Overview of JT-60U Results
toward Establishment of Advanced Tokamak Operation

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Abstract. Recent JT-60U experimental results toward establishment of advanced tokamak (AT) operation are reviewed. We focused on the further expansion of the operational regime of AT plasmas toward higher \( \beta_n \) regime with wall stabilization. After the installation of ferritic steel tiles in 2005, the high power heating in a large plasma cross-section in which the wall stabilization is expected has been possible. In 2007, the modification of power supply of NBs improves the flexibility of heating profile in long-pulse plasmas. The investigation of key physics issues for the establishment of steady-state AT operation is also progressed using new diagnostics and improved heating systems. In weak magnetic shear plasma, high \( \beta_n \sim 3 \) exceeding the ideal MHD limit without conducting wall (\( \beta_N^{\text{no-wall}} \)) is sustained for \(~5\) s (\(~3\) \( \tau_R \)) with RWM stabilization by a toroidal rotation at \( q = 2 \) surface. A combined external current drivers of negative-ion based NB and lower-hybrid waves together with large bootstrap current fraction (\( f_{\text{BS}} \)) of 0.5 can sustain whole plasma current of 0.8 MA for \(~2\) s (1.5 \( \tau_R \)). In reversed magnetic shear plasma, high \( \beta_n \sim 2.7 \) (\( \beta_n \sim 2.3 \)) exceeding \( \beta_N^{\text{no-wall}} \) with \( q_{\text{min}} \sim 2.4 \) (\( q_{\text{pol}} \sim 5.3 \)), \( H_{98(y,2)} \sim 1.7 \) and \( f_{\text{BS}} \sim 0.9 \) is obtained with wall stabilization. These plasma parameters almost satisfy the requirement of ITER steady-state scenario. In long-pulse plasmas with positive magnetic shear, high \( \beta_n H_{98(y,2)} \) of 2.6 with \( \beta_n \sim 2.6 \) and \( H_{98(y,2)} \sim 1 \) is sustained for \(~25\) s significantly longer than the current diffusion time (\(~14\) \( \tau_R \)) without neoclassical tearing modes (NTMs). High \( G \)-factor (\( \beta_n H_{98(y,2)} q_{\text{pol}} \beta_n^{0.5} \)) which is a major of fusion gain) of 0.54 and large \( f_{\text{BS}} > 0.43 \) is suitable for ITER hybrid operation scenario. Based on the plasma for ITER hybrid operation scenario, the high \( \beta_n \) of 2.1 with good thermal plasma confinement of \( H_{98(y,2)} > 0.85 \) is sustained for longer than \(~12\) s at \( n_e/n_{\text{GW}} > 7 \) and \( f_{\text{pol}} > 0.79 \). Physics studies for AT plasmas, H-mode and pedestal physics, and studies on impurity transport, SOL/divertor plasmas and plasma wall interactions are also progressed. The active NTM stabilization system using modulated ECCD, which is synchronized to rotating island, has been developed and the efficiency of modulated ECCD in \( m/n = 2/1 \) NTM stabilization was demonstrated. The intrinsic toroidal rotation driven by the ion pressure gradient and by the ECH is confirmed. The dedicated H-mode and pedestal experiments indicates two scalings, \( H \)-factor evaluated for the core plasma as \( H_{98(y,2)} \sim 0.85 \) and pedestal width scaling of \( \Delta_{\text{ped}} = 0.315 a \rho_{\text{pol}}^{0.2} \beta_n^{0.5} \). New fast diagnostics with high spatial and temporal resolutions reveals the different structure of pedestal pressure between co- and counter-rotating plasma, resulting different ELM size and its radial penetration depth. The tungsten accumulation becomes more significant with increasing the toroidal rotation in the counter-direction.

1. Introduction

The steady-state advanced tokamak (AT) plasmas has been developed in JT-60U to realize economical fusion reactor [1]. High confinement, high normalized beta (\( \beta_n \)), high bootstrap current fraction (\( f_{\text{BS}} \)) and heat/particle handling are important parameters to maintain AT plasmas in steady-state. In the development of AT plasmas in JT-60U, the current profile with large \( f_{\text{BS}} \) is related to the large pressure gradient at the internal transport barrier (ITB) as shown in Fig. 1. The large \( f_{\text{BS}} > 0.7 \) is typically achieved in reversed magnetic shear (RS) plasmas in which a strong ITB having large pressure gradient is formed. In weak magnetic shear (WS) plasmas, a weak ITB having moderate local pressure gradient gives \( f_{\text{BS}} \sim 0.5 \). The flat magnetic shear with the minimum value of the safety factor \( (q_{\text{min}}) \) slightly above 1 was achieved in positive magnetic shear (PS) plasmas with weak ITB and consequent \( f_{\text{BS}} \sim 0.4 \). Based on these various magnetic shear profiles developed in JT-60U, we have investigated preferable current profile (magnetic shear profile) and pressure profile for steady-state operation. The sustainable current profile by the combination of the bootstrap current and the externally driven current with the neutral beam current drive (NBCD) and/or lower-hybrid current drive (LHCD) is examined. In addition, the current profile and pressure profile are also essential for high \( \beta_n \) operation regarding to the ideal MHD limit without conducting wall (\( \beta_N^{\text{no-wall}} \)) and the onset of neoclassical tearing modes (NTMs). It is noted that these AT research directly contribute to the development of advanced operational scenarios in ITER such as hybrid and steady-state scenarios [2].
JT-60U experiments in 2005-2006 focused on the improvement of the plasma performance and physics understanding of the parameter linkage in AT plasmas by reducing the toroidal field (TF) ripple. After the installation of ferritic steel tiles (FSTs) covering ~10% of the vacuum vessel surface, fast ion losses due to TF ripple were substantially reduced. Thus, high $\beta_N$ plasma experiment became possible even in the large volume plasma configuration, where the large ripple losses were observed without FSTs. The remaining issue for steady-state AT operation is to establish high $\beta_N$ plasma operation exceeding $\beta_N^{\text{no-wall}}$ in both WS and RS plasmas such as the sustained duration of high $\beta_N$ plasma in WS plasmas and lower stability limit in RS plasmas. Thus, JT-60U experiments have been focused on further expansion of the operational regime of AT plasmas toward higher $\beta_N$ regime with wall stabilization in 2007-2008. The modification of power supply of NBIs improves the flexibility of heating profile in long-pulse AT plasmas. The physics studies of important issues for establishment of steady-state AT operation and physics basis for ITER are also progressed using new diagnostics and improved heating systems. In this paper, recent JT-60U experimental results after the 21st IAEA Fusion Energy Conference [3] are reviewed.

FIG. 1. (a),(c),(e) Total pressure and safety factor profile in PS (0.9MA), WS (0.9MA) and RS (0.8MA) plasmas. (b),(d),(f) Total current and bootstrap current density profile in PS, WS and RS plasmas. $f_{\text{as}}$ of these plasmas evaluated by ACCOME code was ~0.4, ~0.5 and ~0.9, respectively.


The neutral beam injection (NBI) system and electron cyclotron heating (ECH) system have been improved for 2007-2008 experimental campaign to provide further flexibility to develop AT plasmas and to investigate important physics for steady-state AT operation. In NBI system, the power supply system for three perpendicular- (PERP-) NBIs for central heating has been modified so as to extend the maximum pulse duration up to 30s. In addition to this, 29s injection of negative-ion based NBI (N-NBI) with heating power of 3 MW (80MJ) has been achieved [4]. Thus, the total input energy from NBIs to the vacuum vessel has been progressed up to 445 MJ. In ECH system, four gyrotrons injects high EC power of 2.9 MW for 5s (14.5 MJ). For active NTM stabilization, the power modulation technique using anode voltage control has been developed to obtain high modulation frequency up to 7 kHz [5]. The system has a feature that it can trace the change of mode frequency during the discharge.

3. Extension of Operational Regime of Advanced Tokamak Plasmas

The AT tokamak research in JT-60U is focused on further extension of operational regimes toward higher $\beta_N$ regime for establishment of steady-state AT operation including steady-state and hybrid scenarios in ITER.

3.1. Sustainment of High $\beta_N$ Plasmas Exceeding No-Wall Ideal MHD Limit

Based on WS plasmas with weak ITB at $q_{\text{min}} = 1.2-1.6$ shown in Fig. 1 (c) ($I_p=0.9MA$, $B_T=1.44T$, $q_{95}=3.3$ and $d/a \sim 1.2$), high $\beta_N$ plasmas exceeding $\beta_N^{\text{no-wall}}$ are sustained for several seconds (longer than $\tau_R \sim 1.5$ s, $\tau_R$ is a current diffusion time defined as $\mu_0<\sigma> a^2/12$.
by keeping a toroidal rotation and its shear at $q = 2$ surface so as to suppress the amplitude of the resistive wall mode (RWM) \[7\]. As shown in Fig. 2(a), high $\beta_N \sim 3$ is sustained for $\sim 5$ s ($\sim 3\tau_R$). The waveforms of typical high $\beta_N$ discharge are shown in Fig. 2(b). In the plasma, PERP-NBIs were replaced with N-NBIs at $t = 5.5$-$5.9$ s so as to reduce the amplitude of $n = 1$ energetic particle driven wall mode (EWM), which often trigger RWM. The replacement of PERP-NBIs is also effective to keep co-rotation at $q = 2$ surface above the critical rotation of $\sim 20$ km/s, because the fast ion losses (originating from PERP-NBIs) produce counter-torque input. According to the ACCOME calculation for similar discharge (E49823), the large $f_{BS}$ of $0.46$-$0.5$ together with the large fraction of NBCD of $0.38$-$0.39$ provided the non-inductive current fraction of $0.84$-$0.89$. The stability analysis of similar discharge (E49823) by the ideal MHD stability code, MARG2D \[8\], indicates that $\beta_N^{\text{no-wall}}$ is $\sim 2.6$ and the ideal wall limit ($\beta_N^{\text{ideal}}$) is $\sim 3.2$. These values of $\beta_N^{\text{no-wall}}$ and $\beta_N^{\text{ideal}}$ correspond to $3.0\cdot l$ and $3.8\cdot l$, respectively. With these values as a measure of $\beta_N^{\text{no-wall}}$ and $\beta_N^{\text{ideal}}$, $C_\beta$ defined as $(\beta_N^{\text{no-wall}} - \beta_N^{\text{ideal}}) / (\beta_N^{\text{ideal}} - \beta_N^{\text{no-wall}})$ can be evaluated. Thus, the high $\beta_N$ plasma exceeding $\beta_N^{\text{no-wall}}$ with $C_\beta \sim 0.3$-$0.5$ shown in Fig. 2(b) was sustained for $\sim 5$ s until the $\beta_N^{\text{no-wall}}$ increased with the change in the current profile due to the current penetration in longer time scale than $\tau_R$.

In these high $\beta_N$ discharges exceeding $\beta_N^{\text{no-wall}}$, various MHD events related to the onset of RWM have been observed. In particular, EWM and RWM precursor have often been observed. Since both MHD modes triggers RWM within several seconds after the $\beta_N$ exceeded $\beta_N^{\text{no-wall}}$, as shown in Fig. 3, the suppression of these mode is essential for long-sustainment of high $\beta_N$ plasmas. The RWM precursor is often triggered by EWM and/or ELMs. The dedicated physics experiment reveals that the amplitude of EWM can be reduced by reducing the NBI power from PERP-NBIs (reducing trapped fast ions). Then, the sustained duration of high $\beta_N$ plasma exceeding $\beta_N^{\text{no-wall}}$ can be extended as shown in Fig. 2.

3.2. Sustainment of Full Current Drive Condition in Weak Magnetic Shear Plasmas

As described in previous section, the $\beta_N^{\text{no-wall}}$ increased together with the current penetration leading to the decrease in $q_{\text{min}}$ in WS plasmas. This result implies that external current drivers for off-axis current drive are needed to sustain the WS configuration. In order to establish steady-state WS scenario, WS plasma sustained by fully non-inductive current is developed in JT-60U \[9\]. Figure 4 shows waveforms of a WS plasma with $I_p = 0.8$ MA, $B_T = 2.3$ T and $q_{95} = 5.8$ under full-current drive (full-CD) condition. Pre-heating with counter-on-axis and co-off-axis NBs was applied during $I_p$ ramp-up phase so as to form WS. After the formation of ITB by the main heating at $t = 5.2$ s, the off-axis LHCD with $1.8$ MW (wave frequency $2$ GHz, parallel refractive index $N_p = 1.9$) and slightly off-axis NBCD by N-NBI with $1.2$ MW (beam energy
of 320 keV) were added at \( t = 6 \) s. Then, full-CD condition was achieved as can be seen in the surface loop voltage \( (V_{\text{loop}}) \) shown in Fig. 4(c). The \( V_{\text{loop}} \) inside the plasma became almost spatially uniform and reduced to 0 V at \( t = 8.2 \) s. The sustained period for 2 s corresponds to 1.5 \( \tau_R \). In fact, the current profile at \( t = 7.5 \) s and 8.0 s are almost identical, as shown in Fig. 4(d). The \( f_{\text{BS}} \) of the plasma was 0.5 and the rest of the plasma current fraction was sustained by off-axis LHCD (0.26) and NBCD (0.24). In this WS plasma, the full-CD condition sustained for about 2 s was terminated by the notching of LH power at \( t = 8.3 \) s. Thus, it is expected that the \( q \) profile at \( t = 8.0 \) s shown in Fig. 4(d) can be sustained with off-axis current driver together with large \( f_{\text{BS}} \) of \(-0.5\).

![FIG. 4. Waveforms of (a) \( \beta_N \), (b) injection power of P-NB (thin line), N-NB (medium line) and LH waves (thick line), (c) surface loop voltage. The full-CD condition lasts for \( \sim2 \) s at fixed \( \beta_N \). (d) safety factor profiles at \( t=6.0, 7.0, 7.5 \) and 8.0 s measured by the MSE diagnostics. The \( q_{\text{min}} \) and the \( q(0) \) at \( t=8.0 \) s are 2.1 and 2.4, respectively.]

### 3.3. Expansion of Operational Regime in Reversed Magnetic Shear Plasmas with High \( f_{\text{BS}} \)

The AT tokamak research in RS plasmas with high \( f_{\text{BS}} \geq 0.7 \) has been performed at high \( q_{95} \geq 8 \) so far. But, lower \( \beta \) limit is main issue in this regime for steady-state AT operation. In 2008, further development to expand the operational regime of RS plasmas in terms of lower \( q_{95} \) and higher \( \beta_N \) exceeding \( \beta_{\text{no-wall}} \) is performed by utilizing the wall stabilization with \( d/a \sim 1.1-1.3 \) as shown in Fig. 5(a) [10]. Typical waveforms of the RS discharge \( (I_p=0.8\text{MA}, B_T=2.0\text{T}, q_{95}\sim5.3, \kappa \sim 1.5, \delta \sim 0.39, d/a \sim 1.3) \) with strong ITB are shown in Fig. 5(c). In this discharge, high \( \beta_N \sim 2.7 \) and \( \beta_p \sim 2.3 \) were achieved, although the plasma was terminated by the disruption at \( t \sim 6.1 \) s. The achieved \( \beta_N \) is much higher than previous experiments with \( \beta_{\text{wall}} \sim 1.7-2.2 \) in high \( f_{\text{BS}} \) plasmas. The disruption was caused at \( q_{\text{min}} \sim 2.4 \) (not at the integer value of \( q_{\text{min}} \)) by the RWM \((n = 1)\), of which growth time is the order of the resistive wall time \( (\tau_W \sim 10\text{ms}) \). The stability analysis by MARG2D code indicated that the plasma exceeded \( \beta_{\text{no-wall}} \sim 2.0 \) \((\sim 3\iota)\). Moreover, high confinement enhancement factor \( H_{99\text{(y,2)}}=1.66 \) was obtained at high density regime \((\eta_e\eta_{\text{GW}} \sim 0.87)\) attributed to both internal and edge transport barriers. The reversed \( q \) profile shown in Fig. 1(e) was formed with \( q_{\text{min}} \sim 2.4 \) and its location was \( \rho_{\text{min}} \sim 0.6 \). As observed in the total current profile and bootstrap current profile shown in Fig. 1(f), high \( f_{\text{BS}} \sim 0.9 \) was achieved at the end of the discharge. Since tangential NBIs were used as

![FIG. 5. (a)/(b) Operational regime of RS plasmas with high \( f_{\text{BS}} \). (c) Waveforms of high \( f_{\text{BS}} \) discharge.]
balanced injection, beam driven current was negligible. Based on these RS plasmas, operational regime of RS plasmas with high $f_{BS}$ and high $\beta_N$ is extended towards reactor relevant low $q_{95}$ regime as shown in Fig. 5(a) and (b). The parameters of RS plasmas are close to the required values for steady-state scenarios in ITER. However, a plasma control scheme to avoid disruption (suppress the RWM) should be developed for the establishment of steady-state RS plasma operation.

### 3.4. Development of Long-Pulse Hybrid Scenario with High $\beta_N$ and High Confinement

The long-pulse hybrid scenario has been developed in JT-60U based on the high $\beta_N$ ELMy H-mode plasma ($I_P=0.9$MA, $B_T=1.54$T and $q_{95}=3.2$) with weak ITB. In order to extend the operational regime of hybrid discharges toward higher $\beta_N$ and higher density region for higher neutron fluence, the importance of the central heating to keep a peaked pressure profile and achievable $\beta_N$ in term of NTM avoidance in hybrid discharges have been investigated using new capability of long-pulse injection from PERP-NBIs for central heating. As shown in Fig. 6(a), a long-pulse hybrid discharge with high $\beta_N>2.6$ is sustained for 28s [9, 11]. The high thermal confinement of $H_{H98}(y,2)>1$ characterized by the peaked pressure profile shown in Fig. 6(b) is also sustained for 25 s ($\sim 14\tau$) until $t=29$ s with central heating shown in Fig. 6(c). The peaked profiles are appeared in both temperature and density profiles at $n_e/n_{GW}=0.55$. Although these hybrid discharges have highly peaked pressure profile, the pressure gradient at the mode rational surfaces at $q=1.5$ and 2 is small enough to avoid the onset of NTMs. This NTM avoidance scenario is effective up to $\beta_N=3$. The flat $q$ profile in the core region ($r/a < 0.5$) with $q_{min}$ of about unity is sustained throughout the discharge and it is mainly assisted by the bootstrap current as shown in Fig. 1(b). Thus, only infrequent sawtooth activities are observed in these hybrid discharges and their amplitude are small enough to sustain peaked pressure profile and to avoid the formation of a seed island which could trigger NTMs. The sustainable $\beta_N$ in long-pulse hybrid discharges has been improved by more than 10% as shown in Fig. 6(d). It is noted that high $\beta_N H_{H98}(y,2)$ of 2.6 gives high $G$-factor ($\beta_N H_{H98}(y,2)/q_{95}^2$) of 0.25 ($\beta_N H_{89P}/q_{95}^2=0.54$) and large $f_{BS}>0.43$ under the ITER-like small toroidal rotation ($V_T$) condition ($V_T < 1$kHz). Therefore, these long-pulse hybrid discharges developed in JT-60U are suitable for ITER hybrid operation scenario.

![FIG. 6. (a)Waveforms of long-pulse hybrid discharge. (b),(c) Comparison of thermal pressure profile and power deposition profile between central heating (solid line) and off-axis heating (dashed line) at the same density of $n_e/n_{GW}=0.55$. (d) Sustained duration of high $\beta_N$ plasmas.](image)

### 3.5. Sustainment of High Density and High Radiation Plasmas with High Confinement

Reduction of heat loading appropriate for the plasma facing components such as the divertor and the first wall is also important issue for next step devices. Thus, the long-pulse high
density and high radiation plasmas with good confinement has been developed based on long-pulse hybrid discharge \( I_p = 1.05 \, \text{MA}, B_T = 2.0\, \text{T} \) and \( \eta_{95} = 3.6 \) using impurity gas seeding \[12\]. The puffing rate of Ar gas was controlled by the real-time feedback system to keep the radiation from the edge region of main plasma constant. In addition to the injection of Ar gas, pulsed gas puffing of Ne gas was also applied from the divertor injector to increase the radiation at the divertor region. As the Ne puff was injected, the radiation power at the divertor gradually increased, while keeping the core radiation was almost constant. The total radiation loss fraction, \( f_{\text{rad}} = \frac{P_{\text{rad,main}} + P_{\text{rad,div}}}{P_{\text{net}}} \), was achieved 0.79 at \( t = 14 \, \text{s} \). Although the wall-pumping was not effective at \( t \sim 12 \, \text{s} \) (under wall saturated condition), the high \( \beta_n \) of 2.1 with good thermal plasma confinement of \( H_{198}(y,2) > 0.85 \) was sustained for longer than 12 s at \( n_e/n_{GW} \geq 0.7 \) and \( f_{\text{rad}} \geq 0.8 \) with stored energy feedback control and radiation feedback control. The species of impurity for effective cooling was also investigated as shown in Fig. 7(b). From the viewpoint of the radiation at divertor region with lower temperature, Ne is more effective than Ar. On the other hand, Ar is better radiator for the main plasma. Thus, the combination of Ar + Ne is better way to obtain high \( f_{\text{rad}} \sim 0.8 \) while keeping high \( H_{198}(y,2) \).

4. Progress in Physics Studies for Advanced Tokamak Plasmas

Various physics researches for AT plasmas have been performed in JT-60U in order to resolve important issues for improvement of plasma performance and better prediction of AT plasmas in next step device.

4.1. Active Stabilization of Neoclassical Tearing Modes

The active NTM stabilization system using modulated ECCD, which is synchronized to rotating island at \( \omega = 5 \, \text{kHz} \) as shown in Fig. 8(a), has been developed in JT-60U \[13, 5\]. The system can trace the change in the rotation speed during the stabilization. The efficiency of modulated ECCD in \( m/n = 2/1 \) NTM stabilization is demonstrated as shown in Fig. 8(b). The stabilizing effect of modulated ECCD near the O-point with one gyrotron is similar to the unmodulated ECCD with two gyrotrons. Figure 8(c) also shows the effectiveness of O-point ECCD. The decay time (defined by fitting the magnetic perturbation amplitude as \( \exp[-t/\tau_{\text{decay}}] \) for the initial 300 ms data) reaches a minimum around -10°, which corresponds to O-point ECCD.

FIG. 8. (a) Magnetic probe signal and gyrotron power. (b) Time evolution of magnetic perturbation amplitude for modulated ECCD (AC) followed by unmodulated ECCD (DC). (c) Dependence of decay time of magnetic perturbation amplitude on phase difference.
The minimum electron cyclotron (EC) wave power for complete NTM stabilization of an $m/n=2/1$ mode is experimentally identified as $0.2 < j_{EC}/j_{BS} < 0.4$ for $W_{sat}/d_{EC} \sim 3$ and $W_{sat}/W_{marg} \sim 2$, and $0.35 < j_{EC}/j_{BS} < 0.46$ for $W_{sat}/d_{EC} \sim 1.5$ and $W_{sat}/W_{marg} \sim 2$ ($W_{sat}$, $W_{marg}$ and $d_{EC}$ are full island width at saturation, marginal island width to fully stabilize the NTM and FWHM of ECCD profile, respectively). For complete NTM suppression, it is confirmed that the precise alignment of the ECCD location, $(\rho_{EC}=\rho_{NTM})/d_{ECCD} \leq 0.2$, is required at $W_{sat}/d \sim 1.5$ and $W_{sat}/d \sim 3$ [13]. The modified Rutherford equation (MRE) is tested against experimental data from ASDEX Upgrade and JT-60U to obtain a common coefficient for MRE. The same analysis method to estimate the coefficient of $c_{sat}$ for saturation phase was applied to experimental data obtained in the case of $m/n=3/2$ NTM from both devices. It is found that $c_{sat}$ is the order of unity for both devices [14]. The NTM experiments in JT-60U for this study were performed from IPP Garching using the remote experimental system developed in JAEA [15].

### 4.2. Mechanism of Formation of Toroidal Rotation Profile and Momentum Transport

As described in other section, it has been found that the $V_T$ is important for the performance of ITB and ETB, characteristics of ELM and stabilization of RWM. Thus, detailed properties of momentum transport and intrinsic rotation as the determining process of $V_T$ have been investigated in ELMy H-mode plasmas using transient analysis to evaluate the momentum diffusivity, $\chi_i$, and the convection velocity, $V_{conv}$, separately. The $\chi_i$ evaluated at the core plasma region increases with increasing heating power and decreases with increasing plasma current. The ratio of $\chi_i$ to $\chi_f$ at $r/a = 0.5$ is found to be ~0.7–3 and it increases with increasing $T_i$ as shown in Fig. 9. The ratio of $V_{conv}$ to $\chi_f$ is found to be -2 – -0.5 $(m^2/s)$. The intrinsic rotation driven by the local pressure gradient toward counter direction increased with increasing the ion pressure gradient $(gradP_i)$ [16, 17]. This dependence is quite robust, since the same dependence is found in different plasma conditions ($I_p$, L-mode, H-mode, co-$V_T$, counter-VT and no rotation).

The intrinsic rotation by ECH is also studied in plasmas with balanced NBI heating to minimize the momentum input. As shown in Fig. 10, the measured $V_i$ in the region of 0.2<r/a<0.3 is changed with ECH changed in co-direction, while one in the region 0.3<r/a<0.6 changed in the counter-direction [17]. This response to ECH is determined by the combination of the momentum transport, the intrinsic rotation by $\chi_i$ and the intrinsic rotation by ECH. As for the momentum transport with ECH, it is confirmed that $V_i$ produced by $\chi_i$ and $V_{conv}$ in the region 0.4<r/a<0.7 are not changed. The change in the intrinsic rotation by $\chi_i$ is evaluated as shown by dashed line in Fig. 10. Since the increase of counter-rotation in the region 0.4<r/a<0.6 can not be explained by the change in $\chi_i$ and the momentum transport, the difference (shown by solid line in Fig. 10) between the change in measured $V_i$ with ECH and the change in the intrinsic rotation by $\chi_i$ indicates the intrinsic rotation due to ECH. The generation of the toroidal rotation due to the radial current torque induced by the charge separation of the particle injected from NBI is studied by using one-dimensional multi-fluid transport code TASK/TX, which can evaluate $j\times B$ torque due to the charge separation self-consistently [18]. The simulation reproduced the toroidal rotation driven by the radial current.
from the PERP-NBI. The analysis with varying vertical injection angle (poloidal angle) indicates that the NBI from equatorial port drives the toroidal rotation most efficiently.

4.3. Studies on Internal Transport Barrier

4.3.1. Response of ITB to External Perturbations
Responses of ITB to the cold pulse perturbation by edge fueling of shallow HFS pellet injection and supersonic molecular beam injections (SMBI) installed in collaboration with CEA-Cadarache are investigated [19]. The ion temperature ($T_i$) decreases even in the central region with both pellet injection and SMBI. In the SMBI case, time evolution of $T_i$ measured with newly installed fast CXRS system can be described by cold pulse propagation using the ion thermal diffusivity estimated from power balance analysis. By optimizing the injection frequency and the fueling depth for both SMBI and pellet, high confinement is sustained at high density by keeping strong ITB and enhanced pedestal pressure.

Variety of $T_i$-ITB response is observed with central ECH in weak shear plasmas depending on the target plasmas [19]. When stiffness feature is strong in $T_e$ profile, $T_i$-ITB degraded. On the other hand, $T_i$-ITB is unchanged or even grows, when stiffness feature is weak in $T_e$ profile. The spectrum of density fluctuation measured with reflectometer was not so changed, however, correlation length becomes longer for the degradation case and shorter for the unchanged case after injection of ECH.

4.3.2. Comparison of ITB between LHD and JT-60U
The development of an ITB observed in LHD and JT-60U plasmas in the region of negative magnetic shear are compared in terms of the evolution of the location of ITB and its strength using new charge exchange recombination spectroscopy (CXRS) with high spatial and temporal resolution [21]. The ITB region expands towards plasma core and becomes wide in ITB observed in LHD plasmas, while the ITB region is localized in the narrow region in box-type ITB (strong ITB) observed in JT-60U plasmas. Another difference is the strength of the ITB. The steep $T_i$ gradient of $dT_i/dr$ $\sim$ 60 keV/m is observed in the box-type ITB in JT-60U, while weak $dT_i/dr$ $\sim$10 keV/m is observed in typical ITB in LHD.

4.4. Studies on Ion Cyclotron Emission due to DD Fusion Product
The ion cyclotron emissions (ICEs) due to $^3$He-ions (ICE($^3$He)) and T-ions (ICE(T)) are studied to understand the mechanism of the wave excitation. The fundamental ICE($^3$He) is only observed in plasmas with the line averaged density less than $1.3\times10^{19}$ m$^{-3}$. The second-harmonic ICE($^3$He) is observed in plasmas with higher density region than that for the fundamental mode, while it also disappears at the density higher than $3\times10^{19}$ m$^{-3}$. The wave numbers of the second harmonic ICE($^3$He) and the fundamental ICE(T) are evaluated as around 4 and 8 m$^{-1}$, respectively. The observed density dependence and wave numbers are consistent with the dispersion relation of the magneto-acoustic wave in D-plasmas with minority $^3$He-ions. The fact that the wave number of ICE(T) is longer than that of ICE($^3$He) indicates that the excitation of slow Alfvén waves is the mechanism for ICE(T) [22].

4.5. Studies on Current Decay Time during Disruption
The plasma current decay time during initial phase of the density limit disruption is investigated based on the plasma inductance and resistance evaluated by the experimental
data. The electron temperature profile during the current quench is measured with the ECE diagnostic and with the ratio of He I line emission intensity. The plasma inductance is estimated by the Cauchy-Condition Surface method with magnetics. It is found that the time change rate of the plasma inductance during current quench is an important parameter to predict the current decay time, because the observed current decay time seems to be independent from the electron temperature (plasma resistance) [23].

5. Studies on H-mode and Pedestal Physics

5.1. Properties of Heat Transport in ELMy H-mode Plasmas

The effect of current density profile on the heat transport and edge pedestal performance is investigated in JT-60U. H-mode plasmas with higher $l_i$ due to the current ramp down show higher energy confinement with higher density [24]. The H-factor evaluated for the core plasma ($H_{89\text{core}}$) depends strongly on $l_i$ with the relation of $H_{89\text{core}} \propto l_i^{0.77}$ for the case without sawtooth activities. Center peaked profiles of $n_e$ and $T_e$ are obtained in H-mode plasmas with high $l_i$. While the peripheral current density profiles are largely modified by the current ramp, the pedestal pressure is not significantly changed. The higher energy confinement in H-mode plasmas with high $l_i$ is attributed to the core improvement with the peaked profiles of $n_e$ and $T_e$, while no explicit difference in pedestal profile is observed. The electron heat diffusivity is reduced at the plasma core in high $l_i$ case, resulting in the center peaked $T_e$ profile while the $T_i$ profiles are approximately unchanged.

5.2. Properties of Pedestal Structure

Previous empirical scalings of the pedestal width ($A_{\text{ped}}$) include pedestal $\beta$ and/or $\rho^*$ dependence [25]. In order to separate these two parameters, experiments in hydrogen and deuterium plasmas were performed in JT-60U. As shown in Fig. 11, the log-linear regression analysis indicates the scaling expressed as $A_{\text{ped}} = 0.315 \rho^* \beta_{\text{pol}}^{0.5}$ [26]. It is confirmed that the pedestal data in the $I_p$ ramp experiments described in previous section follows the same scaling as shown in Fig. 11 marked by red and blue symbols.

The width of density pedestal is not always the same as that of the temperature pedestal. Therefore, the mechanism to determine the width of density pedestal is important. In order to understand the effect of neutral on the pedestal characteristics, three-dimensional version of the DEGAS Monte-Carlo code has been used to analyze JT-60U H-mode plasmas. The simulation results show that the increase of edge pedestal density causes a noticeable reduction of 1/e scale length of the neutral penetration. Consequently, the neutral penetration depth decreases with increasing the pedestal density, which leads to the localization of ionization area near the edge pedestal region [27].

5.3. Fast Dynamics of Type I and Grassy ELMs

In order to understand the physics determining the ELM energy loss ($\Delta W_{\text{ELM}}$), fast ELM dynamics of type I ELMs and grassy ELMs have been studied in JT-60U using new fast diagnostics with high spatial and temporal resolutions such as fast CXRS for $T_i$ profile measurement and Li beam probe for $n_e$ profile measurement. The evolution of pressure profile is evaluated for the first time by the detailed profile measurement of the density and temperature pedestals [28]. After a type I ELM crash, it is found that recovery of pedestal density is faster than that of temperature. Just before type I ELM crash, the pedestal ion pressure and its gradient in co-rotating plasmas is higher than those in counter-rotating plasmas as shown in Fig. 12. (20% higher for ion pressure and 12% higher for maximum pressure gradient) It is noted that the pressure gradient inside the top of pedestal is also higher in co-rotating plasmas. These results suggest that the ELM size is determined by the structure of the plasma pressure in the whole pedestal region. It is also found that the type I ELM expels/decreases edge toroidal momentum larger than ion thermal energy [20]. In plasmas
shown in Fig. 12, the ELM affected area is deeper in the order of $V_t$, $T_i$ and then $T_e$. As for the dynamics of grassy ELMs, the collapse of density pedestal is smaller and narrower than that of type I ELMs, as observed in the collapse of temperature pedestal. Thus, it is confirmed that both conductive and convective losses due to grassy ELMs are small.

The relation between $\Delta W_{\text{ELM}}$ and ELM cycle has also been investigated using integrated modeling code, TOPICS-IB. The steep pressure gradient inside the pedestal top broadens the region of the ELM enhanced transport, resulting large $\Delta W_{\text{ELM}}$. A transport model with the pedestal neoclassical transport connected to the SOL parallel transport reproduces the experimentally observed collisionality dependence of inter-ELM transport [29]. For constant inter-ELM transport, the ELM frequency ($f_{\text{ELM}}$) decreases with increasing the $\Delta W_{\text{ELM}}$, so that the ELM loss power ($f_{\text{ELM}} \times \Delta W_{\text{ELM}}$) remains constant. Effects of a toroidal rotation on the stability of the MHD modes in the edge pedestal are also investigated numerically using new linear MHD stability code MINERVA. It is revealed that the destabilizing effect of the sheared toroidal rotation on ELMs. This destabilizing effect becomes stronger as the toroidal mode number of the unstable MHD mode increases [30].

5.4. ELM Control by Electron Cyclotron Heating at Edge Pedestal

The effect of local heating at the pedestal by ECH on ELM characteristics has been investigated in terms of active ELM control. When the pedestal at the top of the plasma at high-field side (HFS) was heated by ~1 MW EC, the $f_{\text{ELM}}$ was increased from ~75 Hz to ~120 Hz. No response was observed, when the pedestal at the top of the plasma at low-field side (LFS) was heated. The increase in $f_{\text{ELM}}$ was not due to the increase in the heating power, since the increase in the injection power of NB I by 2.2 MW (absorbed power of 1.3 MW) only changed the $f_{\text{ELM}}$ from ~70 Hz to ~85 Hz. With the increase in $f_{\text{ELM}}$, the $\Delta W_{\text{ELM}}$ was reduced by ~30%. Thus, the localized pedestal heating by ECH can be considered as a candidate for the active ELM control technique in ITER.

5.5. Properties of Density Profile in ELMy H-mode Plasmas

The peaked density profile is suitable to obtain higher plasma performance as discussed before, the density profiles in LHD heliotron and JT-60U tokamak plasmas are compared in terms of the collisionality ($\nu_b^*$ defined by an electron-ion collision frequency normalized by the trapped electron bounce frequency). The density peaking factors increased with decreasing $\nu_b^*$ in JT-60U, while the $\nu_b^*$ in LHD depends on magnetic configuration [31, 32]. The density peaking factor moderately increases with decreasing $\nu_b^*$ in LHD plasmas with the plasma configuration that the magnetic axis of located at $R_{ax}=3.5$ m, while the factor decreases with decreasing $\nu_b^*$ with $R_{ax}=3.6$ m configuration. In the plasma having higher density peaking, the radial correlation length measured with the correlation reflectometer becomes longer in JT-60U [32].


6.1. Impurity Generation and Transport

The generation and transport of impurities are quite important to obtain high performance plasmas. The behavior of carbon (C) and tungsten (W), which are candidates for the material of ITER divertor tile, are investigated.

6.1.1. Studies on Carbon Transport
In order to investigate the generation and the loss flux balance of $\text{C}^{3+}$, which contributes 60% to the total radiation power in detached divertor plasmas, the generation flux (ionization of $\text{C}^{2+}$ and recombination of $\text{C}^{4+}$) and the loss flux (ionization and recombination of $\text{C}^{3+}$) are compared for the first time. It is found that $\text{C}^{3+}$ is comparably produced by the volume recombination of $\text{C}^{4+}$ and the ionization of $\text{C}^{2+}$ [33]. On the contrary, the volume recombination of $\text{C}^{3+}$ is not detected, and the ionization flux of $\text{C}^{3+}$ is less than 1% of the $\text{C}^{3+}$ generation flux. Thus, the $\text{C}^{3+}$ generation flux is higher by two orders of magnitude than the loss flux. This result suggests that another loss mechanism of $\text{C}^{3+}$ such as transport loss from the X-point is significant.

The carbon density profiles during the ITB formation phase are compared between LHD and JT-60U. The radial profiles of carbon density are evaluated from the radial profiles of intensity of charge exchange line CVI and beam attenuation calculation based on the measured density and temperature profiles. The carbon density starts to decrease due to an outward convection in LHD, while the carbon density tends to increase due to an inward convection in JT-60U associated with the formation of ITB [21]. The convection velocity in LHD plasma is 0.5 m/s (outward), while that in JT-60U plasma is -0.2 m/s (inward). The difference in the sign in the convection of the impurity transport in ITB plasma would be due to the different turbulence structure driving radial flux in particle between LHD and JT-60U.

6.1.2. Studies on Tungsten Transport
Compatibility of W as a plasma facing material has been investigated in JT-60U using the W-coated outer divertor tiles (12 tiles, covering 1/21 toroidal length). It has been found that W accumulation becomes more significant with increasing the toroidal rotation in the counter-direction [33]. As shown in Fig. 13, the impurity accumulation level shows clear Z-dependence, while it does not depend on the generation flux/puffing rate. These experimental results suggest the transport of impurity plays a dominant role to determine the accumulation level. The ECH is confirmed as a method to change the transport of impurity, since impurity accumulation level can be reduced during ECH even in the counter-rotating plasmas.

6.1.3. Studies on Tungsten Deposition at Divertor
Deposition profiles of tungsten released from the outer divertor are investigated. A neutron activation method is used for the first time to accurately measure deposited tungsten. Surface density of tungsten in the thick carbon deposition layer can be measured by this method. Tungsten is mainly deposited on the inner divertor and on the outer wing of the dome [34]. Toroidal distribution of the W deposition is significantly localized near the tungsten sources (W coated tiles) at the outer wing of the dome.

6.2. Fluctuation Characteristics in SOL plasmas
Detailed comparison between fluctuation characteristics at HFS and LFS scrape off layer (SOL) plasma has been made, for the first time, in L-mode plasmas using reciprocating Langmuir probes. Statistical analysis based on probability distribution function (PDF) is employed to describe intermittent (non-diffusion) transport in SOL plasma fluctuations. It is found that the positive bursting events associated with blob-like plasma transport are more frequently appeared at LFS midplane [35]. The PDF for LFS SOL plasma is strongly skewed positively, while the PDF in HFS SOL is close to Gaussian distribution. Conditional averaging analysis of the positive bursting events at LFS midplane indicates that the intermittent feature with a rapid increase and a slow decay are similar to the characteristics predicted for blobs theoretically. Statistical self-similarity is also investigated with Fourier power spectrum, and statistics of waiting-time and duration-time of the fluctuation. It is found, for the first time, that clear statistical self-similarity is observed at LFS SOLs, showing
fractal property of the fluctuation. The scaling exponents disagree with the predictions for the self-organized criticality (SOC) paradigm.

7. Summary
JT-60U experiments have been focused on the further expansion of the operational regime toward higher $\beta_N$ regime with wall stabilization for establishment of steady-state AT operation. In WS plasma, high $\beta_N \sim 3$ exceeding $\beta_N^{\text{no-wall}}$ is sustained for $\sim 5$ s ($\sim 3 \tau_R$) with RWM stabilization by a toroidal rotation at $q = 2$ surface. In RS plasma, high $\beta_N \sim 2.7$ ($\beta_N \sim 2.3$) exceeding $\beta_N^{\text{no-wall}}$ with $q_{\min} \sim 2.4$ ($q_{95} \sim 5.3$), $H_{H98(y,2)} \sim 1.7$ and $f_{\text{BS}} \sim 0.9$ is obtained with wall stabilization. In PS plasma, high $\beta_N H_{H98(y,2)}$ of 2.6 with $\beta_N \sim 2.6$ and $H_{H98(y,2)} \sim 1$ is sustained for 25 s significantly longer than the current diffusion time ($\sim 14 \tau_R$). Based on the hybrid discharge, high $\beta_N$ of 2.1 with good thermal plasma confinement of $H_{H98(y,2)} > 0.85$ is sustained for more than 12 s at $n_e/n_{GW} > 0.7$ and $f_{\text{rad}} > 0.79$. Physics studies for AT plasmas, H-mode and pedestal physics, and studies on impurity transport, SOL/divertor plasmas and plasma wall interactions are also significantly progressed.

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References