrease of the turbulent particle flux accompanying with formation of the 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, fraction of the non-inductive current was about 65% estimated from loop voltage. The plasma in such scenarios with normalized product of $H_{89}*\beta_N \sim 1.2$ was sustained for 7 s at $P_{LHW}\sim 300$kW, $\bar{n}_e \sim 1.5\times 10^{19}/m^3$ and I_p=120 kA, where the factor of H_{89} was about 1.2 and β_N around unity [2]. This is longer than 400 τ_E and 40 τ_{CR}.

3. High l_i mode

The theories predicted that stabilizing effect from magnetic shear on both idea high n ballooning modes and electrostatic micro-instabilities require high positive shear and low or negative shear over a substantial portion of the plasma, which correspond the current profile in the high l_i and NCS regime [22]. Although the high l_i scenario is attractive for the advanced tokamak conception, its compatibility with the steady state requirements is still questionable [23]. High positive shear is obtained when plasma current profile is strongly peaked, which gives a high internal plasma inductance l_i. High l_i plasmas was 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 have been used to create different high l_i target plasmas at a rate of –(0.5-1.2) MA/s in HT-7 [25]. The LHW pulse was applied before or just before ramping down the plasma current. The IBW heating was applied to further increase the plasma beta and improve the plasma confinement. With such a scenario, the steady state value of l_i>1.5 was obtained for a duration of several current diffusion times, which is nearly quasi steady state.

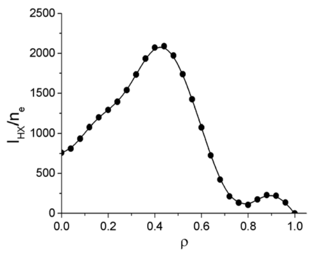

Fig.4 Profile of the hard-x ray emission normalized by the electron density.

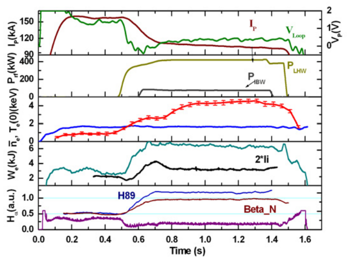

Fig.5 A current ramped down plasma discharges sustained by LHCD.

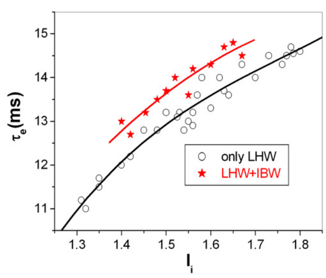

Fig.6 The energy confinement time against l_i from the discharges with the same current total injection power.

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

These experiments have been carried out at central line averaged densities between 1.0 × 10^{19} and 3.0 × 10^{19} m^{-3}. The total injected power P_{LHW} and P_{IBW} were ranged from 250 kW to 800 kW. The storage energy increases nearly 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 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 the level above the ITER-89P scaling when IBW was applied. It is found that the IBW heating can significantly increase the plasma beta at the same total injection power if part of LHW power was replaced by IBW. This effect could be attributed partially to the synergy of LHW and IBW and partially to the particle confinement by the IBW heating. The synergetic effect can improve the power deposition both for LHW and IBW [2,12]. The particle confinement improvement by applying IBW was indicated by the increase of the density and drop of the Ha emission.

The current profile effect on the global confinement has been investigated through changed of the plasma internal inductance, l_i. This was realized at different current ramp rate and the launched LHW parallel wave index $N_{||}$. An increase of the energy confinement time with l_i in the range of 1.3-1.8 is observed at the constant line averaged electron density of 1.5 × 10^{19} and the total injected power of 450 kW. The result is shown in Fig.6. The IBW heating can improve the plasma confinement at the same total injection power if part of 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 sawteeth activity if it existed in the target plasma can be killed after the current ramp down. This observation is very similar to the results in Tore-Supra [24]. Possible reason is that the safety factor q(r) was greater than unity everywhere in the plasmas in the stationary status, although there was no direct measurement of the current density profile.

4. RS-mode

HT-7 has successfully realized off-axis lower hybrid current drive by operating plasmas at high target electron temperature and with higher $n_{||}$, which may cause strong Landau damping of 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>200kA [20,21]. However operation at such high current is marginal on HT-7 due to the limited toroidal magnetic strength. Its confinement could be good as indicated by the factor of H_{89}~1.6 in such an off-axis driven mode, but the plasma beta β is low. The normalized β_N was never higher than 0.6 due to the lower electron densities needed for efficient LHCD.

Combination of IBW heating and LHCD provides an alternative way to create the off-axis fast electron current channel in the well-defined region. Theory and simulation suggested that IBWs can be used in conjunction with LHWs to aid the localization of the non-inductive current generated in the regime of LHCD and to help filling 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 local electron pressure profile [2,4,12]. This feature can be used to tailoring the current density profile on one hand. It could be also 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 features of off-axis heating by IBW are capable to enhance the LHCD plasma performance through extension of the high performance volume via LHW and IBW synergy.

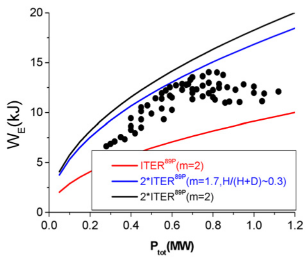

Fig.7 Storage energy versus total injected LHW and IBW powers. The prediction from ITER-89 scaling are plotted as solid lines for two different effective mass: m=2 and m=1.7 for H/(H+D)~30% plasmas.

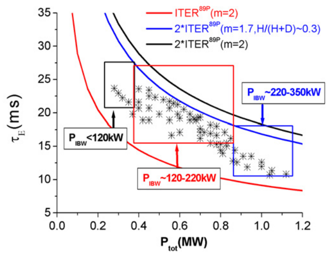

Fig.8 Energy confinement time versus total injected LHW and IBW powers. The predictions from ITER-89 scaling are plotted as solid lines for two different effective mass: m=2 and m=1.7 for H/(H+D)~30% plasmas.

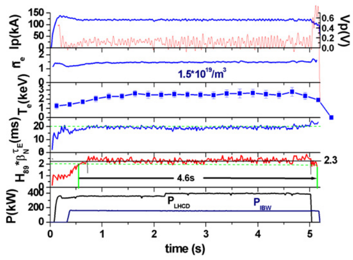

*Fig.9 A LHW and IBW combined plasma discharge was sustained at β_N*H_{89}>2 for 4.6s, which is about 235 energy confinement times.*

HT-7 has produced a variety of discharges with the normalized performance $\beta_N H_{89} > 1\sim3$ in the weak negative reversed shear by combination of LHCD and IBW heating [2]. Recent experiments found that increase of total injection power up to 1 MW did not degrade the plasma confinement considerably in such an operational scenario. For the plasmas (I_P~150kA, $\bar{n}_e \sim 1.5\times10^{19}$ /m^3, $n_{\|}^{peak}$(LHW)=2.3, f(IBW)=27MHz), the incremental energy content increases linearly with additional injected power up to 0.8 MW and saturated at 15 kJ for further higher power as shown in Fig.7. At higher injection power, particularly, higher IBW power, the increased impurity levels caused strong radiation and prevented the further increase of the store 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 isotopic ratio $n_H/(n_H+n_D)$~30%, deduced from H_α and D_α spectroscopic measurements in these discharges.

The global energy confinement time is plotted versus the total power for the discharges performed at I_P=120~150kA and $\bar{n}_e \sim 1.5\times10^{19}$/m^3 in Fig.8. The best plasma performance is close to H_{89}~2 in 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, which bootstrap current contributed considerably. In the discharges with lower injection power, the plasma performance is relative low, although improved confinement was also observed in these plasmas. It is found that the LHCD and/or the synergy between IBW and LHCD were not sufficient to create the 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, increase of the impurity level cause stronger radiation power loss and degraded slightly the plasma confinement on one hand. On another hand, the internal m=1 kink mode, which may correlated with the unfavourate profile of the fast electron produced by LHCD, limited the performance in high beta plasma [2,28].

By proper optimization of operation, particularly, choosing a strategy to avoid MHD activities, high performance discharges under quasi-steady-state in the HT-7 has been realized by applying the RF waves in earlier phase of discharges and extending the RF pulse as long as possible. Figure 9 shows such an optimized discharges by utilizing synergy between LHW and IBW. In this shot, the plasma current was 120 kA, B_T = 1.7 T, $\bar{n}_e(0) = 1.6 \times 10^{19}/m^3$, P_{LHW} ~ 400 kW and P_{IBW} ~ 180 kW. The performance indicated by product of β_N*H_{89} ~2.3 factor was sustained for 4.3 s, about 220τ_E and longer than 20τ_{CR}, in which all plasma parameters and their profile reached stationary. The analysis of these discharges by EFIT and ONETWO shows that 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 internal transport barrier 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 plasma was also observed from the reciprocating Langmuir probe measurements, which means a H-mode edge in such reversed shear operational mode.

5. Long pulse plasma discharges

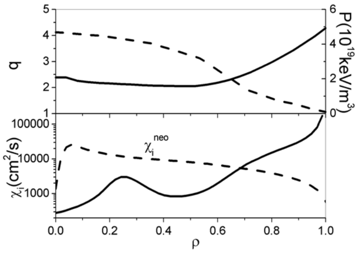

Fig.10 Profiles of plasma pressure and safe factor (top) and ion heat diffusion coefficient for shot in Fig.8 at 2s. The neo-classic ion heat diffusion coefficient is shown as dashed curve in bottom plot.

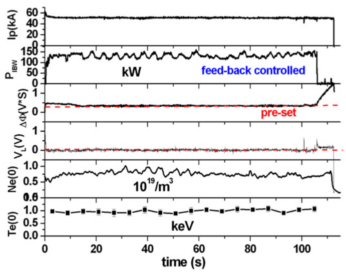

*Fig.11 A fully current driven long pulse plasma discharge was realized by the feed back control of the magnetic swing flux by the LHW power. From top to bottom: plasma current, LHW power, magnetic swing flux in V*S, loop voltage, central electron density and temperature.*

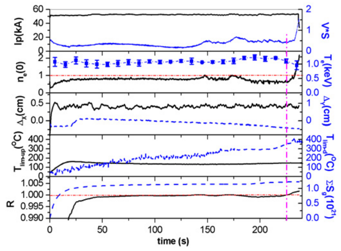

*Fig.12 4 minutes long pulse plasma discharge. Left from top to bottom: I_p, $n_e(0)$, horizontal plasma position, tile temperature in the up-limiter, global particle recycling coefficient. Right from top to bottom: magnetic swing flux in V*S, $T_e(0)$, vertical plasma position, tile temperature in the down-limiter, ΣS_g 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 piezo-electric valve. The magnetic swing flux of the transformer was feedback controlled by the LHW power. This is equivalently to control the loop voltage for full current drive. Figure 10 shows such a plasma discharge, in which the magnetic swing flux of the transformer (Fig.11c) was controlled to be constant for the most duration of the discharge and the loop voltage zero (Fig. 11d). The reproducible long pulse discharge with $T_e(0)$ ~ 1 keV and central electron density $n_e(0)$~0.8×10^{19} m^{-3} has been obtained with a duration of > 200 seconds under this operation mode. The longest plasma discharge was 240s as shown in Fig.12. Figure 13 shows the electron temperature profile at 100 sec. Such discharges can qualify 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 uncontrollable rise of the electron density. This situation is shown by a vertical dash-dot line in Fig.12 at about 220s.

The wall saturation and refresh process was firstly observed only in such long pulse discharges with duration longer than 200s. The deuterium gas was supplied by feed back control to keep the constant central line averaged electron density. The curves in the bottom of Fig.12 show 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 due to the feedback control of gas supply, when the global recycling coefficient R became unity or more. At about 180s, R decreased again below unity and gas was supply again. At the same period, the electron density was decreased dramatically although feedback control of the electron density and gas supply. It means that the wall repeat 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 the steady state tokamak operation.

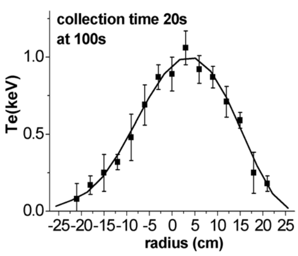

Fig.13 Profile of electron temperature at 100s in discharge of Fig. 11

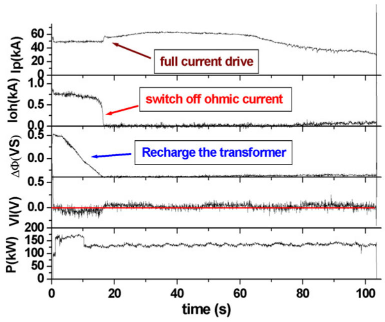

*Fig.14 A long pulse plasma discharge with switch-off of the current in the central solenoid. From top to bottom: plasma current, current in the central solenoid, magnetic swing flux in V*S, LHW power.*

Control of the plasma position is another key issue for achieving steady state plasma discharges. The plasma position was preset and then feedback controlled by our experience in the present experiments. Non-reasonable position caused strong plasma and wall interaction and led to significantly rise of the limiter temperature, which was account for the strong outgassing and uncontrollable density rise. During discharge shown in Fig.12, the temperature of one tile on the lower limiter raised quickly over 350 ^{O}C at about 220s, when the vertical plasma position reached to about -0.3cm. Continuous rise of the limiter temperature caused uncontrollable density rise and ultimately led to the termination of the discharge at 4 minutes. While temperature at all other measurement locations on both top and bottom limiters were kept below 200 ^{O}C during discharge, it indicates importance of the fine alignment for all plasma facing components on one hand. On another hand, new feedback loops to alleviate the plasma and wall interaction are needed to include the temperature information 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, plasma current was really 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 sustained for 80 s. In all these discharges, the measurements of the hard-X ray emission profiles shown the central power deposition of the LHW. The electron density/temperature profiles and plasma inductance were not obviously different from those in partial 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 the low beta plasma 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 reference [29].

7. Summary and conclusion

In the 2003-2004 experimental campaigns, we focused on long pulse plasma discharges and high performance plasma under quasi-steady-state condition 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 both in electron and ion channel were reduced in the outer half portion of the plasma. Although only IBW heating cannot sustain the plasma current in the steady state operation, its features in controlling the electron pressure profile can be applied into the LHCD plasmas to enhance the plasma performance. H-mode plasmas have been produced by off-axis LHCD, which was realized by selecting larger launched $n_{||}^{peak}$ of LHW at 3.1. The status of improved confinement in such plasma discharges has been sustained at the normalized performance of H_{89} >1.2 and β_N around unity for nearly 8 s, which is longer than 400 τ_E. The negative plasma current ramp created a high l_i target plasma. This high l_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 l_i plasma. An increase of the energy confinement time with l_i in the range of 1.2-1.7 is observed at the constant line averaged electron density and the injected power.

Combination of LHCD and IBW heating can produce the reversed shear in current density profiles. The features of IBW in controlling electron pressure profile were integrated into LHCD plasmas to tailor the current density profile and avoid MHD instability and optimized to achieving high performance plasmas. Significant progress in achieving high performance discharges under quasi-steady-state condition in the HT-7 super-conducting tokamak has been realized in such a RS operation scenarios. The normalized performance indicated by the product of $\beta_N H_{89}$ >2.2 was achieved for > 220 τ_E or > 20τ_{CR}. The fraction of non-inductive plasma current was larger than 80% in such discharges with 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. Increase of total injection power up to 1 MW did not degrade the plasma confinement significantly in the RS operational scenario. Further higher plasma performance in this operation mode was limited by the MHD instabilities, recycling and the increased impurity levels at higher injection power.

After installation of the new top-bottom symmetric belt limiter, the capabilities in heat load exhaust and particle removal were significantly enhanced. New modified LHCD launcher allowed long pulse power injection. With these technical improvement, the reproducible long pulse discharge with $T_e(0)$ ~ 1 keV and central electron density >0.8×10^{19} m^{-3} has been obtained with a duration of > 200 seconds. A new operation mode with over-current drive (V_p<0) and without using central solenoid current has been demonstrated for a really fully 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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