REVIEW OF JT-60U EXPERIMENTAL RESULTS
FROM FEBRUARY TO OCTOBER, 1999

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JT-60 Team

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Review of JT-60U Experimental Results
from February to October, 1999

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In 1999, the plasma parameters of reversed shear (RS) plasmas had been extended in 1) DT-equivalent fusion power gain $Q_{\text{DT}}^{\text{eq}} \sim 0.5$ ($n_{\text{e}}(0)T_{\text{e}}(0) \sim 4 \times 10^{20} \text{ m}^{-3} \text{ keV s}$) for 0.8 s and 2) full non-inductive current drive with 80% of the bootstrap current fraction. Physics of the internal transport barriers (ITBs) in RS plasmas, including the energy transport and the formation of ITB, were extensively studied. A nearly full current drive (92% non-inductively) was obtained with negative ion based neutral beam (NNB) injection (360 keV, 3.4 MW) in a high $\beta_p$ H-mode plasma ($I_p = 1.5 \text{ MA}$, $B_T = 3.7 \text{ T}$, $q_{95} = 4.2$) with high plasma performance ($\beta_n = 2.4$ and $H_{99} = 2.56$). Rise in the central electron temperature ($T_e \sim 9 \text{ keV}$) resulted in the current drive efficiency $\eta_{\text{CD}}$ of NNB reached $1.3 \times 10^{10} \text{ A/W/m}^2$, the highest for the neutral beam current drive. As for the H-mode plasmas, decrease in the pedestal ion temperature due to strong gas was found cause degradation in core plasma confinement.

The operation of an ECRF system of 110 GHz, 0.75 MW (torus injection power) had been started in 1999. Changes in plasma current profiles and suppressions of tearing modes and sawtooth oscillations were observed with ECRF heating. The highest $\beta_n$ in RS experiments ($\beta_n \sim 2.8$) was obtained in the plasma configuration with a large wall stabilizing effect and resistive wall modes were observed before the disruptions. New real-time feedback control schemes including plasma stored energy, plasma radiation power and so on were used in routine plasma operations. Runaway electron current was terminated when the plasma surface safety factor was forced to drop below 2 or 3.

In order to increase pumping efficiency of deuterium and impurity neutrals, the outer pumping slot was opened and the both sides pumping was enabled in the W-shaped divertor.
in 1999. In the divertor configuration with the maximum pumping efficiency, $Z_{\text{eff}}$ was reduced to from 2.6-3.0 to 2.3-2.6 in beam heated discharges even with X-point MARFE and $\tau_{\text{He}}/\tau_{\text{e}}$ was reduced by 45% in He exhaust experiments. Feedback control of seed impurity for radiation enhancement, such as Ar, enabled production of ELMy H-mode plasmas with high density (70% of the Greenwald density) and better energy confinement ($H_{\text{99}} = 1.4 - 1.5$). It was found that the chemical sputtering generated the same amount of CD$_4$ and C$_2$D$_x$ molecules in the divertor since a visible divertor spectrometer with 16 ch spatial resolution was newly installed.

Keywords:
JT-60U, Fusion Power Gain, Steady State, Reversed Shear, Internal Transport Barrier, High $\beta_p$ H-mode, Current Drive, NNB, ELMy H-mode, ECRF, W-shaped Divertor, Impurity Exhaust
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JT-60U 1999年2月-10月実験結果のレビュー

日本原子力研究所那珂研究所 炉心プラズマ研究部・核融合装置試験部

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1999年に、負磁気シア放電の長時間化について、1) DT等価核融合増倍率$Q(N_{e0} \tau_{e} T_{e}(0) - 4 \times 10^{20} \text{m}^{-3} \cdot \text{keV} \cdot \text{s})$~0.5の0.8秒維持、2) 80%が自発電流の完全電流駆動を実現した。負磁気シア放電の内部輸送壁(ITB)の物理に関連して、エネルギー輸送やITBの形成等の研究を進めた。高性能(β_n=2.4、H_{95}=2.56)の高β_p H-modeプラズマ(T_e=1.5 MA、B_0=3.7 T、q_{95}=4.2)において、92%の電流駆動を得た。プラズマ中心電子温度の上昇(T_e≈9 keV)により、NNB(3.4 MW、360 keV)による電流駆動効率は、ビーム電流駆動としては最高の1.3 x 10^8 A/W/m²であった。H-mode研究では、強いガスバブルによるベデスタルでのイオン温度の減少が中心プラズマの閉じ込めの劣化の原因となることがわかった。

1999年より周波数110GHz、入射加熱パワー0.75MW(トーンル入射パワー)のECRF加熱系が稼働を開始した。プラズマへの入射により電流分布の変化、テアリング不安定性及び線電流振動に対する抑制効果を観測した。真空容器壁による安定化効果が大きい配位で、負磁気シアプラズマにおいて最高値のβ_n≈2.8を得て、ディスラプション時に抵抗性壁モードを観測した。プラズマの安定係数を2あるいは3以下に落すことにより、逃走電子を減滅できることを示した。プラズマ制御では、新たに導入した蓄積エネルギー、放射損失量他の実時間制御を、実験で日常的に使用した。

重水素及び不純物中性粒子排気の増強のため、1999年にはW型ダイバータにおいて外側排気口が追加され、両側ダイバータからのポンプ排気が可能となった。排気速度が最大となる配位では、ビーム加熱放電でX点MARFEが存在してもZeffが2.6-3.0から2.3-2.6に低減した。ヘリウム排気実験においても、従来より45%改善してτ_{l_e}/τ_{e}≈2.8を増大。放射損失を増大させるAr等の不純物量のフィードバック制御により、高密度(Greenwald密度の70%)で従来より高い閉じ込め性能(H−1.4−1.5)のELMy H-modeプラズマを得た。ダイバータ領域を空間分解能16cmで観測可能な観察器の導入とともに、化学スパッタリングでCD_xとC_P D_x分子の発生がほぼ同程度であることを明らかにした。
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1. Overview of the Experiments in 1999

1.1 Core Plasma

Further effort for the extension of high performance plasma regimes was devoted in 1999 aiming at 1) sustainment of high fusion performance \( Q_{\text{DT}}^{\text{eq}} \) in reversed shear (RS) plasmas, 2) development of high bootstrap current fraction in RS plasmas, and 3) realization of 1.5 MA full-CD high \( \beta_p \) H-mode plasmas. The current drive efficiency of negative ion based neutral beam (NNB) was significantly improved and reached \( 1.3 \times 10^{19} \ \text{A/W/m}^2 \), which was the highest value under the neutral beam current drive scheme. An ECRF system of 110 GHz, 0.75 MW (torus injection power) has been installed to study the local heating and current drive. Effects on the tearing mode and sawtooth oscillation were observed and electron heat transport was studied using ECRF. Extensive studies on internal transport barriers have also been performed. Remarkable reduction of the L-H transition threshold power was documented, as a result of the modification of the divertor geometry. Exploratory study on the resistive wall mode was started. Termination of runaway electrons was also intensively investigated to contribute to the safety design of next step device.

1.1.1 Sustainment of high performance and non-inductive current drive

Efforts to sustain high fusion performance in high current reversed shear (RS) plasmas were attempted. Through optimization of the evolution of plasma beta by using the stored energy feedback control, we could sustain DT-equivalent fusion power gain \( Q_{\text{DT}}^{\text{eq}} \sim 0.5 \) \( (n_e(0)T_e(0) \sim 4 \times 10^{20} \text{m}^{-3} \text{keV}\cdot\text{s}) \) quasi-stationarily for 0.8 s or nearly equal to the energy confinement time in a 2.4 MA RS discharge with an L-mode edge.

A quasi-steady RS plasma accompanying internal transport barriers (ITBs) with a large fraction (~80%) of bootstrap current was realized under the full non-inductive current drive condition. This result demonstrates the basic scenario for steady state operation of advanced tokamak. High confinement \( (H_{99} = 3.3-3.8, \ \text{HH}_{98\%} = 2.1-2.3) \) and high beta \( (\beta_n = 1.9-2.2) \) were sustained for 2.7 seconds or 6 times the energy confinement time \( (\tau_E) \) in a plasma with \( B_T = 3.4 \ T, I_p = 0.8 \ MA, q_{95} \sim 9. \) Here \( H_{99} \) denotes the confinement enhancement factor to the L-mode scaling, \( \text{HH}_{98\%} \) denotes the confinement enhancement factor to the ELMy H-mode scaling, and \( \beta_n \) is the normalized beta. Full non-inductive current drive was achieved with large fraction of bootstrap current together with beam driven current (~25% of plasma current) by tangential beams. The role of bootstrap current for preventing the shrinkage of radius of \( q_{\text{min}} \) and for the sustainment of current profile was confirmed.

Improvement of \( \beta_n \) of stationary RS plasmas sustained by the lower hybrid (LH) current drive was pursued. Feedback control of the distance between the plasma surface and the fist
wall of the vessel in the low-field side was employed and it was found very effective to keep the efficient coupling of LH wave to the plasma. As a result, $\beta_N > 1.5$ was sustained for 1 s with $I_p = 0.85$ MA, $B_T = 2.0$ T and $q_{95} = 4.4$.

With application of NNB injection, the regime of high $\beta_p$ H-mode plasmas with a high fraction of non-inductive current drive, high $\beta_N$ and high confinement was expanded toward a higher $I_p$ regime. A nearly full-CD with $\beta_N = 2.4$ and $H_{pp} = 2.56$ was obtained in a high $\beta_p$ H-mode plasma with $I_p = 1.5$ MA, $B_T = 3.7$ T, $q_{95} = 4.2$. NNB (360 keV, 3.4 MW) and co-tangential positive ion based neutral beam (85 keV, 4.1 MW) were injected for current drive. The calculated non-inductively driven current amounted to 92% of the plasma current. The current drive efficiency $\eta_{\text{CD}}$ of NNB increased with central electron temperature and the value for the above discharge reached $1.3 \times 10^{19}$ A/W/m². This is the highest efficiency in the neutral beam current drive in the world.

An EC wave injection system was installed in 1999 for local heating and current drive. The system has one gyrotron, which is designed to generate 1 MW for 5 s at 110 GHz. Torus injection power was ~0.75 MW. Two more gyrotrons are planned to be installed in 2000. In 1999, preliminary experiments on EC current drive were done and changes in MSE (motional Stark effect diagnostics) pitch angles were observed during EC wave injection into a low density plasma. The current profile control by combination of LH and EC injection was also attempted, where the coupling of injected EC wave to the LH-driven electrons and the modification of LH-driven current profile by EC injection were observed.

1.1.2 Confinement and transport

Extensive studies on ITBs have been performed including threshold power, effect of radial electric field, structures, electron heating, fluctuation measurements.

The power required to form an ITB in a RS plasma was found almost independent to $B_T$.

The momentum source around the ITB layer was found affect the ITB structures and core confinement properties through the change in $E_x$ shear. This result suggests the possibility of control of ITB strength by changing the momentum input.

The structure of ITB was studied in detail. The lower boundary of width of ITB in $T_i$, $\Delta_{\text{ITB}}^i$ was related to the poloidal gyro radius of thermal ion at the ITB shoulder, $\rho_{\text{shoulder}}$. The location of ITB foot, $\rho_{\text{foot}}$ was inside the radius of $q_{\text{min}}$, $\rho_{\text{min}}$ when the gradient of $T_i$ in the ITB layer was large while $\rho_{\text{foot}}$ was located outside $\rho_{\text{min}}$ when the $T_i$ gradient was small.

The negative magnetic shear was found to be formed during LHCD even when the target q profile had $q(0)$ less than unity and the formation of steep gradient in $T_e$ was related to the formation of ITB in the negative shear.

To address the issue whether good confinement is kept under dominant electron heating regime, which is expected in fusion reactors, confinement properties of RS plasma sustained
by LHCD and heated by EC were studied. Clear ITBs and high confinement of $H_\text{99} = 1.8$ were sustained for high $T_e/T_i$ regime with $T_e/T_i = 2$ ($T_e(0) < 6$keV).

By using an X-mode reflectometer, the increase in the radial correlation length in the ITB layer was found during the ITB degradation through a mini collapse in RS plasmas.

Abrupt variations of heat diffusivity were found during the ITB formation in RS plasmas. They occurred within a few milliseconds in a wide region (~30% of plasma minor radius).

As for the H-mode plasmas accompanying edge pedestals, L-H transition threshold power, pedestal structures and confinement degradation in a high-density regime were studied.

The L-H transition threshold power was investigated systematically in successive several tens of discharges with a fixed configuration, $I_p$ and $B_t$, scanning the density up to 60% of the Greenwald density. Remarkable reduction (~30%) of the L-H transition threshold power was found in the region of line-averaged density of 2-3 x10^{19} m^{-3} compared to the open divertor. By using DEGAS code, it was found that the reduction of L-H transition threshold power could be ascribed to the changes of local distribution of neutral particle density and poloidally different local effect of neutral particles on the L-H transition.

Edge pedestal parameters were compared between JT-60U and DIII-D H-mode plasmas and a new scaling on pedestal width was proposed.

Causes of degradation of thermal energy confinement performance with increase in density were analyzed in ELMy H-mode plasmas. The confinement enhancement factor for the energy stored in the pedestal was found kept almost constant while that for the energy stored in the core was found to decrease with the density. The reduction in $T_i^{\text{ped}}$ due to strong gas puffing seems to cause increase in thermal diffusivity in the core.

1.1.3 MHD instabilities and high energy particles

The disappearance of giant (type I) ELMs and appearance of minute grassy ELMs were observed in H-mode plasmas when the triangularity $\delta$, edge safety factor $q_{95}$ and $\beta_p$ were high enough. An edge stability analysis, which was done through the collaboration with GA, suggested that the edge plasma was accessing the second stability regime of high $n$ ballooning mode in the grassy ELMy discharges.

The dependence of onset $\beta_n$ of tearing modes on electron density and collisionality were studied in high $\beta_p$ H-mode plasmas. Tearing mode stabilization using ECRF (~0.75 MW) was attempted and significant reduction of MHD perturbations was observed while complete stabilization was not obtained.

The stabilizing effects of wall were studied in RS plasmas and H-mode (positive shear) plasmas. In both discharges, beta limit seemed to be improved with a large volume
configuration satisfying \( \frac{d}{a} \leq 1.3 \) (\( d \) is the wall radius and \( a \) is the plasma minor radius), and the highest \( \beta_n \) in RS experiments (\( \beta_n \sim 2.8 \)) was achieved. MHD perturbations attributed to resistive wall modes were observed before the disruption.

Stability of RS plasmas was studied focusing on the resistive MHD modes. It was found that double tearing modes were plausible instabilities for major collapses that tended to occur in the condition that the edge safety factor \( q^* \sim \text{integer} \). The experimental results were supported by numerical analysis using the linear stability analysis code MARG2D.

Instabilities with a frequency sweep in a frequency regime of Alfvén eigenmode were studied by using NNB. The threshold of fast ion beta to destabilize these modes was found \( \langle \beta_h \rangle \sim 0.07-0.1\% \) and higher than that to destabilize a continuous TAE. Through the collaboration with PPPL, the analysis of Alfvén eigenmode experiments by HINST code was started.

Effects of ECH/ECCD on sawtooth oscillations were studied. EC wave injection near the \( q=1 \) surface drastically extended the sawtooth period both for co- and counter- injection cases.

Preliminary experiments on D-\(^3\)He fusion was done by injecting D\(^+\) by NNB into a RS plasma into which \(^3\)He was injected by gas puffing and \( \sim 110 \text{ kW} \) of D-\(^3\)He fusion power was evaluated from gamma ray measurements.

### 1.1.4 Plasma control and disruption

Several real-time feedback control schemes were newly developed in 1999, such as the controls of outer gap between the plasma and the wall, plasma stored energy, local temperature gradient, central electron temperature, main plasma radiation power, divertor pressure. The control of plasma stored energy is now often used, especially for RS plasmas, in the routine operation.

Low voltage plasma startup with electric field of 0.24 V/m, which satisfied the ITER requirements, was achieved by using ECRF.

The spontaneous and intrinsic termination of runaway electron current, which took place when the plasma surface safety factor became 2 or 3, was confirmed. The impurity gas puff was found effective to reduce the Halo current during disruption.
1.2 Divertor and Boundary Plasma

A pumping slot in the outer side of the W-shaped divertor was opened since 1999, and pumping from the private flux region in the inner and outer divertors (both-side pumping) has been used to increase effective pumping rate for high density and detached divertor operations. Experiments of helium exhaust, impurity seeding such as argon and neon, and studies of divertor detachment, SOL flow, helium transport and impurity sputtering processes were carried out. Effect of bypass between the inner and outer divertors under a dome on the particle recycling was also investigated.

An increase in the pumping rate in the high density operation was observed for the cases of a small separation between the pumping slot and the outer strike-point (outer-"gap" ≤ 2 cm) with the high NB power injection ($P_{NB} = 10–20$ MW). At the same time, low values of $Z_{eff}$ (2.3–2.6) was maintained during the x-point MARFE, compared to those for the inner-side pumping cases ($Z_{eff} = 2.6–3$). However, those improvements were rather deteriorated for the large gap case (outer-"gap" < 2 cm). Recycling neutral fluxes in the divertor and the in-out asymmetry were relatively similar for the inner-side case and the both-side pumping case with the small outer-"gap", which may be determined mostly by ion flux to the divertor.

Helium confinement ratio, $\tau_p^{H}/\tau_E$, ($\sim 2.8$) for the attached divertor of the ELMy H-mode plasmas was also reduced by 45% compared to that for the inner pumping cases. The improved value was maintained also under the detached divertor condition up to the Greenwald density factor of 90%. Helium recycling processes in the divertor was investigated using IMPMC code, and was found that both effects of charge-exchange process and electron impact ionization processes are important to determine the helium ion and neutral profiles.

The SOL plasma flow has been measured at the outer midplane and the x-point in the low power L-mode ($P_{NB} = 4.4$ MW). The plasma flow pattern did not change for the ion $\nabla B$ drift towards the divertor and puff-and-pump: flow reversal at the outer midplane and flow towards the outer divertor at x-point. However, in the partially-detached divertor, an increase in the plasma flow at the x-point was enhanced compared to that for inner-pump case, maybe due to pumping from the private flux region. Neutral pressures at inner-side dome, outer-side dome and dome top were systematically measured with fast ionization gauges. The results showed that the neutral pressure at the outer-side dome was increased during the divertor detachment, and that the pressure range from $\sim 1$ Pa to 4 Pa was largely extended for the small outer-"gap" compared to that for the inner pumping cases ($\leq 1$ Pa). Pumping at both-sides of the divertor was favorable to extend the operation regime of the partially-detached plasma without appearance of the x-point MARFE, and with in-out symmetrical profiles of particle recycling was maintained.

Chemical processes of heavier hydrocarbons such as C$_2$D$_2$ and C$_2$D$_4$ were intensively
investigated, and found that the total sputtering yield of carbon chemical yield is relatively constant (\( \sim 2\% \)) with increasing the particle flux.

### 1.3 High Density and High Energy Confinement

Maintaining good energy confinement plasma at high density with a reduction of the heat load to the divertor plates is the most important issue for a tokamak reactor. However, degradation of the energy confinement (and H-factor) was generally observed in the high density operations. An increase in neutral density at the main plasma edge caused by large deuterium gas puffing was correlated to the degradation. Using impurity seeding such as neon, good energy confinement was maintained at high density with reducing the deuterium gas puff rate, in the open divertor and in the W-shaped divertor with inner-side pumping. However, control method of the high radiation power from the main plasma edge was not developed yet. Feedback control system of the radiation power using bolometer signals viewing the main plasma was newly developed to control the puff rate of the impurity gases.

The divertor pumping rate was increased using the both-side divertor pumping in 1999, and experiments using more intensive radiators such as argon/krypton were, for the first time, performed for the high power ELMy H-mode plasma in order to extend the high density operation of the good confinement plasmas under the detached divertor condition. Relatively good energy confinement (\( H \sim 1.4 - 1.5 \)) of the ELMy H-mode plasmas were maintained up to 70\% of Greenwald density under the detached divertor condition. H-factor was 30–40\% larger than that for the deuterium gas puff plasmas. Understanding the improvement mechanism in the high density regime is very important for the large tokamak experiments. The analysis of impurity and energy transports was in progress under a collaboration work with PPPL. The knowledge is also useful to explore the radiative improved (RI) mode, which has been obtained in limited plasma discharges of TEXTOR, DIII-D and JET.

As for the steady-state reversed shear plasmas with the internal transport barrier (ITB), impurity seeding of neon and argon were injected to increase the radiation fraction to obtain the detached divertor. At the same time, sustaining the good core confinement region with input power feedback control and contamination of the impurity ions inside ITB were investigated for the different impurity species. Helium removal from the relatively week ITB was so far successful. However, in the strong ITB, sustaining the RS plasma with helium beam injection and the helium removal method were under investigation. Developments of the diagnostics for the heat flux studies such as IRTV, bolometer array, and the tomography analysis of radiation power distribution (under a collaboration work with GA) are in progress.
2. Sustainment of High Performance and Non-inductive Current Drive

2.1 Sustainment of High Fusion Performance in High Current Reversed Shear Plasmas

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1. Introduction

In JT-60U RS plasmas, very high confinement \( \langle n_e \rangle > 3 \) with an L-mode edge is obtained due to significantly reduced transport in the internal transport barrier (ITB) layer that is located at a large radius. The equivalent fusion power gain, \( Q_{\text{DT}} = 1.25 \) was achieved transiently at \( I_p = 2.6 \text{ MA} \) and \( B_s = 4.4 \text{ T} \). In these plasmas, the duration of high performance was limited by beta collapse that was encountered when \( q_{\text{min}} \) became 2. Here, results of efforts to sustain high fusion performance as long as possible are presented.

2. Attempts to obtain an H-mode edge

It was observed that passing through \( q_{\text{min}} \approx 2 \) without collapses or disruptions was possible in ELMy H mode edge RS plasmas at 1.5 MA and 3.5 T. Hence we tried to make an H-mode edge in high current (>2 MA) high field (4.3 T) RS plasmas using a similar technique used to obtain an H-mode in 1.5 MA plasmas or changing NB toroidal momentum input from balanced one to co-directional dominant one. In the discharge shown in Fig. 1, an on-axis counter NB (#7) was turned off and switched to an off-axis co NB (#10) at \( t = 5.7 \text{ s} \). The peripheral electron density, which was measured by YAG-laser Thomson scattering, started to increase at \( t = 5.8 \text{ s} \) and ELMs appeared at \( t = 6.1 \text{ s} \) as shown in Fig. 1.

Thus, we successfully obtained an H-mode at the high \( I_p \) regime using the off-axis co NB. However, in the discharge shown in Fig. 1, ELMs disappeared and the peripheral density started to decline at \( t = 6.8 \text{ s} \) after the off-axis co NB was switched to on-axis counter NB again at \( t = 6.4 \text{ s} \). The peripheral electron density returned to the L-mode level at \( t = 7.9 \text{ s} \) and the discharge terminated into a disruption with \( q_{\text{min}} \approx 1.9 \). When we continued the off-axis co NB injection, we could certainly sustain the H-mode edge, but collapses were encountered around \( t \approx 6.5 \text{ s} \). These collapses are supposed to be related to locked modes since the toroidal rotation at the plasma edge became nearly zero for co-directional injection case (the edge rotates in the counter direction for

\[ \beta_n = 1.09, \quad q_{\text{min}} \approx 1.9 \]

\[ P_{\text{NBI}} [\text{MW}] \]

\[ W_{\text{dia}} [\text{MJ}] \]

\[ D_0 \text{ Div} [\text{A.U.}] \]

\[ Q_{\text{DT}} = 0.3 \]

\[ Q_{\text{DT}} = 0.3 \times Q_{\text{DD}} \]

**Fig. 1.** A RS discharge in which ELMy H mode was obtained by off-axis co-NB (#10) injection.
balanced injection case in JT-60U, which is mainly due to formation of negative radial electric field through ripple loss of beam ions). It is not yet possible to sustain the H-mode edge in high current, high field RS plasmas in JT-60U.

3. Attempts to pass through $q_{\text{min}}=2$ in L-mode edge plasmas

It was also attempted to reduce the beta or to change the toroidal rotation just before $q_{\text{min}}$ became 2 in order to pass though $q_{\text{min}}=2$ in L-mode edge plasmas. For the balanced NB injection case, collapses were observed even with low $\beta_N$ of 0.8~1 and it was found that passing through $q_{\text{min}}=2$ by simply reducing $\beta_N$ was difficult at least in high performance experiments. We attempted the counter injection to passing through $q_{\text{min}} = 2$ considering the possibility of stability improvement by the toroidal rotation or its shear. Figure 2 indicates comparison between the balanced NB case (E34328) and the counter NB case (E34338). In both discharges, injected NB power was stepped down to only 4 MW (2 units) after $t = 6.3$ s. In E34328, the discharge terminated into a disruption at $t = 7.1$ s with $\beta_N$ of $\sim0.8$, while in E34338, the discharge survived beyond $t\sim7.1$ s though a mini collapse was observed. At $t = 7.7$ s, $q_{\text{min}}$ became $\sim1.7$ in E34338 and a clear ITB was observed for the ion temperature ($T_i$) profile. In Fig. 2 (b), profiles of $T_i$ and toroidal rotation velocity ($V_T$) at $t = 7.1$ s are shown for two discharges. The values of $V_T$ in E34338 were certainly smaller than those in E34328 by $\sim0.7\times10^5$ m/s, but the difference in the velocity shear was not large. The smaller ITB radius and a little bit weaker pressure gradient in E34338 may have some effects for stability. Optimization of this technique for passing through $q_{\text{min}}=2$ with higher $\beta_N$ and with larger $q_{\text{min}}$ radius has not been done.

![Fig. 2. (a) waveforms of a discharge in which counter NBs were injected and $q_{\text{min}}=2$ was passed without disruption. (b) Profiles of $T_i$ and $V_T$ at 7.1 s in a discharge shown in (a) (counter rotation) and in another discharge with balanced injection.](image-url)
4. Sustainment of high fusion performance

Since it was found difficult to pass through $q_{\text{min}}=2$ in the high $I_p$ regime, we optimized the evolution of $\beta_n$ or we intended to raise $\beta_n$ as early as possible to extend the high performance period. For this purpose, we employed the stored energy feedback control technique that had been developed. The waveforms of a typical discharge is shown in Fig. 3. In this discharge, the stored energy ($W_{\text{dia}}$) feedback control started at $t = 4.0$ s and continued until the end of discharge. In the second box of Fig. 3, the reference value of $W_{\text{dia}}$ by a dotted line and the measured value of $W_{\text{dia}}$ by a solid line are shown. It is found that $W_{\text{dia}}$ was controlled precisely according to the reference value. $W_{\text{dia}}$ was ramped up from 4 s to 4.8 s to form an ITB with a large radius and was kept constant to suppress collapses at $q_{\text{min}}=3$. After $t = 5.6$ s, $W_{\text{dia}}$ started to increase and reached 5.3 MJ at 6.2 s and was kept constant after that. As the $W_{\text{dia}}$ increased, $H_{89}$ also increased. The timing of ramp up of $W_{\text{dia}}$ (from 2.7 MJ to 5.3 MJ) was adjusted to reach high performance (5.3 MJ) state as early as possible without suffering collapses. As a result, $Q_{DT}^{eq} = 0.5$ was reached at 6.25 s and was sustained until 7.05 s or the beta collapse at $q_{\text{min}}=2$. The duration was 0.8 s or nearly equal to the energy confinement time ($\tau_E = 0.8-0.9$ s). During this period, $\beta_n=1.1-1.2$, $H_{89}=2.5-2.7$, $\langle H_{99} \rangle = 1.4$, $T_e(0)=12-14$ keV, $n_0(0)\tau_e T_e(0) \sim 4 \times 10^{20}$ m$^3$ keV s, were sustained at $B_T=4.35$ T, $I_p=2.4$ MA, $q_{99}=3.4$, $\kappa$ (elongation) = 1.84 and $\delta$ (triangularity) = 0.05.

The sustained duration of $Q_{DT}^{eq}$ and $P_{D_T}^{eq}/P_{\text{NB}}^{abs}$ are plotted in Fig. 4. The discharge E27969 was one in 1996 (before the divertor modification to the W-shaped one) where $Q_{DT}^{eq}=1.05$ was achieved at 2.8 MA. The discharge E34290 was a similar one to E34292 with somewhat lower performance with a longer duration. The sustained duration for $Q_{DT}^{eq} > 0.4$ has been extended from 0.85 s to 1.05 s in E34290. In is noted that $dW_{\text{dia}}/dt/P_{\text{abs}}$ was 30-40%
in E27969 where $\beta_n$ continued to increase until the disruption while it was less than 10% in E34290 and E34292. As a result, the performance in $P_{DT}^{eq}/P_{NB}^{abs}$ was improved significantly; the duration for $P_{DT}^{eq}/P_{NB}^{abs} > 0.45$ was extended 0.5 s to 0.85 s.

5. Summary

Efforts to sustain high fusion performance in high current reversed shear plasmas were attempted. Attempts to sustain an H-mode edge and to pass through $q_{\text{min}} = 2$ were done but they need more adjustments. Through optimization of the evolution of plasma beta by using the stored energy feedback control, we could sustain $Q_{DT}^{eq} \sim 0.5$ quasi-stationarily for 0.8 s or nearly equal to the energy confinement time in a 2.4 MA reversed shear discharge with an L-mode edge. The sustained performance in $P_{DT}^{eq}/P_{NB}^{abs}$ was improved significantly compared to the previous higher $Q_{DT}^{eq} (\sim 1)$ discharges; the duration for $P_{DT}^{eq}/P_{NB}^{abs} > 0.45$ was extended 0.5 s to 0.85 s.

References
2.2 Quasi Steady Reversed Shear Plasmas with High Bootstrap Current Fraction and High Confinement


1. Introduction

In ELMy H reversed shear (RS) plasmas, internal transport barriers (ITBs) were sustained for several seconds without suffering beta collapses.\(^1\) However, the ITB radius continued to shrink according to the shrinkage of reversed shear region caused by penetration of inductive current, which resulted in confinement degradation. Non-inductive current drive is necessary to sustain a hollow current profile stationarily. RS plasmas accompanying ITBs were sustained by lower hybrid current drive with normalized beta \(\beta_N\) of \(~0.9\) and poloidal beta \(\beta_p\) of \(0.7\) in JT-60U.\(^2\) The bootstrap current fraction was estimated \(~23\%\). Higher beta and higher bootstrap current fraction (>70\%) are required for steady state operation of tokamak reactors to reduce the circulating power for the non-inductive current drive.\(^3\) It is also expected that the hollow current profile can be sustained stationarily by the bootstrap current, which is peaked at ITB layer. Here, we report on the demonstration of quasi-steady high-confinement RS plasma with ITBs by the large bootstrap current.

2. High bootstrap current fraction, high confinement reversed shear plasma

To realize high bootstrap current fraction, RS plasmas with an ELMy H-mode edge and with high triangularity (\(\delta \sim 0.4\)) in a high q regime (\(q_{95} = 7.5-9\)) were optimized. A high \(q_{95}\) operation was employed to obtain high \(\beta_p\), which is necessary to achieve high bootstrap current fraction, within the attainable beta limit, which was typically \(\beta_N \sim 2\) in JT-60U RS plasmas.\(^1\) A typical configuration is shown in Fig. 1 together with NB lines used in this experiment. Here, \(B_t\) is 3.4 T, \(I_p\) is 0.8 MA, the triangularity (\(\delta\)) is 0.43 and \(q_{95} \sim 9\). A high value of \(\delta\) was employed to obtain high \(\beta_N\) since improvement of \(\beta_N\) with \(\delta\) had been obtained in JT-60U high \(\beta_p\) H-mode plasmas.\(^4\) Two units of co-directional to \(I_p\) tangential beam are employed for the NB current drive. Injected NB power was 2-2.2 MW per unit. Since off-axis current drive is believed to be necessary to sustain a RS profile, the magnetic axis of the plasma was elevated so that the trajectory of one unit passed through \(\rho \sim 0.5\) while that of the other near the magnetic axis (\(\rho \sim 0.1\)). Here \(\rho\) denotes the normalized radius. The former is denoted by “off-axis” while the latter by “on-axis” in Fig. 1.

![Fig. 1. Plasma configuration and beam lines.](image)
Waveforms of a typical discharge are shown in Fig. 2. The plasma current was initially ramped up to 1 MA and decreased to 0.8 MA later. The initial overshooting of $I_p$ was meant to reduce $q_{\text{min}}$ rapidly to an expected steady-state value and to form strong ITBs safely below the beta limit. The ITB was formed during the $I_p$ ramp and $\beta_N \sim \beta_p \sim 1.5$ was obtained during the first flattop (1 MA) with moderate triangularity ($\delta \sim 0.28$). After that $\delta$ was further increased and $I_p$ was hereby ramped down to reach the final configuration of $I_p = 0.8$ MA and $\delta \sim 0.43$ at $t = 6.9$ s. The values of $\beta_N$ and $\beta_p$ increased by the $I_p$ ramp-down and reached $\beta_N \sim 1.9$ and $\beta_p \sim 2.6$ at $t = 7$ s. The edge density (not shown here) increased at $t \sim 4.6$ s due to an H-mode transition and regular ELMs appeared at $t \sim 7.6$ s (see the bottom box of Fig. 2).

The plasma stored energy was feedback controlled to stay constant (2.2 MJ) after $t = 5$ s by adjusting the power of perpendicular beams. During $t = 7.3-10$ s the measured value exceeded the reference and only pre-programmed units were injected; two units of coparallel, half unit of counter-tangential, and one modulated (duty of 33%) unit of "on-axis" perpendicular. Then, $\tau_E = 0.4-0.5$ s, $\beta_N = 1.9-2.2$, $\beta_p = 2.6-3.2$ and $H_{89} = 3.3-3.8$ were sustained for 2.7 s ($\sim 6 \tau_E$). The thermal component of plasma stored energy was 80-82% and the enhancement factor of thermal confinement to the ELMy H-mode scaling (ITER Physics Basis 98(y.2)5), $H_{89y2}$, was 2.1-2.3. These values ($H_{89} \sim 3.5$, $H_{89y2} \sim 2.2$) are the highest ones sustained quasi-stationarily in JT-60U. Here, the fast ion loss due to toroidal field ripple or charge exchange with neutrals, which is estimated to be $\sim 20\%$, is not subtracted from the

Fig. 2. (a) Waveforms and (b) profiles of a RS plasma in which high confinement and high bootstrap current fraction were sustained for 2.7 s. In (a) from the top, plasma current ($I_p$) and NBI power ($P_{\text{NB}}$), $\beta_p$ (dotted line) and $\beta_N$ (solid line), $H_{89}$ and $H_{89y2}$, $q_{\text{min}}$ surface loop voltage $V_{\text{surf}}$, deuterium recycling emission at the divertor. In (b), from the top, ion temperature $T_i$ and electron temperature ($T_e$) profiles at 7.5 s, electron density ($n_e$) profiles at 7.5, 8.5, 9.5 s, $q$ profiles at 7.5, 8.5, 9.5 s.
heating power for the calculation of $\tau_{\text{tr}}, H_{89}$ and $H_{\text{hub2}}$. The high confinement period was terminated at $t = 10$ s when “off-axis” co-tangential NB turned off and perpendicular NBs were injected instead 0.1 s later to sustain the stored energy. The shrinkage of ITB radius was observed at $t = 10$ s, which might be caused by the rapid change of toroidal rotation.

Profiles of $T_i$ and electron temperature ($T_e$) at $t = 7.5$ s and those of $n_e$ and $q$ at $t = 7.5, 8.5, 9.5$ s are shown in Fig. 2 (b). We can see that $n_e$ and $q$ profiles were sustained almost stationarily ($q_{\text{min}} \sim 3.6$, $\rho_{\text{qmin}} \sim 0.65$, $\rho_{\text{foot}} \sim 0.7$) for $t = 7.5-9.5$ s. Here $\rho_{\text{foot}}$ denotes the normalized radius of outside edge of steep gradient in $n_e$. The steep gradients or ITB in $T_i$ and $T_e$ were also sustained.

The radial profile of calculated non-inductive current density $j_{\text{BS}}$ and $j_{\text{BD}}$ are shown in Fig. 3. The beam driven current density $j_{\text{BD}}$ was evaluated using ACCOME code. The profile of $j_{\text{BS}}$ has its peak around $\rho \sim 0.6$ because of the large pressure gradient at the ITB layer.

The fraction of bootstrap current was 80-88% during $t = 7.5-11$ s in E35037. This value is also the highest sustained quasi-stationarily in JT-60U. The amount of beam driven current was evaluated to be ~0.20 MA or 25% of $I_p$ during $t = 7-10$ s. The sum of bootstrap current and beam drive current exceeds the total plasma current for $t = 7.5-10$ s and the full non-inductive current drive is expected. This can also be confirmed from the surface voltage $\sim < 0$ shown in the bottom box of Fig. 2 (a).

The sustainment of large $\rho_{\text{qmin}}$ was attained only in high $\beta_p$ discharges. In Fig. 4, time evolution of $\rho_{\text{qmin}}$ was compared in three discharges. In E35007, the ITB disappeared at $t \sim 7.5$ s due to the decrease of heating power. Then, $\beta_p$ dropped and the position of $q_{\text{min}}$ moved quickly inward. In E35029, the ITB was sustained with $\beta_N \sim 1.6$ and $\beta_p \sim 2.4$, then $\rho_{\text{qmin}}$ was sustained at a smaller value of ~0.55 than that in E35037. These clearly indicate the role of bootstrap current for preventing the shrinkage of $\rho_{\text{qmin}}$ and for sustainment of current profile.

3. Sustainment of high beta and high confinement

Figure 5 shows $\beta_N$ versus duration in RS plasmas. In previous ELMy H-mode RS plasmas, sustained $\beta_N$ for several seconds was 1.1-1.2. Higher $\beta_N$ of ~1.6 was also obtained
but only within 1.5 s. In high $q$ (low $I_p$) regime, $\beta_n \sim 2$ was sustained for 2 s and $\beta_n > 1.5$ for several seconds.

The confinement has been also improved remarkably. In previous ELMy H mode RS plasmas $H_{99}$ was less than 1.7 \(^1\). The outstanding confinement achieved in E35037 is attributed to the large ITB radius \(^6\) as shown in Fig. 2 and the high edge stability due to high $\delta$. In previous ELMy H-mode RS plasmas with lower $\delta$ ($\sim 0.27$) and $\beta_n$ ($< 1.3$), the radii of ITB and $q_{\text{min}}$ were smaller due to the penetration of inductive current, which resulted in lower confinement. In E35029 shown in Fig. 4, where $\rho_{\text{qmin}}$ was smaller than E35037, $H_{99}$ was also lower ($\sim 2.8$) than that in E35037. Hence, it is confirmed that the high confinement can be sustained in RS plasmas if $\rho_{\text{qmin}}$ can be sustained by some non-inductive (bootstrap or external) current drive. Progress in $\beta_n$ and $H_{99}$ is illustrated in Fig. 6.

Transiently, $H_{99}=4.1$ has been achieved in a discharge (E34856), which is the highest one in JT-60U. Other typical parameters are $B_t=3.4$ T, $I_p=0.9$ MA, $q_{99}$ = 7.1, $\delta$= 0.34, $W_{\text{es}}$ = 3.45 MJ, $\beta_n$ =2.63, $n/n_{\text{GW}}$=0.64, $\tau_e$=0.55 s ($\text{d}W_{\text{es}}/\text{d}t/P_{\text{abs}}=0.19$) and $\beta_n$, $H_{99}$=10.8.

4. Summary

We have realized a quasi-steady RS plasma with a large fraction (~80%) of bootstrap current under the full non-inductive current drive condition and shown that the large radii of ITB and $q_{\text{min}}$ were sustained by the bootstrap current, which leaded to the sustainment of high confinement. This result demonstrates the basic scenario for steady state operation of advanced tokamak.

References
2.3 Sustainment of RS plasmas with $\beta_N > 1$ by LHCD

S. Ide, T. Fujita, T. Suzuki and M. Seki

1. Introduction

It has been shown experimentally that a plasmas with reversed magnetic shear (RS) can achieve improved confinement with formation of an internal transport barrier (ITB). A key issue to maintain high performance of a RS plasma is to maintain a safety factor ($q$) profile to be stable against MHD criteria. On JT-60U, application of non inductive current drive by lower hybrid waves (LHCD by LHWs) onto RS plasmas has been intensively investigated towards steady state sustainment of RS configuration with ITBs. Quasi-steady sustainment of RS configuration by mean of LHCD was firstly demonstrated in 1995 [1], but at low $\beta_N$ since applied power was low. Also ITBs were not clearly observed in the discharge. By increasing neutral beam power ($P_{NB}$), a RS plasma with clear ITBs at modest $\beta_N$ ($\sim 0.9$) was succeeded to maintain in quasi-steady state by LHCD [2]. Following these results, the next objective is demonstration of sustainment of higher $\beta_N$ by means of LHCD. In 1999 campaign, sustainable $\beta_N$ was raised up to 1.5 and the results are shown in this article.

2. Experimental Setup

The experiments were performed with deuterium discharges. Two multi-junction type launchers were used for LHW injection. A typical plasma cross-section and position of the launchers are schematically shown in Fig. 1. The one installed in a port at poloidal angle $\theta \sim 45^\circ$, consists of four rows of eight multi-junction modules (MJ) which have 3 sub-waveguides and is referred to as the “C-Launcher” [3]. The other one is installed in a horizontal port and consist of two rows of four MJ modules which have 12 sub-waveguides [4] and is referred to as the “B-Launcher”. These launchers are schematically shown in Fig. 1. One of the key parameters in wave-plasma coupling conditions is a spatial distance between the outermost closed flux surface and the first wall.

Fig. 1: A typical plasma cross section which was used in the experiments. Two multi-junction type LHW launchers are shown schematically as well. A gap between plasma and first wall which was FB controlled was indicated.
In this campaign, feedback control of plasma-wall distance was adopted in order to keep LHW coupling to a plasma. A difference in major radius between outer-tip of a plasma column and first wall at horizontal plane ($\delta_0$) was FB controlled (Fig. 1.)

3. Experimental Results

The target plasma was produced in a same manner with which RS plasmas had been produced commonly in JT-60U. Neutral beam heating was applied at the fast plasma current ($I_p$) ramp up phase in order to retard current penetration. The injection of LHW was started after $I_p$ reached flat top to maintain a $q$ profile. A discharge scenario which was basically similar to one with which ITBs were sustained successfully in 1998 campaign was adopted. Higher $\beta_N$ was pursued by simply raising $P_{NB}$ by both preprogramming and FB control. Thanks to the $\delta_0$ FB control, LHW coupling became easier to handle. Then $P_{NB}$ was FB controlled by using the stored energy FB control for easier operation. In Fig. 2 shown are temporal evolutions of $P_{LH}$, $V_e$, $\beta_N$ and $P_{NB}$ for three cases; (a) $P_{NB}$ was increased moderately, (b) $P_{NB}$ was raised further for higher $\beta_N$ and (c) $B_t$ was decreased in order to raise $\beta_N$.

![Fig. 2: Temporal evolutions of $P_{LH}$, $V_e$, $\beta_N$ and $P_{NB}$ for three cases; (a) $P_{NB}$ was increased moderately, (b) $P_{NB}$ was raised further for higher $\beta_N$ and (c) $B_t$ was decreased.](image)

![Fig. 3: The electron temperature and density profiles, from the top box corresponding to each cases shown in Fig. 2.](image)

The electron temperature and density profiles measured by YAG Thomson scattering measurement are shown in Fig. 3 for each cases at later phase of LHCD. As shown in the figure, clear ITBs are found to be maintained during LHCD in each cases.

As shown in Fig. 2, effect of LHCD can be observed in $V_e$. As LHCD was applied $V_e$
started to decrease due to non-inductive LHCD. Especially in 2.0 T cases in which $V_I$ fell to almost zero during, although not in full period, LHCD indicating most of the current was driven by LHWs. In the case of 1.7 T (Fig. 2 (c)), decrease in $V_I$ is less than that in the other cases, suggesting less driven current. This might be attributed to poorer accessibility of the LHW at the low $B_t$.

The safety factor profile was tried to be estimated for the experiments from the MSE measurement. However, due to unstable conditions in one of the MSE systems, which has more measurement points, quality of the $q$ profile reconstruction was limited. The $q$ profile estimated at 7.5 and 9 s in E34533 are shown in Fig. 4. That at 6.5 s in E34539 are also shown in the same figure. Unfortunately in the later phase of the discharge, the MSE signals were blinded by background radiation. So the $q$ profile at the later phase was not available. It is suggested that a RS configuration was well maintained during LHCD phase. In Fig. 5, $\beta_N$ is plotted against time duration in which that value of $\beta_N$ was maintained. As shown in the figure, $\beta_N > 1.5$ was maintained about 1 second.

![Fig. 4: The safety factor profiles measured in the discharges E34533 and E34539.](image)

### 3. Summary

In RS plasmas with LHCD, sustainable $\beta_N$ value was tried to be raised. The NB heating power was increased so as to increase $\beta_N$ and also $B_t$ was decreased to push $\beta_N$ further.

![Fig. 5: $\beta_N$ is plotted against time duration in which that value of $\beta_N$ was maintained.](image)

### Reference


2.4 Extension of the High Integrated Performance Regime with High Current Drive Efficiency of NNB

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abstract

With increasing plasma current and application of the negative ion based neutral beam (NNB) injection, JT-60 expanded the discharge regime with a high fraction of non-inductive current drive (bootstrap + neutral beam current drive), high $\beta_N$, and high confinement toward ITER. The highest NB current drive efficiency of $1.3 \times 10^{19}$ A/W/m$^2$ was achieved in this series of experiment.

1. Introduction

With its main purpose to provide physics bases for ITER and for concept optimization of the steady-state tokamak reactor, JT-60U has been optimizing the tokamak operation scenario for simultaneous achievement and sustainment of 1) high confinement, 2) high $\beta_N$, 3) high bootstrap current fraction, 4) full non-inductive current drive and 5) efficient heat and particle exhaust [1-5]. For the integration, we have utilized the weak positive shear ELMy H mode (high-$\beta_p$ ELMy H-mode) and the reversed magnetic shear ELMy H-mode characterized by both the internal transport barrier (ITB) and the edge transport barrier, and expanded the discharge regime toward ITER. This section reports the results obtained in the high-$\beta_p$ ELMy H-mode.

2. Integrated performance and High Current Drive Efficiency

From the integration point of view, Fig.1 shows the achieved levels which consist of 6 axes; H-factor ($=\tau_{\text{ITER}} / \tau_{\text{ITER89P}}$), $\beta_N$, $f_{\text{BS}}$ (fraction of bootstrap current to plasma current $I_p$), $f_{\text{CD}}$ (fraction of non-inductive driven current to $I_p$), fraction of radiated power to absorbed heating power, and line averaged electron density normalized by the Greenwald density. The full scale for each axis corresponds to the design value of SSTR. The design values of an example of the reduced cost ITER steady-state operation are also indicated. In Fig.1(a), the high-$\beta_p$ ELMy H-mode data at $I_p = 1$MA is shown by the bold solid lines with shadowed area. This discharge almost satisfies the requirements of SSTR and ITER-steady-state for H-factor, $\beta_N$, $f_{\text{BS}}$ and $f_{\text{CD}}$ [1]. However, for density and radiation power, integration has not been completed. The dotted lines correspond to another 1MA discharge with strong gas puffing. It is possible to increase density and radiation, however confinement is degraded largely. Therefore, increased density and radiation are the key remaining issues for integration.
Figure 1. Integration of the key plasma performances. The full scale values correspond to the design values of SSTR. (a) The full non-inductive current drive high- $\beta_p$ ELM'y H-mode discharge (shadowed area) at $I_p=1$MA and the high density ELM'y H discharge (1MA) with strong gas puffing (dotted lines). (b) The near full non-inductive current drive high $\beta_p$-ELM'y H-mode discharge at $I_p=1.5$MA with N-NB. In (a) and (b), design values of an example of the reduced cost ITER steady-state operation are also shown.

Figure 2. Time evolution of the near full non-inductive current drive high- $\beta_p$ ELM'y H-mode discharge at $I_p=1.5$MA and $B_t=3.7$T with N-NB (360keV, 3.4MW), tangential P-NB (85keV, co-4.1MW, counter-0.8MW) and perpendicular P-NB (12.8MW).
Figure 3. Profiles of electron density, electron temperature and ion temperature for E33885.

We recently extended the plasma current region with integrated core plasma performances of H-factor, $\beta_N$, $f_{BS}$ and $f_{CD}$ close to the ITER requirements up to $I_p = 1.5$ MA by N-NB injection (Fig.1(b)) [5]. The time traces and profiles of this high-$\beta_p$ ELMy H-mode discharge E33885 are shown in Figs.2 and 3. We could achieve nearly full-CD with H-factor=2.56 ($HH(y2)=1.1$), $\beta_N=2.4$ at $I_p=1.5$MA, toroidal field $B_t=3.7$T, elongation $\kappa=1.6$, triangularity $\delta=0.35$ and $q_{95}=4.2$ with N-NB (360keV, 3.4MW) and P-NBs (85keV) of co-tangential (4.1MW) for current drive and heating, counter-tangential (0.8MW) for MSE measurement and heating (but anti drive) and perpendicular (12.8MW) for heating. The one turn surface voltage is decreasing during the N-NB pulse ($t=5.9 - 6.48$s) and the value is negative. The internal inductance is almost constant in the later phase of the N-NB pulse and the internal poloidal field by MSE measurement is also constant. (Before N-NB injection, although the one turn voltage is close to zero, the resistive voltage is still positive because of negative values of $\frac{d\phi}{dt}$). According to calculation by the ACCOME code, the total beam driven current is 670 kA including 410kA by N-NB, and the bootstrap current is 710kA. In total, the calculated non-inductive driven current fraction is 92%. Figure 4 shows that current drive efficiency $\eta_{CD}$ of N-NB is increasing with central electron temperature [2,6] and the value for E33885 is $1.3 \times 10^{19}$ A/W/m$^2$ [5], which is the highest efficiency for the neutral beam current drive. In this discharge, $\eta_{CD}$ of the positive ion based NB (P-NB) is $0.8 \times 10^{19}$ A/W/m$^2$.

Figure 4: Current drive efficiency of N-NB increasing with central electron temperature. The maximum value reached $1.3 \times 10^{19}$ (A/W/m$^2$).
Figure 5: Expansion of the discharge regime with high fraction of non-inductive current drive > 70 %, high $\beta_N > 2.4$ and high confinement $H > 2$ toward ITER on the $\rho_p^* - v_e^*$ plane.

By increasing plasma current, we have expanded the discharge regime with high a fraction of non-inductive current drive > 70%, high $\beta_N > 2.4$ and high confinement $H > 2$ toward ITER. Figure 5 shows such expansion on the $\rho_p^* - v_e^*$ plane, where $\rho_p^*$ is normalized poloidal Larmor radius and $v_e^*$ is collisionality.

References

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2.5 Current profile control by combination of LH- and EC- injection

M. Seki, O. Naito, T. Kondoh, S. Ide, Y. Ikeda and JT-60 Team

The Lower Hybrid Current Drive (LHCD) is the most promising method to drive plasma current with high efficiency in the tokamak machines. Especially, a large amount of non-inductive driven current up to 3.6 MA has performed by LHCD by using high-directive spectrum in JT-60U. And current profile control was also attained by changing N/N spectrum of multi-junction type LH antenna. On the other hand, current drive techniques aiming for high efficiency had been studied by injection LH and electron cyclotron wave (EC) at the same time in the small and medium tokamak machines [1, 2]. Even in JT-60U, current drive experiments by means of LH injection with EC newly installed has started from 1999. The experiment was focused on current profile control study by changing injection angle of EC instead of changing spectrum of LH in this campaign.

The current profile control by combination of LH- and EC- injection was characterized by increase in hard x-ray signal, namely enhanced energetic electrons compared with LH only case. For example, the intensity of hard x-ray signal depends on toroidal magnetic field under a spectrum of LH and an angle of EC. This means that the injected EC wave couples with LH-driven electrons via resonance condition with doppler effect and relativistic effect. If the resonance condition satisfies at the central part, the enhanced energetic electrons lead peaking current profile. And then current profile control by changing injection angle of EC was tried under the same spectrum of LH as shown Fig.1. The injection angles of EC were 30, 39 and 51 deg. in poloidal direction. The EC power was ~700kW of X-mode at 110GHz, and the LH power was ~1000kW at 2GHz, respectively. Plasma current was 1MA and toroidal magnetic field was 3T at center. The electron cyclotron resonant position was high-field-side peripheral plasma as indicated in Fig.1. According to experimental data, internal inductance li remarkably decreased at 39 deg case.

![Diagram](image_url)

*Fig. 1. Injection angles of EC.*
This experimental result suggests that current profile becomes flat by EC injection during LHCD. Hard x-ray profile shown in Fig.2 is consistent with behavior in li. This allows us to control current profile by changing injection angle of EC under the fixed spectrum of LH. Fortunately, the fixed spectrum of LH leads simplification in LH system, because spectrum control is not required for changing current profile.

Furthermore intensity of hard x-ray signal is large in the case of the injection angle of 30 deg. It seems that absorption of EC power is weak by bulk plasma. And this un-absorbed EC power might ionize neutral gases in front of LH antenna, indeed, reflection coefficient was affected during EC injection. This result showed the possibility of improvement of coupling property through increasing electron density in front of LH antenna even in a long distance between LH antenna and outside plasma.

It is shown experimentally that current profile is controlled by injection angle of EC wave during LHCD with the same spectrum. This combination of LH- and EC-injection could drive plasma current efficiently even in the peripheral region.

3 Physics of Plasma Confinement

3.1 On threshold power for ITB formation in JT-60U RS plasmas [1]

S. Ide, Y. Sakamoto, T. Suzuki and T. Fujita

It has been shown experimentally that a plasmas with reversed magnetic shear (RS) can achieve improved confinement with formation of an internal transport barrier (ITB). Study of conditions to form ITB is one of most important issues in such a RS plasma research not only from a view point of unveiling the physics mechanisms but also seeking possibility of having improved confinement plasmas with ITBs in a fusion plasma. There are many physics and operational parameters which can be keys for ITB formation. Among them, the toroidal magnetic field \( B_t \) is one of the most important ones since \( q \) changes directly with \( B_t \). Also, from a view point of constructing a new machine in which improved confinement by ITB formation, how small (or large) \( B_t \) can be is a very crucial issue.

On JT-60U, the dependence of power required to form an ITB in a RS plasma \( (P_{th}) \) on \( B_t \) was investigated. In the experiments the neutral beam (NB) power \( (P_{NB}) \) was scanned during the \( I_p \) ramp-up phase at different \( B_t \) with other parameters, including \( I_p \), and the plasma configuration as fixed as possible. The experiment was carried out both in deuterium and hydrogen plasmas with the same configuration and operational scenario but with different \( n_e \).

The experimental results show that the minimum absorbed power required ITBs to form is almost independent to \( B_t \). Both the experiments on deuterium and hydrogen show the same independence. However, required power was different between deuterium and hydrogen experiments, which might be attributed to difference in \( n_e \). The power absorbed by the plasma, which differs from the injected power due to shine through, ripple loss and so on, was evaluated by using the OFMC code for some cases. The deposition profile differs between the hydrogen and the deuterium cases.

Further investigation on ITB formation will be continued, concerning other parameters.

Reference
3.2 Structures of Internal Transport Barrier in JT-60U Reversed Shear Plasmas


1. Introduction

It is important to understand the structures of ITB because reversed shear discharge, which is the leading scenario of steady state operation for ITER, have achieved high confinement owing to the ITB formation. In JT-60U reversed shear plasmas, internal pedestals were formed in the density and temperature profiles, which had the ITB shoulder and the ITB foot. It has so far been reported that the spatial width of the pedestal of ETB scales with poloidal gyro radius of thermal ions. However structures of ITB have not been investigated in detail. Investigations of ITB structures such as the spatial width and the location are important for the understanding of physics mechanisms of ITB.

2. Width of internal transport barrier

In JT-60U reversed shear plasmas, thermal and particle diffusivities quickly decreased to the neoclassical level within the narrow region. As a result, the internal pedestals were formed in the density and temperature profiles; these profiles had steep gradients for the ITB layer and reduced gradients for other regions. In order to investigate the detail $T_i$ and $V_T$ profiles measured by CXRS, JOG experiment was performed in some discharges, where plasma was moved inward with 1 m/sec for the steady state period. The detail $T_i$ and $V_T$ profiles are shown in Fig. 1. The $T_i$ profile is flat inside the ITB and the $V_T$ profile has notched structure near the ITB. This flat profile is a characteristic common with $T_e$ and $n_e$ profiles even for a peaked heating power deposition profile. Figure 2 shows the temporal evolution of the ion temperature profiles with the ion banana width and the safety factor profiles in the reversed shear plasma. In the core region, the large banana width about 0.3 m is formed because of the large safety factor about 10 and high ion temperature about 15 keV. This could
be one possibility of the flat profile. According to the recent neoclassical theory, however, the banana width close to the magnetic axis is modified to the potato-shape orbits in the reversed shear configuration, which is shorter than that estimated from standard neoclassical theory. Further discussion is necessary for the formation of the flat profiles. The notched structure frequently appears in the measured the $V_T$ profiles of the carbon impurity in the reversed shear plasmas.

It is interesting whether common physics exists in both ETB and ITB. It has been reported that the spatial width of the pedestal of ETB scales with poloidal gyro radius of thermal ions. In addition, general properties of electrostatic turbulence suggest that the barrier width scales as the ion gyro radius or possibly the ion poloidal gyro radius. The relation between $\Delta_{\text{ITB}}^{\text{Ti}}$ and the poloidal gyro radius, $\rho_{\text{pi,shoulder}}$, is shown in Fig. 3, where $\rho_{\text{pi,shoulder}}$ is evaluated at the shoulder. It is clearly seen that the lower boundary of $\Delta_{\text{ITB}}^{\text{Ti}}$ was related to the $\rho_{\text{pi,shoulder}}$.

The internal transport barriers in the JT-60U reversed shear plasmas were formed in ion temperature, electron temperature and electron density profiles. Figure 4 shows the comparison of ITB width in ion temperature, electron
temperature and electron density. The ITB width of electron temperature is approximately the same as that of ion temperature. On the other hand, the ITB width of electron density tends to be wider than that of ion temperature, provided that electron density profiles have some gradient inside ITB.

3. Location of internal transport barrier

The safety factor value and profile affect the internal transport barrier formation and propagation in the core improved confinement plasmas. The ITB appeared at first time in a core region and then moved outward and final location of the ITB stagnated near the location of $q_{\text{min}}$, $\rho_{\text{qmin}}$, in the reversed shear plasmas.

Figure 5 shows ion temperature, safety factor profiles in the case of (a) narrow ITB width with steep gradient of $T_i$ and (b) wide ITB width with reduced gradient of $T_i$. The $\rho_{\text{foot}}$ located inside $\rho_{\text{qmin}}$ in the case of (a). On the other hand, the $\rho_{\text{foot}}$ located outside $\rho_{\text{qmin}}$ in the case of (b). Figure 5(c) shows the relation between the gradient of $T_i$ and the difference of $\rho_{\text{foot}}$ and $\rho_{\text{qmin}}$. The reduced gradients of $T_i$ seem to allow $\rho_{\text{foot}} > \rho_{\text{qmin}}$. These results suggest that the outward propagation of the ITB is limited by increasing instability drive such as high n ballooning mode when the local gradient at the ITB foot goes to positive shear region across the $q_{\text{min}}$ surface.

Figure 6 shows the relation between $\rho_{\text{shoulder}}$ and $\rho_{\text{qmin}}$, where $\rho_{\text{qmin}}$ is the location of minimum magnetic shear. The strong correlation of these locations was obtained. The reason
of this correlation is not clear. If the large bootstrap current is formed in the ITB layer, $\rho_{\text{shoulder}}$ locates near $\rho_{\text{min}}$. In the case of off-axis and broad current drive by lower hybrid wave, however, $\rho_{\text{shoulder}}$ locates near $\rho_{\text{min}}$. The causality of this correlation will be examined by modification of current profile in future.

Fig. 6. the relation between $\rho_{\text{shoulder}}$ and $\rho_{\text{min}}$

REFERENCES
3.3 Role of Radial Electric Field and Plasma Rotation in the Time Evolution of Internal Transport Barrier in JT-60U


The effects of radial electric field and plasma rotation on the characteristics of internal transport barriers (ITBs) and core confinement properties are studied in JT-60U reversed shear plasmas in quasi steady state phase. The combination of on-axis and off-axis tangential NBI in the co and counter directions in JT-60U allows a variety of the toroidal rotation profiles. Four different combinations of on-axis and off-axis tangential NBI, that is (a) [CO,CO], (b) [CO,CTR], (c) [CTR,CTR] and (d) [CTR,CO] are compared. Here, [CO,CTR] denotes the combination of on-axis co tangential injection (#9 beam) and off-axis counter tangential injection (#8 beam), for example. Plasma parameters are $I_p = 1.5 \text{ MA}$, $B_t = 3.8 \text{ T}$, $q_{eff} = 5.7$ and $P_{abs} = 5.6 \sim 6.6 \text{ MW}$. The radial electric field, $E_r$, is calculated by the momentum balance equations and heat momentum balance equations parallel to the magnetic field. The momentum source profile estimated by the OFMC code is taken into account.

In the case of (a) [CO,CO] and (c) [CTR,CTR], in which the momentum source is unbalanced around the ITB layer, the "notch" feature of toroidal rotation velocity, $V_t$, of carbon impurity at the ITB disappears rapidly. The $E_r$ shear becomes weak and the gradients of $T_i$, $T_e$ and $n_e$ at ITB layer decreases. The core energy confinement is remarkably degraded. The similar results were also obtained in the extended reversed shear experiments in TFTR.

On the other hand, the case of (b) [CO,CTR] and (d) [CTR,CO], in which the momentum source is nearly balanced around the ITB layer, the "notch" feature of $V_t$ and the strong $E_r$ shear remain during the NBI heating. The core energy confinement property is maintained. It is clear that the sustainment of the ITB layer and core energy confinement in the reversed shear plasmas is strongly affected by the momentum source profile at the ITB layer.

The strong unbalanced co rotating plasma with slightly larger absorbed power ~ 8 MW is also analyzed. Although the momentum source is large at the ITB layer, profiles of $T_i$, $T_e$ and $n_e$ do not change and the core energy confinement is retained. The degradation of ITB layer by unbalanced momentum source is compensated by the additional core heating power, which promotes the pressure gradient at the ITB layer.

Reference
3.4 Correlation measurement during ITB degradation by a minor collapse in JT-60U reversed shear plasma

K. Shinohara, R. Yoshino, R. Nazikian, and T. Fujita

Introduction

In JT-60U, high performance plasmas with an internal transport barrier, ITB are obtained with reversed or weak magnetic shear profile. It is important to understand what happens and reduces the transport in the ITB region in order to operate such high performance plasmas in fusion reactors. Here we show a result of radial correlation measurement in the ITB region during ITB degradation in JT-60U reverse shear plasma.

Diagnostics

Correlation measurement are performed by using a core correlation reflectometer[1]. The reflectometer consists of four channels, two of which operate in fixed frequency and the other two channels are tunable. The frequencies of the launched waves are 115 and 130 GHz for the fixed frequency channels, and $122.5 \pm f_{\text{bank}}$ GHz, where $f_{\text{bank}} = 2.73, 5.28, 6.37, 6.93$ and $7.27$ GHz, for the variable channels. The tunable channels can step through the five frequencies every 60 ms. The correlation of the fluctuations is determined from the correlation between fixed and variable frequency channels and therefore the radial profile of the correlations can be measured every 60 ms in a discharge. The right hand polarized extraordinary, X-mode, is used. Quadrature phase detection is used to measure the complex amplitude, namely electric field, of the reflected wave. The cut-off position of the reflectometer is located within the ITB region of the electron density.

Measurement at minor collapse

Figure 1 shows the temporal evolution of the discharge with a minor collapse. The toroidal field and the plasma current are 3.73 T and 1.5 MA respectively. The discharge gas is hydrogen.

Large difference between $T_i(\rho = 0.31)$ and $T_i(\rho = 0.54)$ at 5.5 − 6 s in Fig. 1(c) indicates the existence of the ITB of the ion temperature. A minor collapse occurs near 6.1 s. Drops of the electron density, the stored energy and the ion temperature can be seen in Figs. 1(b) and (c). The ITB of the electron density relaxes after the collapse as shown in Fig. 2. The correlation is plotted in Fig. 3 as a function of the radial separation, $\delta r$, of two channels. The correlation is that of the complex amplitude for frequencies with highest correlation with higher than 10 kHz. The measurement location lies in almost the middle of the ITB of the electron density as shown in Fig. 2. The complex amplitude in the range of the frequency can be dominated by the wave scattered by the density fluctuations near the cut-off layer, and the radial profile of the correlation of the complex amplitude can be used to
infer that of the density fluctuations. The measured value is fit by an exponential function, 
\( \exp(-\delta r / \Delta L) \). The radial decay length, \( \Delta L \), which can scale with the correlation length, is 
also shown in the figures. The decay length increases from 12 mm to 62 mm, which 
suggests that the density correlation length significantly increases after the minor collapse. 
Therefore, in the case of an abrupt degradation with a minor collapse, the relaxation of ITB 
correlates with the enhancement of the radial correlation length of the electron density 
fluctuations.

**Discussion and Summary**

The radial mode width of the electrostatic drift wave in the linear theory is estimated by 
the equation, \( |\rho, L_n / s|^{1/2} \), where \( \rho \) is the ion Larmor radius, \( L_n \) is the scale length of the 
ion temperature and \( s = r/q dq/dr \) is the magnetic shearing parameter [2]. The radial mode 
width estimated by the above equation is 20 – 40 mm and is comparable with the observed 
radial decay length.

By using an X-mode reflectometer, radial correlation measurement in the ITB region 
have been performed for the first time during ITB degradation phase in a JT-60U reversed 
shear plasma. It was observed that the radial decay length became longer when the ITB of 
the electron density deteriorated in an abrupt degradation case with a minor collapse.

**References**

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Figure 1: Waveform of R/S plasma with a minor collapse at about 6.1s. The toroidal field is 3.73 T. The discharge gas is hydrogen.

Figure 2: Profile of electron density. Open circle shows the value at 6.05s. Closed square shows the value at 6.3s. The measured point is shown by an arrow.

Figure 3: Profile of coherence of measured complex amplitude. Open circle shows the value at 6.05s. Closed square shows the value at 6.3s. The decay length is shown.
3.5 **Abrupt Variations of Heat Diffusivity during ITB Formation in Reverse Shear Discharges** [1,2]


New features of the space-time evolution of the internal transport barrier (ITB) were highlighted during recent JT-60U reverse shear (RS) experiments. An ITB evolution in RS plasmas is often a combination of the fast time-scale processes and the gradual ones. Fast time-scale processes are the common intrinsic features of JT-60U RS plasmas and are seen as the simultaneous (within a few milliseconds) rise and decay of electron temperature ($T_e$) on two zones separated by a region without variation of $T_e$ ("bipolar" perturbation). The region without variation of $T_e$ is located near the position of the minimum safety factor profile for many fast processes. The present robust result is that the region of fast-time-scale improvements of electron heat diffusivity is wide in space (around 0.3 of the minor radius) and well extended to the zone of $T_e$ decay [1].

A new source of Heat Pulse Propagation (HPP) is found in JT-60U RS plasmas. HPP is created by the abrupt in time and wide in space variation of heat diffusivity. Such phenomena called "events" are seen as the simultaneous rise and decay of $T_e$ on two zones. For the event, the region of strong $T_e$ rise ($\sim 20$ keV/s) is well localized ($\sim 4$ cm) in space initially. Later in time, a slow diffusive broadening of the rising $T_e$ perturbation is seen throughout ITB region. The HPP is studied analytically and numerically. Values of the electron heat diffusivity as low as $\sim 0.1$ m$^2$/s are found in the region with $\sim 8$ cm width fully located in positive shear zone. A similar low value of the ion heat diffusivity is obtained for ion HPP. An important consequence of HPP analysis is an absence of electron and ion "heat pinch" in the ITB region [2].

**References**


3.6 Confinement improvement in a region of $T_e > T_i$ in JT-60U [1]

S. Ide, Y. Sakamoto, T. Suzuki and T. Fujita

Study of improved confinement plasmas with internal transport barriers (ITBs) is one of the most important issues toward steady state operation of a fusion plasma. To date, the best confinement performance has achieved with neutral beam (NB) heating with the beam energy from a few tens to about a hundred keV. With such NB heating, most of the power is firstly absorbed by the ions, moreover quite a large amount of particles are fueled by the beams therefore the fueling profile is strongly coupled to the NB power deposition. However, on the other hand in a fusion plasma, heating is dominated by the $\alpha$ particles and most of the power is transferred to the electrons firstly. Also the situation of particle fueling would be very different from the current NB heating experiments. Therefore, it is necessary to investigate improved confinement characteristics with electron heating dominant and also with low central fueling, or fueling decoupled from the heating profile. On JT-60U, by using a variety of the electron heating tools, such as N-NB and RF systems, the region of improved confinement study has been extended towards dominant electron heating and low fueling situations.

To approach the issue, confinement improvement was investigated against the ratio of $T_e/T_i$ in the plasma core region ($T_e/T_i(\text{core})$) of RS plasmas. In JT-60U, both in high $\beta_p$ and reversed magnetic shear (RS) plasmas with good confinement, explored region was that where $T_e/T_i < 1$, except a deuterium RS discharge with LHCD [2] and an N-NB induced H-mode plasma [3]. Now data have been extended to the $T_e/T_i \geq 1$ region in RS plasmas utilizing LHRF and ECRF system. Experiments were mainly conducted in hydrogen plasmas. In a RS plasma with usual NB heating at the current rampup phase, $I_p = 0.9$ MA and $B_t = 3.7$ T at the plasma center, LHRF power of 1.5 - 2 MW was injected at the current flat top in order to sustain ITBs. The injected NB power was 5 - 7 MW. Operated range of the central electron density was $1 - 2.5 \times 10^{19}$ m$^{-3}$. ECRF heating of 0.75 MW was also applied to increase $T_e/T_i$ further. Even with LHCD only, $T_e/T_i$ could be higher than unity in some cases. ITBs were found to be maintained. This suggests that improved confinement characteristics remained at ITB region. By applying ECRF central heating, $T_e/T_i$ increased further. Resultant range of the central $T_e$ was 2.5 - 6 keV. As a result, confinement improvement factor $H^{\text{op}}$ of about 1.8 was maintained even with $T_e/T_i = 2$. (In the estimation of $H^{\text{op}}$, 50% of $P_{\text{NB}}$ was assumed to be lost due to ripple, orbit and charge exchange losses.) Improved confinement is expected to be maintained in those plasmas since, clear ITBs were still kept existing.

Reference
3.7 Effects of LHCD on electron temperature profile

Y. Ikeda, and K. Ushigusa

Peaked electron temperature profiles are often observed during lower hybrid current drive (LHCD) experiments in JT-60. In some cases, the electron temperature at the plasma center increases up to \(\sim 14\) keV, but the time constant of its increase is usually much longer than the energy confinement time and comparable to the current diffusion time (several seconds). To investigate on the relation between the evolution of plasma current profile and the electron temperature profile, LHCD experiments were conducted with current profile measurement by the motional Stark effects (MSE) diagnostics.

The lower hybrid waves were injected into a plasma with \(I_p = 0.8\) MA, \(B_p = 2.1\) T, \(q_{\text{eff}} = 4.8\), \(\bar{n}_e = 0.8 \times 10^{19} \text{ m}^{-3}\), from \(t = 7.5\) s to \(t = 12.5\) s. This relatively late injection of LH was to allow the current penetration and wait for the current profile to settle in a stationary state. The LH power was 1.5 MW, the wave refractive index parallel to the magnetic field (\(N_g\)) was 1.75, and the plasma current was fully driven by LH waves (the one-turn loop voltage was \(-0.04 \sim -0.08\) V). Immediately after the beginning of LH injection, the electron temperature at 5% of the minor radius started to increase and continued to increase during 5 seconds of LH injection. The electron temperature at 27% of minor radius also increased from 1.5 keV to 2.5 keV in 0.3 seconds, but after that it remained almost constant during the rest of the LH injection period.

The safety factor profile \(q(r)\) measured by MSE was initially monotonic and its value at the plasma center was below unity (\(~ 0.9\)). Concomitantly, a sawtooth oscillation with a period of 50 ms was observed before the injection of LH, but it disappeared immediately after the LH injection. One second after the LH injection, \(q(r)\) near the plasma center flattened and rose above unity. Three seconds after the LH injection, the magnetic shear within 25% of the minor radius became negative. Correspondingly, the electron temperature at 18% of minor radius increased, and formed an internal transport barrier (ITB) at about one fourth of the minor radius.

When the LH driven current profile was less off-axis either by decreasing \(N_n\) or increasing the safety factor, the central electron temperature increased more rapidly initially, but then an instability was induced and a crash in the core electron temperature occurred. Also, no apparent ITB in the electron temperature profile was observed, but this might be due to the insufficient number of measurement points in the plasma core. On the other hand, when the LH driven current profile was more off-axis, no significant increase in the central electron temperature was observed.

The identification of LH deposition profile and the transport analysis are now under way.
3.8 Reduction of L-H transition threshold power under the W-shaped pumped diverter geometry in JT-60U

T Fukuda, T Takizuka, K Tsuchiya, Y Kamada and N Asakura

Remarkable reduction of the L-H threshold power was documented in JT-60U under the W-shaped pumped diverter geometry, in comparison with the results of previous open divertor. The range of density was extended to 0.6n_{GWL}(=Ip/πa^2), and apparently stronger than linear density dependence was found at high density. Accordingly, the threshold power scaling has been re-established in the international database group and modified to the form

$$P_{th} = 3.24 B_T^{0.75} [n_{20}]^{0.6} R^{0.98} a^{0.81} M^{-1}.$$  \hspace{1cm} (1)

The prediction interval was reduced to 32 MW for ITER-FEAT, whereas the previous scaling (2), for which the open divertor results were consistent with, provides 57 MW.

$$P_{th} = 0.45 B_T [n_{20}]^{0.75} R^2 x (0.6 n_{20} R^2)^{+0.25}.$$  \hspace{1cm} (2)

However, the new scaling (1) considers the JT-60U results as a scatter and is not concerned with the strong density dependence stated above. If it were to be included, the scaling takes the form

$$P_{th} = 9.46 x 10^{-3} B_T [n_{19}]^{1.25} R^3.$$  \hspace{1cm} (3)

and it predicts 93 MW for ITER-FEAT. Here, the regression analysis was performed only with the JT-60U database to derive the scaling (3), where the reduction of edge temperature as well as the nonlinear increase of edge density at high density were also observed. Although the scalings (2) and (3) show substantially different density exponents, the threshold power exhibit similar dependence on n_e^{0.95}, regardless of the divertor geometry. Therefore, it is suggested to elaborate a scaling with n_e^{0.95}.

The effect of divertor geometry on the L-H transition threshold power, which has long been an issue of controversy, was investigated to the detail with emphasis on edge plasma quantities, including the poloidal distribution of neutrals. The reason why the neutrals are emphasized is that atomic processes cannot be ignored to understand the L-H transition at the plasma edge, especially at a high density, where a large amount of fuel gas is supplied from the outside.

The performed neutral particle density analysis with DEGAS indicates that at lower n_e^{0.95}, neutral penetration length becomes larger, and its negative contribution to the threshold power can be predominant. Namely, the edge ion collisionality (ν^{-*}) defined at right before the L-H transition is slightly reduced, partly owing to the reduction of Z_{eff} in the W-shaped divertor. However, due to the decrease of density and consequent increased heating efficiency, the power threshold is notably reduced. At medium to high density region, however, n_o^{0.95} near the X-point in the W-shaped divertor is substantially larger than that of the open divertor. On the other hand, ν^{-*} is near unity, whilst it is much lower in the open divertor, although the high density data is scarce for the open divertor. It indicates that the negative contribution of the edge neutral particles, related to the charge exchange friction loss, disappears with an increase of density, which might be ascribed to the scattering model of trapped ions outside the separatrix near the X-point. Here, it should be noted that n_o^{0.95} at the midplane for the W-shaped divertor is approximately 60% of that of the open divertor. Further indications is that the value of ν^{-*} starts to decrease at high density close to the Greenwald limit, which may be indicative of the fact that the negative influence of the edge neutrals near the midplane or inside the separatrix is taking over the scattering effect near the X-point. Thus, the negative CX and positive scattering effects of neutral particles compete with each other in different density range and contribute to the apparently complex dependence of the density dependence of the threshold power.

References
3.9 Role of Neutral Particles for L-H Transition under the Different Divertor Geometry in JT-60U


1. Introduction

The JT-60U divertor geometry was recently modified to W-shape with pumping capability from both sides of the septum in the private region, for the effective particle and heat flux control. In the systematic density scan (up to 60% of Greenwald density) dedicated for the H-mode studies, remarkable reduction of the H-mode threshold power of more than 30% was observed in the region $\bar{n}_e=2-3\times10^{19}\text{m}^{-3}$, compared to the open divertor, as shown in Fig.1. In order to clarify the causalities, related to the reduction of the threshold power, we have investigated the effect of radiation power and neutral particles of which unfavourable effect for H-mode transition is theoretically predicted. As to the radiation power, difference between the open and the W-shaped divertors is not large enough to account for the apparent reduction of 30% at the maximum. Therefore, we have focused on the analysis of edge neutrals, considering that an excess of neutrals in the edge would hinder the H-mode transition.

2. Evaluation of poloidally local neutral density

We were continuously researching about the effect of edge neutral particles for H-mode transition in JT-60U from a period of open divertor geometry. We used DEGAS code in order to evaluate edge neutral density in this analysis. In open divertor case, poloidally averaged neutral density normalized electron density ($n_0/n_e$) at plasma edge was correlation with edge collisionality ($v_{i,\text{eff}}^*$), where $v_{i,\text{eff}}^*$ denotes as follows,

$$v_{i,\text{eff}}^* = \frac{n_e Z_{\text{eff}}}{n_i Z_i^2} \frac{16\sqrt{\pi} e^4 \ln \Lambda Z_i^4}{32m^{1/2} (2T)^{3/2}} \frac{3em^{1/2} (2T)^{1/2}}{e^{1/2} \left( \frac{2T}{m_i} \right)^{1/2}}$$

In the basis of analysis using poloidally averaged value, $n_0/n_e$ in the case of W-shaped divertor became larger than that of open divertor case, as shown in Fig.2. This means that transition power became smaller with edge neutral density. This seems to be a phenomenon against the
theoretical prediction that edge neutral particles prevent the H-mode transition. Therefore, we newly analysed the neutral densities at some points along separatrix in order to evaluate the poloidally local effect for the H-mode transition.

Figure 3 shows line-averaged density dependence of neutral densities (a) near the X-point and (b) at the midplane. Here, we use the calculated density at the mesh just inside separatrix as edge neutral density. In W-shaped divertor case, edge neutral density near the X-point right before H-mode transition became larger in comparison with the open divertor case.

In this evaluation, so far, we excluded the particle source from the outer baffle plate due to recycling. Therefore, neutral density at the midplane seemed to be underestimated. The evaluation including this local source is in progress. However, this effect will be small against the neutral density near the X-point. In the following discussion, we focused the activity of neutral particles near the X-point.

![Fig. 3](image)

**Fig. 3** Edge plasma density dependence of edge neutral densities (a) near the X-point and (b) at the midplane. In this evaluation, so far, local recycling source from outer baffle plate was not included. Therefore, neutral density seemed to be underestimated, especially at the midplane.

### 3. Discussion

In the previous section, we found that neutral density near the X-point was larger over the whole density range in the W-shaped divertor. On the other hand, large neutral density near the X-point provides lower normalized threshold power in the range of $n_e = 2 \sim 3 \times 10^{19} \text{m}^{-3}$ as
shown in Fig. 4. This evaluation result is consistent to the result of Dα emission. Figure 5 shows the density dependence of Dα signal intensity at a sightline on the X-point right before the H-mode transition in the range of $\bar{n}_e=2\sim3\times10^{19} \text{m}^3$ where reduction of transition power was clearly observed. The Dα intensity is larger for the W-shaped divertor, which indicates that the neutral density is higher near the X-point in the W-shaped geometry, in comparison with the open divertor.

Reduction of the threshold power with increased neutral density near the X-point is also documented at JFT-2M tokamak4), and it is theoretically interpreted in a way that scattering of trapped ions by neutral particles can enhance the formation of radial electric field.5) As well as this, it was considered that the changes of local neural particle density distribution was contributed to reduction of transition power in JT-60U.

4. Summary

In the basis of analysis using poloidally local neutral density, we found that poloidal distribution of neutral particles was changed by the modification of divertor geometry, and compression of neutral particle density towards the X-point could result in the reduction of the H-mode threshold power. Thus, we have shown that reduction of threshold power in W-shaped geometry can be ascribed to the changes of local neural particle density distribution and poloidally different local effect of neutral particles on H-mode transition.

References

Fig. 4  Transition power as a function of neutral density near X-point in the region of $\bar{n}_e = 2-3 \times 10^{19} \text{m}^{-3}$. Transition power tended to become smaller with neutral density near X-point.

Fig. 5  Density dependence of $D\alpha$ intensity at the sightline on X-point just before H-mode transition. In this region of density, $D\alpha$ intensity increased after modification of divertor geometry.
3.10 Degradation of Thermal Energy Confinement of ELM My H-mode Plasmas in JT-60U

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1. Introduction

High density operation with improved energy confinement is essential for a tokamak reactor to produce sufficient fusion power. ITER is envisaged to operate ELM My H-mode discharges at a high density close to the Greenwald density limit $n_{GW}^*$. However, the $H$-factor has been observed to decrease with an increase in plasma density in many experiments [1-4]. Therefore, the clarification of the dominant causes of this degradation is the urgent issue in recent tokamak research. ELMs are considered to affect the energy confinement mainly by two basic mechanisms. One is the direct energy loss from the region near the plasma boundary. The other is an influence of the edge pedestal structure imposed by the destabilization of ELMs on the thermal energy confinement of the core plasmas. The stiff temperature profiles in H-mode plasmas, which are the evidence for an edge-core relationship, have been found on the other devices [2,4-7].

2. Density dependence of thermal energy confinement

ELMy H-mode experiments were performed in JT-60U at $I_p = 1.8$ MA, where $n_{GW}^*$ corresponds to $(8.1-8.7) \times 10^{19}$ m$^{-3}$. The toroidal magnetic field $B_t = 3.0$ T and $q_{95} = 2.9-3.1$. Neutral beam injection (NBI) power $P_{NBI}$ was scanned in steps from 4 to 13 MW. With deuterium gas puffing, the line-averaged electron density $\bar{n}_e$ was varied on a shot by shot basis from $2.4 \times 10^{19}$ to $4.5 \times 10^{19}$ m$^{-3}$. The maximum $\bar{n}_e$ reached was $(0.50-0.53) \times n_{GW}^*$. Elongation $\kappa$ of 1.48 to 1.55 and triangularity $\delta$ of 0.16 to 0.19 were fixed. The plasma volume $V_p$ is in the range of 60-63 m$^3$. The plasma major radius $R_p$ and the minor radius $a_p$ were in the ranges of 3.24-3.25 m and of 0.81-0.85 m, respectively.

2.1. Density dependence of the core and edge pedestal confinement

Figure 1 shows the $H$-factors of ELMy H-mode plasmas with $P_{NBI} = 8-13$ MW as a function of $\bar{n}_e / n_{GW}^*$. The $H$-factor of thermal plasma, $H_{th}^{LJT-60}$, decreases continuously from 1.6 to 1.0 with an increase in $\bar{n}_e / n_{GW}^*$ from 0.30 to 0.53. In low density plasmas, high $\beta_p$ H-mode is observed in some cases where an internal transport barrier (ITB) can be clearly formed. Figure 2(a) and (b) plot the electron density at the shoulder of the H-mode pedestal, $n_{eA}^{ped}$, and at the center, $n_e(0)$, as a function of $\bar{n}_e$. In type-I ELMy phase, $n_{eA}^{ped}$ and $n_e(0)$ increase in proportion to $\bar{n}_e$. The electron and ion temperatures at the shoulder of the H-mode pedestal, or $T_{eA}^{ped}$ and $T_{iA}^{ped}$, decrease from 1.6 to 0.8 keV and from 2.8 to 1.0 keV, respectively (see figure 2(c)). The central temperature of ions, $T_i(0)$, gradually goes down to the central temperature of electrons, $T_e(0)$, with an increase in $\bar{n}_e$ as shown in figure 2(d).

By evaluating the energy stored in the pedestal component, $W_{th}^{ped}$, and in the core component, $W_{th}^{core}$, their density dependences are shown in figure 3(a). Here, $W_{th}^{ped}$ and $W_{th}^{core}$ are given as:
\[ W_{\text{th}}^{\text{red}} = \frac{2}{3} k_{B} \int_{P} \sum_{j} n_{j}^{\text{red}} T_{j}^{\text{red}} dV_{p} \]  
(1)

\[ W_{\text{th}}^{\text{core}} = W_{\text{th}} - W_{\text{th}}^{\text{core}} \]  
(2)

With an increase in \( \bar{n}_{e} \) from \( 2.9 \times 10^{19} \) to \( 4.5 \times 10^{19} \) m\(^{-3} \) at \( P_{\text{NBI}} = 13.0 \) MW, \( W_{\text{th}}^{\text{ped}} \) and \( W_{\text{th}}^{\text{core}} \) are almost constant in the range of 0.8-0.9 MJ and of 1.2-1.5 MJ, respectively. However, it should be noted that \( W_{\text{th}}^{\text{core}} \) increases in the low density region only with beam fueling: \( \bar{n}_{e} = (2.7-2.9) \times 10^{19} \) m\(^{-3} \).

Figure 3(b) shows the enhancement factors of the core and pedestal confinement (\( H_{\text{ONL}}^{\text{core}} \) and \( H_{\text{ONL}}^{\text{ped}} \)) based on the offset nonlinear scaling \([8]\). It is observed that \( H_{\text{ONL}}^{\text{ped}} \) gradually decreases from 0.7 to 0.5 with an increase in \( \bar{n}_{e} / n_{GW}^{\text{ped}} \), while \( H_{\text{ONL}}^{\text{core}} \) remarkably decreases from 1.3 to 0.8. Here, \( H_{\text{ONL}}^{\text{ped}} \) is considerably small in these discharges. This result is observed at a low triangularity \( \delta \), in which the edge pressure gradient becomes small. Besides, it may be worth pointing out, in passing, that ITER H-mode confinement database predicts \( H_{\text{ONL}}^{\text{ped}} \approx 0.75 \) for JT-60U ELMy H-mode plasmas, where the pedestal component is statistically determined as the offset part of \( W_{\text{th}} \), independent of the heating power \( P_{\text{abs}} \) \([8]\).

2.2. Correlation between the core and edge pedestal confinement

The classification of ELM behavior in \( (n_{e}^{\text{ped}}, T_{e}^{\text{ped}}) \) space is shown in figure 4. The arrow (i) indicates a relatively high density discharge. Both \( n_{e}^{\text{ped}} \) and \( T_{e}^{\text{ped}} \) increase through type-III ELM region with the heating power after the L-H transition, and type-I ELMs appear when \( T_{e}^{\text{ped}} \) increases further with high power heating. In type-I ELM phase, an increase in \( n_{e}^{\text{ped}} \) is accompanied by a reduction in \( T_{e}^{\text{ped}} \) so that the energy stored in the pedestal can be kept constant. The arrow (ii) indicates a relatively low density (no type-III phase) discharge. Since the L-H transition occurred at
quite a low density, ELM-free phase has been sustained until type-I ELMs appeared. The confinement degradation can be linked to the relatively low pedestal temperatures because \( n_{\text{ped}} \) and \( T_{e,\text{ped}} \) correspond with each other in type-I ELMy phase. Figure 5(a) indicates the variations in \( H_{\text{ONL-Core}}^{\text{core}} \) and \( H_{\text{ONL-Pedestal}}^{\text{core}} \) in type-I ELMy phase as a function of \( T_{i,\text{ped}} \). A continuous increase of \( H_{\text{ONL-Core}}^{\text{core}} \) is observed with an increase in \( T_{i,\text{ped}} \), although the variation of \( H_{\text{ONL-Pedestal}}^{\text{core}} \) is small. However, \( H_{\text{ONL-Pedestal}}^{\text{core}} \) is saturated with a further increase in \( T_{i,\text{ped}} \). Here, it should be noted that the saturated region of \( H \)-factor corresponds to the low density region without deuterium gas puffing, in which \( W_{\text{th}}^{\text{core}} \) can increase with density as shown in figure 3(a).

The influence of the pedestal structure on thermal energy transport of the core plasma is evaluated by using the effective thermal conductivity of core component \( \chi_{\text{eff}}^{\text{core}} \):

\[
\chi_{\text{eff}}^{\text{core}} = \frac{Q_{\text{core}}}{\sum n_j V T_j}
\]

(3)

where \( Q_{\text{core}} \) is a heat flux across the torus surface at \( r_{\text{ped}} \) and the temperature gradient \( V T \) was derived from \( \Delta T(0) = T(0) - T_{\text{ped}} \) as a characteristic value which can determine the energy transport of the core plasma:

\[
Q_{\text{core}} = \frac{P_{\text{th}} - dW/dt}{4\pi^2R^2 r_{\text{ped}}}
\]

(4)

\[
\sum n_j V T_j = \sum n_j \frac{\Delta T_j(0)}{r_{\text{ped}}}
\]

(5)

In the low pedestal temperature region, \( \chi_{\text{eff}}^{\text{core}} \) gradually decreases with an increase in \( T_{i,\text{ped}} \) (see figure 5(b)). For \( T_{i,\text{ped}} \geq 2.2 \text{ keV} \), a slight increase or constancy of \( \chi_{\text{eff}}^{\text{core}} \) is observed as expected in figure 5(a). The reduction in \( T_{\text{ped}} \) caused by the increase in \( n_{\text{ped}} \) due to gas puffing seems to bring about the deterioration of the core energy confinement in a low pedestal temperature region. It is observed that once a critical temperature is exceeded the impact of the pedestal temperature on the energy confinement weakens. Consequently, the edge temperature and density may play a significant role as the boundary condition in determining the thermal energy confinement of the plasma core.

3. Discussions

It should be discussed whether the degradation of thermal energy confinement of the core plasma is caused by ELM bursts or an underlying mechanism of thermal energy transport connecting the edge
pedestal with the core plasma. If a steep pressure gradient were the driving source of ELM burst, the destruction of pressure profile at the edge could be avalanchingly conveyed towards the center. Since it is reported that a reduction in the edge temperature due to the H-L back transition brought about a reduction in the core temperature in time scale much shorter than the energy confinement time $\tau_E$ [9], ELMs are considered to be the phenomena not merely located at the edge plasma but there is a large transport structure connected with the core plasma. As a matter of fact, the energy confinement during H-mode depends in many cases strongly upon the temperature at the shoulder of the H-mode pedestal \cite{2,4,7} as shown in figure 5. With an increase in gas puffing rate, $T_{\text{Ped}}$ decreases continuously because of the constancy of $W_{\text{th, Ped}}$ during type-I ELMs, and $\chi_{\text{core}}$ increases and the $H$-factor decreases simultaneously. However, the edge temperature $T_{\text{Ped}}$ itself is determined by the limiting pressure gradient, the pedestal width and the plasma density, so that each causality of them must be clarified in the future study. From these observations, if the higher sustainable $W_{\text{th}}$ is produced in ELMs H-mode discharges, $T_{\text{Ped}}$ is expected to be the higher even in a high density region and the higher $H$-factor is obtained. Although the plasma configurations were fixed in a series of these experiments, the large pedestal width and large $W_{\text{th, Ped}}$ are obtained in ELMs H-mode plasmas with high triangularity as shown in figure 6. These discharges have achieved thermally high energy confinement \cite{10}.

4. Conclusions

The dominant causes of the degradation of thermal energy confinement with increasing plasma density were analyzed in ELMs H-mode plasmas. The energy stored in the pedestal, $W_{\text{th, Ped}}$, is kept almost constant because of the action of type-I ELMs, while the core component, $W_{\text{th, Core}}$, also tends to remain constant. The enhancement factor of the core confinement remarkably decreases with $\tilde{n}_e$. The reduction in $T_{\text{Ped}}$ due to strong gas puffing seems to bring about an increase in $\chi_{\text{core}}$ in a low pedestal temperature region. An increase in the energy stored in the pedestal plasma due to high triangularity is expected to produce thermally improved energy confinement even at a high density.

References

3.11 Comparison of Edge Pedestal Parameters between
JT-60U and DIII-D H-mode Plasmas

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Edge pedestal parameters (pedestal temperature and density, pedestal width) between JT-60U and DIII-D are studied. For the discharges used in the study, the JT-60U H-mode operation space is in the high edge ion temperature and low edge electron density region, whereas DIII-D discharges have higher edge electron density and lower electron temperature. The pedestal width scaling has been studied separately in each machine. We tested these pedestal scalings against each other by using JT-60U ion temperature pedestal width data and DIII-D electron temperature pedestal width data. Three previous scalings for pedestal width ($\Delta \propto \varepsilon^{0.5} \rho_{\text{ion}}^{0.3} \Delta R^{0.66}$, $\Delta R \propto (\rho_{\text{ped}}/R)^{0.4}$) are tested using temperature pedestal widths from both machines. The relations between pedestal width and dimensionless parameters are examined. We propose a new pedestal width scaling for $T_e$ and $T_i$ based on this study of the two machines. The new scaling includes normalized poloidal gyroradius and Greenwald density: $\Delta \propto a \rho_{\text{ped}}^{0.4} n_G^{0.3} \kappa^{-1.5}$ as shown in Fig.1. The result of the comparison of these scalings is that the new scaling and $\Delta R \propto (\rho_{\text{ped}})^{0.4}$ are well fitted and the fitting errors are almost the same as experimental error. However, further study is necessary for scaling parameters.

Fig.1
The new scaling of pedestal width for JT-60U and DIII-D. All data are in ELM-free H-mode. Full circles show JT-60U ion temperature pedestal width (65% of linear fit width), open squares show DIII-D electron temperature pedestal width (TANH fit width). The graph is plotted in log-log space.


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4 MHD Instabilities and High Energy Ions

4.1 Disappearance of Giant ELMs and Appearance of Minute Grassy ELMs in JT-60U High Triangularity Discharges


Abstract. In JT-60U H-mode plasmas, giant (type I) ELMs disappear and minute grassy ELMs appear when triangularity \( \delta \), edge safety factor \( q_{95} \) and \( \beta_p \) are high enough: Complete suppression of giant ELMs was observed at \( \delta \sim 0.45, q_{95} \sim 6 \) and \( \beta_p \sim 1.6 \). At higher \( \delta \) (0.54), giant ELMs can disappear at a lower \( q_{95} \) (~4.0). In the grassy ELMMy H-mode, edge temperature and pressure can be higher than those in giant ELMMy H-mode and a favorable confinement can be sustained without increase of impurity concentration. An edge stability analysis suggests that the edge plasma is accessing to the second stability regime of high \( n \) ballooning mode in the grassy ELMMy discharges [1].

1. Introduction

In JT-60U, the edge pressure gradient of H-mode is usually limited by appearance of giant ELMs. At low triangularity (\( \delta \leq 0.3 \); \( \delta \) is defined at the separatrix viz. \( \delta = \delta x \)), the edge pedestal parameters do not grow in the ELMMy phase. The critical \( \alpha \)-parameter just before the first giant ELM is almost constant over the wide range of the plasma current (\( I_p = 0.4 - 4.5 \text{MA} \)) and \( q_{95} \) [2,3]. The critical \( \alpha \)-parameter increases with internal inductance \( I_1 \) and close to the high \( n \) ideal ballooning mode limit evaluated by the time dependent analyses including edge bootstrap current. Frequency of the giant ELM (\( \sim 10 - 200 \text{Hz} \)) is roughly proportional to \( P_{\text{abs}} / (B^2l_1/(Rq_{95}^2)) \) [2]. The critical \( \alpha \)-parameter increases with triangularity [2,4]. Recently, in a long pulse ELMMy H-mode discharges, a slow growth of the pedestal structure in the giant ELMMy phase was observed when \( \delta \) is high (\( \sim 0.45 - 0.5 \)). The pedestal width (8-15 cm) and pedestal temperature (\( T_i \sim 1 - 2 \text{keV} \)) as well as the edge \( \alpha \)-parameter increase gradually with a long time constant of ~2 sec which is \( \sim 10 \times \tau_E \) [5]. In addition, at a high \( \delta > \sim 0.4 \), giant ELMs tend to disappear and, instead, minute grassy ELMs appear.

In fusion reactors, a high confinement should be sustained in H-mode discharges without harmful ELM heat pulses inducing erosion of divertor plates [6]. The grassy ELMMy H-mode [7] operation is one of the candidates to satisfy this condition. However, understanding of its operation condition is in an early stage [8]. One clue for understanding of change in ELM types is effect of plasma shape [7,9,10]. The purpose of this paper is to report the phenomenology of grassy ELMs observed in JT-60U high triangularity discharges.

![Figure 1. Comparison of a low-\( \delta \) giant ELMMy and a high-\( \delta \) grassy ELMMy discharges (\( I_p = 0.6 \text{MA}, B_t = 2 \text{T}, q_{95} = 8.2 \)).](image-url)
2. Grassy ELMy H-mode Regimes

In JT-60U, grassy ELMs are observed in high triangularity discharges. Figure 1 compares a low-δ giant ELMy discharge and a high-δ grassy ELMy discharge with similar peripheral electron density (r/a=0.7). At δ=0.12, edge T_e (r/a=0.88) saturates after appearance of giant ELMs. On the other hand, at δ=0.47, edge T_e continues to increase in the grassy ELMy phase. In addition, the trace of C VI brightness shows that carbon impurity concentration is not increasing in the grassy ELMy phase. In case of giant ELMs, the ELM frequency f_{ELM} is proportional to the absorbed heating power. Whereas, f_{ELM} of grassy ELMs is much higher at a given absorbed power. Another key parameter for appearance of grassy ELMs and disappearance of giant ELMs is the peripheral safety factor q_{95}. Figure 2 compares a low-q_{95} (3.5) and a high-q_{95} (5.9) discharges at I_p=1MA. In the low-q_{95} case, giant ELMs appear throughout the heating phase. In case of high-q_{95}, the grassy ELMy phase appear after t ~ 5 s. (The ELMs before t ~ 5s seem to be mixture of type III and giant ELMs.) In this case, edge T_e is also increasing. (The event at t=5.9 s is the β_p collapse caused by the internal low-n instability.) In the grassy ELMy phase, edge temperature can be higher than that in giant ELMy phase at a given Ip as shown in Fig.3.

At I_p=1MA, we conducted a scan of δ and q_{95} (B_T=2.6-3.6T). Figure 4 shows change of the ELM activity observed by divertor D_α signal (an integrated value over the divertor area including both inner and outer hit points). Giant ELMs are replaced by grassy ELMs i) as q_{95} increases (a -> b -> c) at fixed δ (0.45 - 0.49) and β_p (1.7-1.9), ii) as δ increases (e -> d -> c) at fixed q_{95} (~6) and β_p (1.8-1.9), and iii) as β_p increases (h -> c) at fixed q_{95} and δ. So far in JT-60U, complete suppression of giant ELM has been observed at δ>~0.45, q_{95}>~6 and β_p>1.6. At a higher δ=0.54, an almost pure grassy phase appeared even at a relatively low q_{95} = 4.0 (Fig. 4 (g)). Figure 5 shows the magnetic probe signals (dB/dt) for the discharges (a) - (e) treated in Fig.4. The expanded signal (a’) shows that a giant ELM accompanies a large magnetic burst. Whereas, as shown in (c’), grassy ELM bursts are much frequent (in this case ~ 1kHz) and

Figure 3. Ion temperature and electron density at the edge pedestal shoulder (I_p=1MA, δ=0.45 - 0.5)
amplitude is small. At low-q95 (3.5) with the pure giant ELMy phase, fELM increases linearly with the net heating power. However, at higher q95 where giant and grassy ELMs are mixed, fELM for giant ELMs becomes irregular.

3. Edge Parameters for Giant and Grassy ELMy H-modes

For detailed investigation of the H-mode edge at high δ, Figs. 6 compares the giant ELMy discharge E32358 and the grassy ELMy discharge E32511 (the discharges in Fig.2 and (a) and (c) in Figs.4 and 5). These discharges are so called high-βp ELMy H-mode characterized by both internal and edge transport barriers [11]. In E32358 (q95 =3.5), giant ELMs survive throughout the heating phase. The pedestal width (8-15cm) and pedestal temperature (T_e ~ 1-2keV) as well as the edge α-parameter increase gradually with a long time constant of ~2 sec which is ~10 x τ_E [4]. In E32511 (q95 =6.1), the grassy ELMy phase
Figure 6. (a) & (c): Profiles of a Giant ELMy high $\beta_p$ H-mode discharge at a low-$q_{95} (=3.5)$, $I_p=1$MA, $B_t=2.1$T and $\delta=0.45$. The stability analysis shows that the edge $\alpha$ parameter is critical for the high-$n$ ideal ballooning mode. (b) & (d): Profiles of a Grassly ELMy high $\beta_p$ H-mode discharge at a high-$q_{95} (=3.5)$, $I_p=1$MA, $B_t=3.6$T and $\delta=0.49$. The stability analysis shows access to the second stable regime of the high-$n$ ideal ballooning mode.

appear after $t \sim 8$ s. Then the pedestal width $\Delta_{\text{ped}}$ increases further and the internal inductance decreases simultaneously. Although the causality between the slow growth of the pedestal and the edge current profile evolution is not clear so far, there may be a feedback loop between them. In the fully developed phase of the edge pedestal, the stored energy sustained by the edge pedestal $W_{\text{ped}}$ is almost similar in both E32511 and E32358: $W_{\text{ped}}=0.53$MJ in E32511 at $t=8.9$s and $W_{\text{ped}}=0.50$MJ in E32358 at $t=9.2$s.

Stability analyses for the ideal high $n$ ballooning mode was conducted with a detailed evaluation of the edge bootstrap current by MSE measurement and EFIT [12]. The results are shown in Fig.6. In the high-$q_{95}$ discharge, the gap for the second regime access is open. On the other hand, in case of the low-$q_{95}$ discharge, the plasma edge is in the first regime and the pressure gradient hits the critical value for the ballooning mode. The access to the second stability regime is one of the candidates of disappearance of giant ELMs.

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4.2 Tearing Modes and Their Stabilization in Steady State High $\beta_p$ H-mode Discharges*

A. Isayama and Y. Kamada

1. Introduction

In the steady state high $\beta_p$ H-mode discharges, long sustainment of high beta plasma is interrupted by low-n tearing modes $^{1,2}$. Characteristics of the the tearing mode have been investigated since 1997 $^{3,4,5}$. In 1999, detailed investigation into density dependence of onset $\beta_N$ have been made with a fixed plasma configuration $^6$. Collisionality dependence have been also investigated. Tearing mode stabilization have also been performed using the electron cyclotron wave system installed in 1999.

2. Tearing modes in the steady state high $\beta_p$ H-mode discharges

Typical waveform of the steady state high $\beta_p$ H-mode discharge, where a 3/2 mode appears, is shown in Fig. 1. Typical plasma parameter is as follows: $I_p=1.5$ MA, $B_t=3.7$ T, $R=3.3$ m, $a=0.93$ m, $q_95=4.6$, triangularity $\delta=0.35$, $V_p=60$m$^3$. In this discharge, NB power is gradually decreased after the appearance of the 3/2 mode. As the NB power is decreased, amplitude of the 3/2 mode measured by saddle coils also decreases. Relation between the normalized beta and the square root of the amplitude of the 3/2 mode divided by the mode frequency ($\sim$ island width) is shown in Fig. 2. The value of $(\beta/\ell)^{0.5}$ increases with the normalized beta. According to the modified Rutherford equation $^7$,

$$\frac{d\beta}{dt} = \Delta' - 5.4\epsilon^2 (L_q/L_p)(1-\epsilon^2) \beta_p \frac{w}{w^2 + w_d^2} + \frac{a_1e^{1/2}(L_q/L_p)\beta_p w}{w^2 + w_d^2} - \frac{a_2g(\epsilon, \nu_0)(L_q/L_p)^2\beta_p \rho_{th}^2}{w^2 + w_d^2}$$

$a_1$ and $a_2$ are positive definite.

$$g(\epsilon, \nu_0) = \frac{\epsilon^{1/2}}{\omega_0 \omega_\nu} \quad \text{for} \quad \frac{\nu_0}{\epsilon} < 1$$

$$= 1 \quad \text{for} \quad \frac{\nu_0}{\epsilon} \omega_\nu \gg 1$$

only the third term of the right-hand side, which corresponds to the effect of the neoclassical helical bootstrap current, has a positive dependence on beta value if we assume $dw/dt=0$ and other terms are unchanged. This result suggests that one of the candidates of the tearing mode is the neoclassical tearing mode.

3. Dependence of onset $\beta_N$ on electron density and collisionality

From a series of the high $\beta_p$ H-mode discharges, it is found that onset $\beta_N$ increases with electron density $^5$. In order to investigate the density dependence more precisely, density scan was performed with a fixed plasma configuration shown in Fig. 3. Typical plasma parameter is as follows: $I_p=1.5$ MA, $B_t=3.7$ T, $R=3.2$ m, $a=0.8$ m, $q_95=4.0$, $\delta=0.1$, $V_p=55$m$^3$. Density dependence of the onset $\beta_N$ of a 3/2 mode is shown in Fig. 4, where the positive and linear dependence of onset $\beta_N$ is clearly shown. The curve, which lies in the lower beta region than the threshold, corresponds to the discharge where no mode was observed throughout the discharge. From this figure, it can be concluded that higher beta region is more unstable for the tearing mode.

According to the neoclassical tearing mode theory, the onset $\beta_N$ normalized by Lamor radius depends on collisionality. In Fig. 5, collisionality dependence of the onset $\beta_N$ is

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shown. Closed circles and closed squares correspond to the discharges for $B_I=3.5-3.7$ T and $B_T=2.1$ T, respectively. As can be seen in this figure, the onset $\beta_N$ weakly depends on collisionality. The triangle in Fig. 5 is the discharge where no tearing mode was observed. Waveform of this discharge is shown in Fig. 6. Typical plasma parameter at $t=6.7$ s is as follows: $I_p=1.5$ MA, $B_I=3.7$ T, $W_{diss}=5.7$ MJ, $R=3.2$ m, $a=0.8$ m, $q_{95}=4.0$, $\delta=0.1$, $V_p=55$m$^3$. At $t=6.12$ s, EC wave of 750 kW is injected to the center of the plasma. After the injection, central electron temperature increases about 1.5 keV even at high density ($n_e(0)=3.5\times10^{19}$m$^{-3}$ at $t=6.1$ s), and high beta ($\beta_p=2.0$, $\beta_N=2.4$) and high confinement plasma ($H^{TERP}=2.6$) is sustained for 1.4 sec until turn-off of the EC wave. The tearing mode may be stable in the low collisionality region, and destabilization of the tearing mode can be avoided by profile control.

4. Stabilization of a 3/2 tearing mode by off-axis ECH/ECCD

In 1999, tearing mode stabilization experiment was carried out using the EC injection system for the first time in JT-60U. Typical waveform of the discharge, where the $m/n=3/2$ mode is destabilized, is shown in Fig. 7. Typical plasma parameters are as follows: $I_p=1.5$ MA, $B_I=3.7$ T, $R=3.2$ m, $a=0.8$ m, $q_{95}=4.0$, $\delta=0.1$, $V_p=55$m$^3$. The resonance layer for the fundamental 110 GHz EC wave locates at $R=3.15$ m. At $t=5.8$ s, an $m/n=3/2$ mode is destabilized. The value of the normalized beta at the mode onset is about 1.5, and line averaged electron density measured by FIR interferometer with tangency radius of $r/a=0.4$ is $2.4\times10^{19}$m$^{-3}$. The frequency of the mode is about 6 kHz. Unmodulated electron cyclotron wave of about 750 kW was injected from $t=7.5$ s to 8.3 s. Because of rather high density and off-axis EC wave injection, electron temperature at the mode location does not change significantly. Thus, effects of ECCD are expected to be more important than those of ECH in this discharge. Profiles of amplitude and phase of electron temperature perturbations in this discharge measured by the heterodyne radiometer are shown in Fig. 8. The amplitude is normalized by DC component of the electron temperature perturbations. The amplitude becomes small, and the phase is inverted across $R=3.62$ m, which suggests that the center of the magnetic island locates at this position. The optimum injection angle was determined from the measurement with the heterodyne radiometer so that the EC wave could be deposited at the center of the island. After the EC wave injection, magnetic perturbations with $n=2$ measured by saddle coils were slightly decreased as shown in Fig. 7(d). Amplitude of electron temperature perturbations near the center of the magnetic island was also decreased as shown in Fig. 9. However, so far, disappearance of the magnetic perturbations and electron temperature perturbations, which means complete stabilization of the tearing mode, is not observed in this series of discharges. Analysis with the Fokker-Planck code shows that total driven current was about 20-30 kA, which may be insufficient to the complete stabilization.

5. Summary

Characteristics of tearing modes in the steady state high $\beta_p$ H-mode discharge was investigated. In 1999, density dependence of onset $\beta_N$ was investigated in a fixed configuration. It is found that the onset $\beta_N$ almost linearly increases with electron density. Collisionality dependence of the onset $\beta_N$ was also investigated. The onset $\beta_N$ normalized by
Lamor radius has a positive dependence on collisionality ($\sim n_e^{0.36}$). In some of the discharges with low collisionality, no tearing mode was observed even when beta value was high. This result suggests that the tearing mode is stable in the low collisionality region. Tearing mode stabilization experiment was performed. So far, complete stabilization has not been successful. The tearing mode stabilization experiment is planned after the increase of the injection power (up to 2.3 MW) in 2000.

References

![Fig.1: Typical waveform of the steady state high $\beta$ H-mode discharge.](image1)

![Fig.2: Dependence of $(dB/dt)^{0.5}$ on $\beta_N$.](image2)

![Fig.3: Typical plasma configuration for investigation of onset $\beta_N$ and tearing mode stabilization experiment.](image3)

![Fig.4: Density dependence of onset $\beta_N$ of an m/n=3/2 mode.](image4)
Fig. 5: Collisionality dependence of onset $\beta_n$ of an n/n = 3/2 mode.

Fig. 6: Waveform of the high $\beta_n$ H-mode discharge with EC wave injection.

Fig. 7: Waveform of a tearing mode stabilization experiment.

Fig. 8: (a) Plasma shape and ray of EC wave. (b) Amplitude and Phase of electron temperature perturbation near the magnetic island.

Fig. 9: Time evolution of (a) plasma current, NB power and EC wave power, (b) magnetic perturbation with n = 2, (c)-(f) amplitude of electron temperature perturbation.
4.3 Effect of ECH/ECCD on Sawtooth Oscillations*

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1. Introduction

Sawtooth oscillations are commonly observed in many tokamaks. In JT-60U, it has been reported that sawtooth period can be changed by ion cyclotron resonance heating and its combination with NB injection or lower hybrid current drive 1). However, it has not been investigated the effects of EC wave on the JT-60U plasma. After the installation of the EC wave injection system, effects of the EC wave on sawtooth oscillations have been investigated. First, dependence of sawtooth period on injection angle (i.e. absorbed position) is investigated. Next, effects of ECH and ECCD are investigated by changing the direction of plasma current and toroidal field.

2. Dependence of sawtooth period on EC wave injection angle

Plasma configuration is shown in Fig. 1, and plasma parameters just before the EC wave injection are as follows: $I_p=1.4\text{ MA}, B_t=3.7\text{ T}, R=3.15\text{ m}, a=0.81\text{ m}, q_{95}=4.5, V_p=55\text{ m}^3$. Line averaged electron density with tangency radius of $r/a=0.4$ is $0.9\times10^{19}\text{ m}^3$. The resonance layer locates at $R=3.15\text{ m}$. Toroidal direction of the EC wave corresponds to co-ECCD direction. In this experiment, EC wave was injected to a Ohmically heated plasma. Rays of the EC wave for each injection angle are shown in Fig. 1. When the angle is 48 degree, the ray of the EC wave passes the plasma center. As the angle decreases, the injected EC wave is deposited at off-axis region.

In Fig. 2, time evolution of central electron temperature measured by the grating polychromator is shown. While sawtooth appears shortly after the EC wave injection for 48 degree injection and 46 degree injection, no sawtooth is observed for 270 ms after the EC wave injection for 44 degree injection. The deposition profile estimated from the increment of electron temperature shows that the injected wave is deposited near the inversion radius for the 44 degree injection. On the other hand, for the 48 degree injection and the 46 degree injection, the EC wave is deposited inside the inversion radius. As the angle is further decreased, the sawtooth-free period decreases again. The relation between the maximum sawtooth-free period and the injection angle is shown in Fig. 3. When the angle is 48 degree, 46 degree and 42 degree, the maximum sawtooth-free period is about 60-70 ms, which is about twice longer than that in the Ohmic heating phase. On the other hand, when the angle is 44 degree, the sawtooth-free period is extended to 270 ms, which is 9-10 times longer than that in the Ohmic heating phase and 4 times longer than that for the 48 degree, 46 degree and 42 degree injections.

Dependence of the sawtooth period on central electron temperature just before the sawtooth crash is shown in Fig. 4. It is theoretically shown that the period of sawtooth oscillation is proportional to $T_e(0)^{1.5}$. In Fig. 4, sawtooth period calculated from this scaling is also shown. Coefficient for the scaling is determined from the sawtooth period at the Ohmic heating phase. For the 48 degree, 46 degree and 42 degree injections, sawtooth period in experiment is comparable to the theoretically predicted one, which suggests that the increase in the sawtooth period is attributed to the increase in electron temperature. On the other hand, the maximum sawtooth-free period obtained for the 43 degree, 44 degree and 45 degree injections is much longer than the predicted value ($\approx 50\text{ ms}$). Thus, the long sawtooth-free period is not attributed to the increase in electron temperature. It is thought that it was caused by change of pressure profile and/or current profile. As shown in Fig. 4, the sawtooth-

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free period suddenly decreases after the first crash. Although the sawtooth period after the
first crash is comparable to the calculated one, it still has dependence on the injection angle.
When the EC wave is injected to the central region such as 48 degree injection, the period is
shorter than the calculated one. On the other hand, when the EC wave is injected to the
peripheral region such as 42 degree injection, the period is longer than the calculated one.
This result suggests the controllability of the period of the sawtooth oscillation, which is
useful to the control of heat and particles at the central region.

3. Sawtooth period for co-ECCD and counter-ECCD
Since toroidal injection angle is fixed at about 15° in the EC wave injection system,
EC wave is injected with finite N//. In order to distinguish between the effect of ECH and that
of ECCD, direction of I_p and B_r were changed. In Fig. 5, injection angle dependence of
maximum sawtooth period for co- and counter-ECCD case is shown. Plasma shape and rays
of EC wave are shown in Fig. 6. Plasma configuration is the same as that in Fig. 1, except for
plasma species. In this case, EC wave was injected to hydrogen plasma. Both for co-ECCD
and counter-ECCD cases, maximum sawtooth oscillation period increases when the injection
angle is 44 degree and 51 degree, which corresponds to the deposition near the q=1 surface.
On the other hand, maximum sawtooth-free period for co-ECCD case is longer than that for
counter-ECCD case, which suggests that both ECH and ECCD contribute to extend the period
of sawtooth oscillation.
As described above, extension of the sawtooth-free period disappears after the first
crash. Next, sawtooth period after the saturation of electron temperature was investigated.
Time evolution of sawtooth period for 44.5 degree injection, which corresponds to the
deposition near the q=1 surface, is shown in Fig. 7(a). Even after the first crash, sawtooth
period for co-ECCD case is longer than that for counter-ECCD case, which suggests that both
ECH and ECCD contribute to extend the sawtooth period for off-axis injection. On the other
hand, on-axis counter-ECCD has stabilizing effect if driven current is large enough to change
current profile. Time evolution of sawtooth oscillation for 48 degree injection is shown in Fig.
7(b). So far, no clear difference is observed between co- and counter-injection cases.
From this series of discharges, controllability of the sawtooth period can be discussed.
Time evolution of sawtooth period is shown in Fig. 8. The sawtooth period is normalized by
that obtained from the T_e(0)^1.5 scaling. As shown in Fig. 8, the normalized sawtooth period is
smaller than unity for (a) center injection and (b) counter injection near the q=1 surface,
which shows the destabilizing effect. On the other hand, the normalized sawtooth period is
larger than unity for co-injection near the q=1 surface, which shows stabilizing effect. These
results indicate that sawtooth period can be controlled by ECH/ECCD.

4. Summary
After the installation of the EC wave injection system in 1999, effects of ECH/ECCD
on sawtooth oscillations was investigated. It is found that EC wave injection near the q=1
surface drastically extends the sawtooth period. It is also found that both ECH and ECCD
contribute to the sawtooth stabilization for the injection near the q=1 surface. Sawtooth
stabilization by counter-ECCD near the magnetic axis has not been observed. From this series
of discharges, it is found that there is a possibility of controllability of sawtooth period by
ECH/ECCD.
References

Fig. 1: Plasma shape and rays of EC wave.

Fig. 2: Time evolution of electron temperature during EC wave injection.

Fig. 3: Dependence of maximum sawtooth oscillation on EC wave injection angle.

Fig. 4: Relationship between sawtooth oscillation and central electron temperature.
Fig. 5: Dependence of normalized sawtooth oscillation on EC wave injection.

Fig. 6: Plasma configuration and rays of EC wave.

Fig. 7: Time evolution of sawtooth period.

Fig. 8: Time evolution of sawtooth period for various injection angle and toroidal direction.
4.4 Wall Stabilizing Effect and MHD Activity in Large Volume H-mode Plasmas

Y. Sakamoto, Y. Kamada, S. Takeji

1. Introduction

For the steady state operation of a tokamak reactor with a high bootstrap current fraction, stabilization of the low-n pressure-driven kink mode by a nearby conducting wall is essential to reach a sufficient plasma beta value. However, for a wall with finite conductivity, the stabilizing image currents in the wall decay, then unstable resistive wall mode (RWM) grows with an exponential growth time of the order of the wall resistive decay time $\tau_w$. On the other hand, some theoretical work suggests that the RWM can be stabilized for times much longer than $\tau_w$ if the plasma is rotating sufficiently fast relative to the wall. In this section, we describe the first experimental results of wall stabilizing effect and MHD activity in large volume H-mode plasmas close to the wall.

2. Experimental Results

Figure 1 shows the typical waveform and the plasma configuration for this experiment. Here, the toroidal magnetic field is 1.87T, the plasma current is 1.1MA, the triangularity is 0.37, plasma volume is 80m$^3$ and the wall located at 1.1~1.3 times the minor radius. Neutral beam (NB) of about 22.4MW was injected at $t=3.1$sec into the low target density $n=4.4\times10^{18}$m$^{-3}$ for the effective central heating. This plasma has no sawtooth oscillation due to early NB injection, which means safety factor at the magnetic axis was kept above unity. The value of $\beta_n$ was increased quickly after the NB injection. Then $\beta_n=3.42$, $\beta_p=1.84$, $W_{du}=3.3MJ$, $q_{95}=3.4$ and $H_{89}=2.6$ was achieved at $t=3.6$sec, where the giant ELM occurred, after 0.1sec $\beta_p$ collapse was observed. Here, the fast ion loss due to ripple field is not taken account for the estimation of $H_{89}$. This plasma configuration has large ripple field. The loss due to ripple field is estimated to be 58.4% using OFMC code. If this loss is subtracted from the heating power, $H_{89}$ will become 6.1, where $P_{abs}=8.96MW$, $dW/dt=6.51MW$. Figure 2 shows the electron density and electron temperature profiles. These profiles have both edge and core transport barriers, respectively. Unfortunately, ion temperature profile was not obtained in this discharge. There is possibility that this discharge exceeds the ideal no-wall stability limit. However it is not clear because of the lack of ion temperature profile. In Sec. 3, the ideal stability analysis will be discussed using similar discharge. Figure 3 shows MHD activity before the collapse, where the resistive mode ($m/n=3/1$) was excited with decreasing frequency.

3. MHD Stability Analysis

In order to investigate the wall stabilizing effect, MHD stability was analyzed by ERATO-J code for similar discharge which has $\beta_n=3.29$, $l_i=0.85$ and $q_{95}=3.46$ at $B_T=1.86$ and $I_p=1.1MA$. Figure 4 shows pressure profile in each component, which are thermal, fast ion
and total pressure profiles. Pressure peakedness is 4.2. This is very high value compared to previous database, which was obtained for plasmas far from the wall. Previous database indicates $\beta_p$ collapse occurred below $\beta_n=2.0$ when the pressure peakedness was around 4.2 [1].

Figure 5 shows the radial mode structure for n=1 analyzed with no wall, where safety factor on the magnetic axis was assumed 1.2. From the ideal stability analysis, this plasma was unstable with no wall. On the other hand, the plasma was stable when the wall located at 1.2 times the minor radius. These results indicate the possibility that the discharge exceeded the ideal no-wall stability limit.

Reference
Fig. 1. Typical waveform and the plasma configuration

Fig. 2. Electron density and electron temperature profiles
Fig. 3. MHD activity before the collapse, where \( m/n = 3/1 \) mode was excited with decreasing frequency.

Fig. 4. Total, thermal and fast ion pressure profiles.

Fig. 5. Radial mode structure for \( n=1 \) analyzed with no wall.
4.5 Wall Stabilization of Reversed Shear Discharges in JT-60U


1. Introduction

Demonstration of high beta tokamak operation is important to prove a feasibility of tokamaks as an economical fusion reactor and to realize a steady state operation with high bootstrap current fraction, $f_{BS}$. In the case of ITER-FEAT, for instance, challenging plasma parameters such as $H_{N} = 1.5$ and normalized beta $\beta_{N} = 3.2 \sim 3.5$ are supposed for the low current steady-state operation. Since reversed magnetic shear configuration can be consistent with large $f_{BS}$ [1], reversed shear discharges have inherent potential for non-inductive steady-state operation without large current drive power. On the other hand, ideal magnetohydrodynamic, MHD, stability limit with the wall at infinity, $\beta_{N}^{\text{wall}}$, of reversed shear configuration is relatively lower than that of the conventional configurations owing to the broader current profile, i.e. lower plasma internal inductance, $\ell_{i}$ of reversed shear discharges. Wall stabilization of ideal low $n$ (n: toroidal mode number) kink modes is, therefore, a key issue to realize a high beta discharge as $\beta_{N} > \beta_{N}^{\text{wall}}$ with large $f_{BS}$.

Both of theoretical and experimental studies of wall stabilization and resistive wall modes which are destabilized at $\beta_{N} > \beta_{N}^{\text{wall}}$ are recently being intensively carried out [2].

2. Improved stability by the JT-60U wall

Typical plasma shapes of high performance discharges in JT-60U are shown in Fig.1. Experiments for improved confinement have been usually carried out by employing elongated plasma shapes ($\kappa \lesssim 1.8$) such that the plasma surface is distant from the outer wall to reduce the effect of the toroidal ripple ((a) and (b) in Fig.1). Plasma volume, $V_{p}$, of such shapes are $V_{p} < 65m^{3}$ and a ratio of the wall radius, $d$, to the plasma minor radius, $a$, is $d/a \geq 1.5$ which seemed to be too distant to see a stabilizing effect of the JT-60U wall on pressure driven external kink modes. We employed plasma shapes with larger plasma volume, $V_{p} \geq 70m^{3}$ and with smaller distance between the plasma surface and the outer wall ($d/a \leq 1.3$) to study the wall stabilization on JT-60U ((c) in Fig.1).

Plasma performance of reversed shear discharges are often limited by disruption or major collapse in the wide range of $\beta_{N}$. Figure 2 shows the values of $\beta_{N}$ as a function of the edge safety factor $q^{*}$ at the time of disruptions or major collapses in reversed shear discharges with L-mode and H-mode edges. In the case of plasma shapes with $d/a > 1.5$ (i.e., (a) and (b) in Fig.1), the achievable $\beta_{N}$ is lower than 2.3 in the L-mode edge discharges, and the achievable $\beta_{N}$ was slightly improved up to 2.6 by getting the H-mode edge. One of the remarkable results is that the achievable $\beta_{N}$ is improved further by employing a plasma shape with $d/a < 1.3$ ((c) in Fig.1) up to $\beta_{N} \sim 2.8$. Since the data points denoted as "H-mode edge" are a selected data set of the highest $\beta_{N}$ discharges with the H-mode edge near the same $q^{*}$, significant improvement of the achievable $\beta_{N}$ is considered to be caused by wall stabilization by employing the plasma shape such that the plasma surface is close enough to the wall. In many of these discharges with $d/a < 1.3$, the achieved $\beta_{N}$ exceeded an empirical beta limit scaling that $\beta_{N} \sim 4\ell_{i}$. 
The other remarkable result is that no disruption nor major collapse occurred in lower $\beta_N$ region in discharges with $d/a < 1.3$. One of the reasons is the improved operation scenario of reversed shear discharges. Normalized beta $\beta_N$ was kept to be lower than unity during the plasma current ramp up phase. In such a current ramp operation, ideal low $n$ kink instabilities associated with the plasma surface current could be suppressed [3]. The other plausible reason is that resistive instabilities (tearing modes) which can be a causal instability for the low $\beta_N$ disruption or minor collapse are stabilized by the wall. Actually, a stability parameter of tearing modes $\Delta'$ can be changed to be negative (stable) when a conducting wall is close enough to the plasmas surface even though $\Delta'$ is positive (destabilizing) with a distant conducting wall [4]. It is also known that tearing modes can be stabilized by resistive walls, provided the plasma rotation frequency exceeds $\tau_w^{-1}$ ($\tau_w$ is the magnetic field penetration time into the wall) and a characteristic tearing growth rate [5,6]. An estimated $\tau_w$ of the JT-60U wall made of stainless steal with 10 mm thickness is about 10 millisecond and a characteristic tearing growth time may be order of millisecond or longer. A typical toroidal rotation frequency of the JT-60U reversed shear plasmas is several kilo hertz. The plasma rotation frequency seems to be large enough for stabilization of tearing modes by the JT-60U wall.

Fig.1. Typical plasma shapes of reversed shear discharges in JT-60U. (a): High elongation ($\kappa \sim 1.8$) for stability at high plasma current, (b): high triangularity ($\delta \geq 0.3$) for improved edge stability and (c): large volume ($V_p \sim 75m^3$) with $d/a \sim 1.3$ for wall stabilization.

Fig.2. Normalized beta $\beta_N$ versus edge safety factor $q^*$ at the time of disruptions or major collapses in several type of JT-60U reversed shear discharges. Open circles are data of the L-mode edge discharges with the plasma shape (a) in Fig.1. Closed diamonds are ones of the highest $\beta_N$ data obtained in discharges with the H-mode edge with the plasma shape (a) or (b) in Fig.1. Squares are ones obtained in discharges with $d/a \leq 1.3$ ((c) in Fig.1)
3. Observation of resistive wall modes

Improvement of stability owing to the stabilizing effect of the JT-60U wall was demonstrated as referred to above. Here, MHD instabilities observed in such wall stabilized discharges are described. Figure 3 shows waveforms of a reversed shear discharge in which the achieved $\beta_N$ exceeded the calculated stability limit of the $n = 1$ ideal kink mode with the wall at infinity. The plasma volume was 77 m$^3$ and the ratio of the wall radius to the plasma minor radius was $d/a \lesssim 1.3$. The calculated no-wall ideal stability limit is $\beta_{N{\text{no-wall}}}^\infty \sim 2.2$ in the similar discharge with $\ell_1 \sim 0.7$. The normalized beta $\beta_N$ increased gradually in time and exceeded $\beta_{N{\text{no-wall}}}^\infty$ at $t \sim 6$ second. The discharge terminated in disruption at $\beta_N = 2.6$. A set of eight saddle sensors located toroidally in the midplane inside the vacuum vessel detected radial magnetic perturbations with $n = 1$ right before the disruption. Growth rate of the $n = 1$ mode is 120 s$^{-1}$ which is corresponding to $\tau_w^{-1}$. No other mode numbers such as $n = 2$ and $n = 3$ were observed at the same time. Plasma rotation frequency in the toroidal direction, $f_{\text{tor}}$, was about 4 kHz in the counter-direction to the plasma current at the radial region corresponding to the safety factor $q = 3$ and $q = 4$ rational surfaces and didn’t change significantly until the beginning of growth of the $n = 1$ mode. Since time resolution of the plasma rotation measurement by means of the charge exchange recombination spectroscopy is 16.7 ms, we couldn’t observe the change of $f_{\text{tor}}$ during the growth of the $n = 1$ mode right before the disruption.

![Figure 3: Waveforms of (a) $\beta_N$ and plasma rotation frequency in the toroidal direction $f_{\text{tor}}$ at $q \sim 3$ and $q \sim 4$, (b) time derivative of $n = 1$ radial magnetic perturbations. Calculated ideal stability limit ($\beta_N \sim 2.2$) with the wall at infinity is denoted as a broken line in (a).](image)

![Figure 4: Contour plot of time derivative of the perturbed radial magnetic field measured by the toroidal array of saddle loop sensors inside the vessel.](image)

Figure 4 shows time evolution of the perturbed radial magnetic field. The contour plot of the perturbed radial magnetic field reveals clearly an $n = 1$ mode rotating with the frequency $f \sim 20$Hz $\sim 1/(2\pi \tau_w)$ in the counter-direction of the plasma current (the same direction with the plasma rotation) which corresponds to the direction of the ion diamagnetic drift. The feature of the $n = 1$ mode appeared in the condition that $\beta_N \gtrsim \beta_{N{\text{no-wall}}}^\infty$ are identical with that of resistive wall modes predicted theoretically [7] and reported experimentally on DIII-D [2]. Thus, we can say that resistive wall modes are also observed on JT-60U.
One of the differences of resistive wall modes between JT-60U and DIII-D is that no clear reduction of the plasma rotation is observed near the mode rational surface \((q = \text{integer})\) on JT-60U. In the case of DIII-D, significant reduction of the plasma rotation was observed when \(\beta_N > \beta_n^{\text{sec-wall}}\). The experimental evidence of the relation between the amplitude of the resistive wall mode and reduction of the rotation frequency on DIII-D is considered to be a supporting evidence of a theory in which torque balance is taken into account to determine plasma rotation self-consistently [8]. Plausible reasons of the difference between JT-60U and DIII-D are (i) time resolution of the plasma rotation measurements of \(\sim 17\) ms is not enough to detect the change during excitation of the resistive wall, (ii) the plasma rotation of \(\sim 4\) kHz with the estimated perturbed radial magnetic field of several gauss might be already in the lower unstable branch in Gimblett’s theory. Detailed analysis of the relation between the plasma rotation and the perturbed radial magnetic field by resistive wall modes will be done to compare the experimental result with the theoretical model.

4. Conclusions

Stabilizing effects of the JT-60U wall on pressure driven low \(n\) kink modes was confirmed and reversed shear discharges with \(\beta_N > \beta_n^{\text{sec-wall}}\) were obtained by employing plasma shapes with \(d/a \leq 1.3\). MHD perturbations that are attributed to the resistive wall mode are observed followed by disruption in the wall-stabilized high beta discharges.

References

4.6 Stability of Reversed Shear Discharges in JT-60U[1]


Magnetohydrodynamic (MHD) stability has been studied in discharges with negative central magnetic shear, i.e., reversed shear, configuration in JT-60U. The disruptive upper limit of achievable normalized beta, $\beta_N$, is $\beta_N \lesssim 2$ at $q_{\text{min}} \sim 2$ ($q_{\text{min}}$: the minimum safety factor) in discharges with the L-mode edge and is close to the stability limit against an ideal $n = 1$ ($n$: toroidal mode number) kink mode with the free boundary condition [2] (Fig.1). On the other hand, precursors with the resistive time scale (typically $\gamma^{-1} \geq 0.3 \text{ms}$) often appear right before those major collapses. Numerical analysis revealed that some of these major collapses occur in the stable beta region against ideal modes (A and B in Fig.1). Thus, we confirmed that resistive instabilities give rise to major collapses at the lower $\beta_N$ regime than the ideal stability limit. Linear stability analysis with a two-dimensional marginal stability analysis code, MARG2D [3], showed that a stability parameter of tearing mode, $\Delta'$, is positive (destabilizing) at the outer $q = 3$ surface and a stability parameter of double tearing modes, $\Delta(0)$, is negative (destabilizing) at the $q = 3$ surface in the experimental situation (Fig. 2). This numerical result is not in contradiction to the experimental one. Moreover, we found that major collapses often occur when the safety factor at the plasma surface is close to integer values, suggesting that an external kink mode is related to the causal mechanism of major collapses. The numerical analysis revealed that a stable external kink mode ($m=5$ in the case of Fig. 2) plays key role to destabilize tearing modes in the reversed shear configuration.

![Image](image_url)

**Fig.1.** Occurring points of major collapse (or disruption) in the $\beta_N$-$q_{\text{min}}$ plane. Here, closed circles are collapses which had no clear precursor or had precursor with $\gamma^{-1} < 0.3 \text{ ms}$ and open circles are those which accompany the precursor with $\gamma^{-1} > 0.3 \text{ ms}$. A solid line shows the stability boundary against an ideal $n = 1$ kink mode calculated for a typical reversed shear equilibrium in the free boundary condition [2].

![Image](image_url)

**Fig.2.** Eigenfunctions of $m = 2 \sim 5$ ($n = 1$) modes ($m$: poloidal mode number) calculated by using MARG2D with the free boundary condition ($b/a = 1.8$) together with the $q$ profile in the equilibrium. Here, the equilibrium has $\beta_J = 0.85$, $q_{\text{min}} \sim 2.1$ and the edge safety factor, $q_{\text{edge}} = 4.87$. The calculated eigenvalue is positive which means that the plasma is stable against the ideal $n = 1$ kink mode in this case.

**References**

4.7 Fast particle experiments in JT-60U\textsuperscript{1}

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In this paper, results of fast particle experiments that were performed in JT-60U are presented. Fast particles were created with Ion Cyclotron Range of Frequency (ICRF) and Negative ion based Neutral Beam Injection (N-NBI) heating.

In ICRF heated plasmas, Toroidicity and Ellipticity induced Alfvén Eigenmodes (TAEs and EAEs) are usually observed before and after giant sawtooth crashes. We have used these TAEs and EAEs to study the behavior of the \( q \)-profile in the plasma center. This is possible because these AEs are residing at well determined \( q \)-surfaces in the plasma core. It was found that just before the sawtooth crash the central value of the \( q \)-profile, \( q_0 \), reached values between 0.8 and 0.9 and at the crash \( q_0 \) did not usually relax back to unity. Immediately after the crash values from below to above one were found. For sawteeth where \( q_0 \) relaxes back to above one a typical change in \( q_0 \) was found between 0.15 and 0.2. The observations that \( q_0 \leq 1 \) after the crash agree well with the rigid shift and quasi-interchange flow model of Kolesnichenko et al. [2] but the observations in which \( q_0 \) relaxes back to values larger than one immediately after the sawtooth crash are more difficult to explain with the current sawtooth models.

In ICRF heated plasmas, first results were obtained to measure the radial TAE structure with X-mode reflectometry [3]. These measurements are in qualitative agreement with NOVA-K simulations [4].

The N-NBI system is a good tool to create large populations of fast passing particles. When N-NBI beams were injected into a sawtoothing plasma, the sawtooth period increased by a factor of 2.5. This increase may be attributed to the change in the current and pressure profiles inside the \( q = 1 \) radius and could also be related to the pressure of the fast particles.

In N-NBI experiments, designed to excite AEs, a new class of bursting, chirping and stationary frequency modes was reported. Most but not all of these modes scale with with the Alfvén velocity. One group of modes that does not follow the Alfvén scaling starts well inside the lower Alfvén continuum but they chirp up in about 200 ms to the TAE gap. These modes can be associated with Resonant-TAEs [5]. Another group of modes that does not follow the alfvén scaling are short bursting modes (duration up to about 20 ms) that appear in the lower Alfvén continuum. Both the bursting modes that scale with the Alfvén velocity and those that appear in the Alfvén continuum chirp on a 2 to 20 ms time scale. Frequency up- and down-chirping has been observed as well as simultaneous up- and down-chirps. A thorough theoretical investigation has to be made to understand the chirping and bursting behavior of these modes.

5. N.N. Gorelenkov, et al. Stability properties of Toroidal Alfvén modes driven by fast particles. Accepted for publication in Nucl. Fusion.
4.8 Excited condition of instability with a frequency sweep in a frequency regime of Alfvén Eigenmode in JT-60U

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1. Introduction
Instability in a frequency regime of Alfvén Eigenmodes (AEs) can cause anomalous transport of α particles in burning plasmas with high α particle pressure [1]. Investigation of instability with a frequency sweep in a frequency regime of AEs is important, because some instabilities with a frequency sweep induce enhanced transport of energetic ions [2-4] and theory for explaining the frequency sweep is being progressing [5]. Here we show the recent results of investigation of excited condition of instability with a frequency sweep in an frequency regime of AE.

2. Instability with a frequency sweep in NNB injected plasma
Figure 1 shows temporal evolutions of plasma parameters and a frequency spectrum of magnetic fluctuations. A toroidal magnetic field on the axis (B_0) is 1.2 T and an energy of NNB (E_{NNB}) is 360 keV and a power of NNB (P_{NNB}) is ~ 3.2 MW. The discharge gas is D_2 and the beam was D_0. A preheating by a Positive-ion-based Neutral Beam (PNB) during the current ramp-up phase is performed in order to have a safety factor profile, q-profile, with low central shear. In Fig. 1 (c), we can see a mode whose frequency changes slowly from ~ 20 kHz at ~ 3.6 s, ~ 100 ms after a start of NNB injection, to ~ 60 kHz at 3.8 s. Here a slowing-down-time, \tau_n, of NNB is ~ 300 ms in this discharge. The frequency of ~ 60 kHz is a Toroidicity induced AE (TAE) frequency in this discharge and the frequency of ~ 25 kHz is inside the Alfvén continuum. Here we call this mode slow-frequency-sweep mode (Slow FS mode). A toroidal mode number of this Slow FS mode is n = 1. We can also see other modes around ~ 60 kHz which have a burst-like behaviour and its frequency changes around the TAE frequency by 15 – 20 kHz in 1 - 10 ms. Here we call this mode fast-frequency-sweep mode (Fast FS mode). A toroidal mode numbers of the Fast FS modes are 1 and 2.

3. <β₀> scan and \nu_{β₀}/\nu_A scan
We performed a B_0 scan and a power scan of NNB, which change β₀, in order to understand the Slow FS mode and Fast FS mode. B_0 scan was performed at 1.2, 2.1, and 3.5 T with E_{NNB} = 350 - 360 keV and P_{NNB} ~ 3.2 - 3.4 MW. Plasma current (I_p) was chosen to keep a similar surface safety factor, q_s ~ 5 at the I_p flat top. The Fast FS mode and the mode similar to Slow FS mode were observed at 2.1 T. A continuous TAE was also
observed at 2.1 T. However the Slow FS mode and the Fast FS mode were not observed and only a continuous TAE was observed at 3.5 T. A volume-averaged $\beta_h$ ($<\beta_h>$) is estimated $\sim 0.1$ % at 2.1 T and $\sim 0.07$ % at 3.5 T. Thus these result might suggest that Slow FS modes and Fast FS modes require relatively large $<\beta_h>$, and the threshold to destabilize these FS modes exist between $<\beta_h>$ $\sim$ 0.1 and 0.07 % in the weak shear plasmas studied. Another result from this $B_{T0}$ scan is that we observed TAE at $B_{T0} = 3.5$ T even in NNB plasma. This means TAE driven by NNB ions can be also destabilized with $<\beta_h>$ $\sim$ 0.05 % similar to TAE induced by ICRF in JT-60U, even though the mechanism of producing parrallel fast ions is different.

A $P_{NNB}$ scan was performed at $\sim 3.2 - 3.4$ MW and $\sim 2.5 - 2.8$ MW with $E_{NNB} = 360$ keV, $B_{T0} = 1.2$ and 2.1 T. The timing for clear Fast FS modes to appear was delayed for the 2.5 MW case compared with the 3.2 MW case. On the other hand, a continuous TAE was clearly observed for the 2.5 MW case and the continuous TAE changed into a Fast FS mode in the later phase. The amplitude of the Fast FS mode is 2 - 3 times larger than that of the preceding continuous TAE. It is considered that $\beta_h$ increases with time, thus this result might suggest that the TAE become Fast FS mode when $\beta_h$ reaches some threshold.

The above motioned experiments were performed in $v_{br}/v_\Lambda \sim 0.6$-0.9. We also performed $v_{br}/v_\Lambda$ scan by increasing electron density with hydrogen beam and $v_{br}/v_\Lambda$ has been increased up to $v_{br}/v_\Lambda = 1.55$, which is similar to expected $v_{br}/v_\Lambda$ in ITER, as shown in Fig. 2 ($v_{br}$; the beam injection velocity in the toroidal direction, $v_\Lambda$; the Alfvén velocity, $v_\Lambda$; the birth velocity of $\alpha$ particles). Slow FS mode and Fast FS modes were destabilized for the condition of $v_{br}/v_\Lambda = 1.2 - 1.4$ and $<\beta_h>$-0.15% included in the domain of $\alpha$ particle condition in ITER.

4. Summary

A $B_{T0}$ scan and a power scan of NNB are performed in order to understand Slow FS modes and Fast FS modes. We had a result that the threshold to destabilize these FS modes exist between $<\beta_h> \sim 0.07$ and 0.1 % from a $B_{T0}$ scan. Slow FS modes and Fast FS modes could be caused by larger $\beta_h$ or larger gradient of $\beta_h$ than a continuous TAE.

We also observed these instabilities with a frequency sweep were destabilized for the condition of $v_{br}/v_\Lambda = 1.2 - 1.4$ and $<\beta_h>$-0.15% included in the domain of $\alpha$ particle condition in ITER.

Reference

Figure 1: (a) shows temporal evolutions of plasma current, Ip, power of NNB, PNNB, and power of PNB, PPNB. (b) shows temporal evolutions of a line-averaged electron density, $n_e$, and neutron emission rate, Sn. (c) shows the temporal evolutions of the frequency spectrum of magnetic fluctuations.

Figure 2: A domain of experiment of instability with a frequency sweep and AEs by N-NB. Open circles show instability with a frequency sweep. A domain of $\alpha$ particles in ITER, a excitation domain of "Chirping mode", which is an instability with a frequency sweep in 1 - 10 ms, in DIII-D, and a domain of DT plasmas with $\alpha$ particles in TFTR(1),(2) and JET(3) are also shown. TAE was observed in (1), not observed in (2) and (3).
4.9 Study of Alfvén Modes Driven by NNBI Ions in JT-60U

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Modes in the Alfvén frequency range observed in JT-60U during the NNBI heating at energies $E_{\text{th}}=360\text{keV}$ in discharge \#32359\textsuperscript{1} were analyzed using the kinetic nonperturbative code HINST, which is able to resolve new resonant branches of the toroidal Alfvén modes called resonant TAE (RTAE). The common feature for these modes is the frequency evolution or frequency chirp, which, however, may be different in terms of physical mechanisms as can be seen from very different time scales of the frequency chirping. Frequency chirping is a common phenomenon for experiments with strong fast particle pressure. The properties of Alfvén modes in the presence of a strong drive from fast particles was subject of this study. Theory has predicted that in the presence of a strong drive, TAEs will be strongly modified to new types of mode, the Resonant TAE \textsuperscript{2}. We showed that RTAE properties of linear response to the changing plasma and/or fast particle parameters through frequency change can be used to explain both gradual and rapid frequency chirp (bursting modes \textsuperscript{1}) observed in analyzed experiments.

Ideal MHD code NOVA-K is limited by applicability of its perturbative approximations to the JT-60U NNBI experiments with strong drive from beam ions, and, thus, we use the nonperturbative fully kinetic code HINST \textsuperscript{3}, which stands for high-$n$ stability code. This code is able to reproduce RTAE branches with arbitrary drive. It includes bulk plasma and fast particle Finite Larmor Radius (FLR) effects. Radiative damping supported by trapped electron collisional effects and ion Landau damping are also included. Even though HINST is able to reproduce robust solutions with high toroidal $n$ numbers that have radially localized mode structures, it can be used for medium-$n$ to low-$n$ modes in the local version of the HINST without resolving two-dimensional (2D) structure.

Different scenarios and physical mechanisms are proposed to explain experimental observations of AE frequency chirp. Our approach is purely linear, however we can predict the response of the plasma to fast particles and with different fast particle distributions without discussing how a given distribution is formed. Such an approach includes the right physical mechanisms responsible for the frequency chirping in experiments. Note that nonlinear effects should be important and need to be considered, but it was not within the

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scope of this study. Nonlinear evolution of a model bump on tail instability is able to demonstrate rapid frequency chirping as was shown in Ref. 4. However, the full picture may be obtained only by combining the nonlinear model with proper nonperturbative plasma dispersion analysis.

We found two different types of modes when the gradually chirping mode on a time scale of a few hundred milliseconds was initially observed. Lower frequency mode is close to the kinetic ballooning mode (KBM) branch. This branch is KBM without fast particles and is stable. When the fast particle beta is introduced, this mode becomes localized in ballooning space and has finite growth rate, and can be called RTAE as it transforms into TAE later. Another solution at higher frequency, RTAE is closer to the TAE in frequency and in mode structure. Calculations show that the most unstable mode is the low frequency RTAE and thus is expected to be excited first. Its frequency increases by ~ 50% when fast particle pressure is building up, HINST predicts frequency change for t=3.8sec plasma parameters from f=17kHz to f=30kHz with fast particle beta at the center evolving from \( \beta_p(0)=0\% \) to \( \beta_p(0)=1\% \). The lowest n number modes are the most unstable. These conclusions agree with the experimental observations of the lowest n numbers n=1-2 AE. With time, the fast particle beta profile becomes broader in radial direction so that the region of strongest gradient is moving outwards. Both low-frequency and high-frequency RTAEs merge into one branch at r/a>0.55. The maximum growth rate is shifted to r/a=0.55 with a higher mode frequency. HINST predicts the frequency of RTAEs for this case to be f=50kHz, which is increased due to the higher beta and stronger shear at larger minor radius. Calculated mode frequency chirp qualitatively agrees with that experimentally measured.

Rapidly chirping modes at t=4.0-4.5sec have a characteristic time scale of t=3-5msec. That cannot be related to the slow equilibrium changes, such as considered above. Detailed modeling of rapidly chirping modes needs to include both the nonlinear wave particle interaction and correct nonperturbative plasma response to the evolving beam distribution. We choose the most unstable location for the RTAE for this time at r/a=0.55 and vary the distribution function, which is allowed in the HINST code, so that we may have the combination of both beam like and the slowing down distributions at the same time. We changed the distribution from the beam like to the slowing down distribution and found that around 20% frequency change can be expected when particles are transported radially by RTAE. This is in qualitative agreement with the amount of the observed rapid frequency chirp. Interesting to note that at fast particle beta given by the calculated ratio \( \tau_{chirp}/\tau_{re}=1/4 \)

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1 Princeton Plasma Physics Laboratory, Princeton University
the RTAE is close to the instability threshold and has relatively low growth rate, which ultimately supports our model.

Reference

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4.10 Effects of N-NB fast ions on sawtoothing plasma

K. Tobita, A. Isayama, T. Suzuki

Extension of sawteeth (ST) period has been commonly observed during ICRF or NBI heating in tokamaks. ST stabilization by trapped fast ions is theoretically understood [1], explaining the sawteeth stabilization observed in minority ICRF heating on TFTR and JET [2]. As to JT-60U, N-NBI experiments with 350 keV tangential beams also show the extension of ST period, but the stabilization by trapped particles does not match this case.

The experiment of ST during N-NBI (3 MW) indicates [3] that: i) The ST period $\tau_{ST}$ was extended up to 315 ms with N-NBI of 3.5 MW; ii) The ST period increased with the inversion radius $r_1$ as observed commonly; iii) When N-NB injected into a plasma with a small inversion radius ($r_1/a \sim 0.1$), $r_1$ was expanded slightly ($r_1 = 8 \rightarrow 12$ cm), while no change in $r_1$ was observed for a larger $r_1/a \sim 0.3$; iv) The extension of $\tau_{ST}$ early (until $\sim 0.5$ s) after the N-NBI onset was proportional to the neoclassical resistive time $\tau_R$, while $\tau_{ST}$ was stretched largely late in time during N-NBI (later than $\sim 0.5$ s after the N-NBI onset); iv) The central deposition of N-NBI led to longer $\tau_{ST}$.

Several explanations can be raised for the extended ST; i) electron temperature rise by N-NBI, ii) ST stabilization by trapped fast ions produced via pitch angle scattering, and iii) ST stabilization due to a change in current and pressure profiles. The explanation for the absence of ST in supershots on TFTR with tangential NBI is a decrease in $r_d q'(r_d)$ down to a stability criterion of the two-fluid collisionless $m=1$ reconnection mode [4]. Probably this is not our case where the N-NB beam-driven current in the central region leads to an increase in $r_d q'(r_d)$ contrarily. A possible ST stabilization mechanism proposed by Lazaros is that the radial diffusion of passing ions could non-resonantly interact with MHD modes, destabilizing ST [5]. However, N-NBI sometimes enhanced the amplitude of MHD modes, which does not always match his theory. N-NBI highly increases hot ion pressure inside the q=1 surface, which can be possibly a clue to understanding the ST during N-NBI. This effect should be investigated later on at higher beam power (5-10 MW).

References
4.11 Preliminary experiments of D-^3He fusion power production

T. Nishitani, Y. Shibata, K. Tobita and Y. Kusama

1. INTRODUCTION

The D-^3He reaction is attractive for the alternative fusion reactor, because it does not produce neutrons, and for the estimation of alpha particle behaviors in D-T plasma because the D-^3He reaction produces almost same energy alphas as D-T reaction. The reaction cross-section has strong dependence on the incident energy of D or ^3He. In JET, 140 kW of the D-^3He fusion power was generated with ^3He minority heating by ICRF[1]. The negative ion-base neutral beam injector (N-NBI) [2] with 500 keV is suitable for the D-^3He experiment because the D-^3He reaction cross-section is maximized at D energy of ~500 keV, and has a potential to produce ~1MW of the D-^3He fusion power in 10 MW injection [3].

Reaction equations of D-^3He are shown in Eq.(1) and (2). Almost all of the D-^3He reactions produce protons and alphas which are confined in the plasma, so the direct measurement of the D-^3He reaction rate is very difficult. However, D-^3He reaction has a small branch to produce ^4Li and 16.7 MeV gamma-ray as shown in Eq.(2). We can evaluate the D-^3He reaction rate with the 16.7 MeV gamma-ray measurement.

\[
\begin{align*}
\text{D} + ^3\text{He} & \rightarrow \text{p} \ (14.7 \ \text{MeV}) + \alpha \ (3.6 \ \text{MeV}) \ [100\%] \\
\text{D} + ^3\text{He} & \rightarrow ^4\text{Li} + \gamma \ (16.7 \ \text{MeV}) \ [0.0025\%]
\end{align*}
\]

2. DIAGNOSTIC SET-UP

In the energy range higher than 10 MeV, gamma-ray are measured with electron and positron pair creation process. The cross-section of the pair creation increases with \(Z^2\) where \(Z\) is the charge number of the detection material. So the high \(Z\) material is suitable for the detection of 16.7 MeV gamma-rays. We employed the BGO (Bi\(_2\)Ge\(_3\)O\(_{12}\)) scintillator, whose density is 7.13 g/cm\(^3\), with a size of 3" \(\times\) 3".

Figure 1 shows the arrangement of the gamma-ray detector on JT-60U. The detector is located 15 m below the plasma center and measures emitted gamma-rays in a vertical line-of sight. The floor penetration with \(4 \times 8\) cm\(^2\) is used as a pre-collimator. The detector is mounted inside the heavy collimator which used be for the NE213 neutron spectrometer. The collimator is made with polyethylene as a fast neutron moderator, borated polyethylene as a thermal neutron absorber, and lead as a gamma-ray shield.

Generally, energy and efficiency calibrations of gamma-ray detectors are done with radio-isotope (RI) gamma-ray sources. However, RI sources with the gamma-ray energy higher than 3 MeV are not available. The energy output of the BGO detector was calibrated preliminary with \(^{137}\text{Cs}\) (0.667 MeV) and \(^{60}\text{Co}\) (1.17 and 1.33 MeV). Extrapolation from those energy to 16.7 MeV may give large uncertainty. We calibrated the detector in the high energy region using neutron capture gamma-rays from structural materials in DD discharges. Figure 2 shows the gamma-ray spectrum measured in DD discharges, where \(\text{H}(n,\gamma)\text{D}[2.22 \ \text{MeV}], \ ^{16}\text{O}(n,\gamma)^{17}\text{O}[4.14 \ \text{MeV}], \ ^{52}\text{Cr}(n,\gamma)^{53}\text{Cr}[7.94 \ \text{MeV}], \) and \(^{58}\text{Ni}(n,\gamma)^{59}\text{Ni}[9.00 \ \text{MeV}]\) are identified.
Fig. 1. Arrangement of the gamma-ray detector on JT-60U.

Detection efficiency calibration is much difficult for 16.7 MeV gamma-rays. We calculated the efficiency with the Monte Carlo code MCNP-4B[4] with 3-dimentional modeling of the BGO detector, where the incident gamma-ray beam with 50 mm diameter is well collimated onto the detector head. Figure 3 shows the detection efficiency of the full energy peak and single photon escaped peak as a function of the gamma-ray energy. Single photon escaped peak which corresponds to escape of the photon of the positron annihilation from the detector is almost comparable for 16.7 MeV gamma-rays. Because the energy resolution of the BGO detector is about 14%, we could not distinguish the full energy peak and single photon escaped peak which is 0.511 MeV lower than full energy peak. Here we adopted 40% of the total efficiency including single photon escaped peak.

Fig. 2. Gamma-ray spectrum during DD discharges.

Fig. 3. Detection efficiency of BGO detector calculated with MCNP-4B.
Count rate of the BGO detector for gamma-rays higher than 15 MeV is measured with 10 ms time duration during whole discharges (15 s). Pulse high spectrum is measured with 300 ms time duration and 500 channels of the conversion gain in the CAMAC ADC.

3. EXPERIMENTS

We tried the preliminary D-³He experiments with N-NBI in reversed shear (RS) plasmas [5]. The RS plasma has rather high electron temperature in wide region, which is preferable to increase the fast ion density due to longer showing down time. Figure 4 shows the waveforms of the D-³He experiment in JT-60U.

![Waveforms of the D-³He discharged with N-NB injection in JT-60U.](image)

Gas of ³He was puffed in the plasma initiation and just before the N-NB injection of 3.5 MW with 360 keV. Positive ion NBs of deuterium were used to create the RS configuration and increase the electron temperature. The real time feedback control of the stored energy was used in the current ramp-up phase, therefore the positive NB power was modulated. ECRF of 0.4 MW(plasma input power) was injected into core region to boost the electron temperature. The count rate of the gamma-ray s increases during the N-NB injection.

Figure 5(a) shows the gamma-ray spectrum during N-NB injection in the D-³He experiment compared with that of DD discharge with almost same neutron emission. Two broad peaks around 16 MeV and 14 MeV are identified in the D-³He discharge, which corresponds to γ₀ and γ₁ shown in Fig.5(b). The D-³He fusion power is evaluated to be 110±30 kW from the counts of γ₀ whose fraction is 0.0025% for the total D-³He reaction rate.
Fig. 5. (a) Gamma-ray spectrum during N-NB injection in the D-^3He discharge compared with that of DD discharge with same neutron emission rate, and (b) level scheme of ^5Li.

REFERENCES

4.12 Neutron emission rate during P-NB heating in JT-60U
A. Morioka, T. Oikawa, S. Higashijima, H. Shirai and T. Hiroishi

1. Introduction

It is important to study the heating efficiency and the effect of magnetohydrodynamic (MHD) activity in negative ion based neutral beam (N-NB) heated plasmas. One good way to study these effects is to measure the neutron yield of these discharges and compare it with predictions from TOPICS code [1] and OFMC code [2]. In the calculations of the TOPICS and OFMC code experimental data is used and the calculations are very sensitive to the radial profiles. In this section we report on a study of the neutron yield in a number of positive ion based neutral beam (P-NB) heated discharges without MHD activity in order to assess the accuracy of the OFMC and TOPICS calculations from the measured plasma profiles. The neutrons were detected with an absolute calibrated fission chamber using a neutron source ($^{252}$Cf). The accuracy of the detector is about 10% [3].

2. Calculation

We have calculated the neutron yield with the TOPICS and the OFMC codes. The TOPICS code is a 1.5D transport code that includes the neutron yield from thermal, beam-thermal and beam-beam reactions but it does not include ripple loss, banana drift loss and charge exchange loss of fast ions. Shinethrough losses however are included. The OFMC code is a Monte Carlo code which we were used to calculate ripple, banana drift and charge exchange loss of fast ions. This gave us an effective beam power that was used in the TOPICS code.

To calculate the neutron yield, the other input for the calculations consists of electron temperature profile, electron density profile, ion temperature profile, Zeff profile and radiation loss profile. P-NB injection power is also included.

The electron temperature profiles were measured with the Ruby Thomson Scattering system, YAG Thomson Scattering system and ECE grating polychromatatar. Electron density profiles were measured with the Ruby and YAG Thomson scattering systems. Line averaged electron density were measured by FIR and CO2 interferometers. The electron density profiles were normalized to the FIR interferometer. Ion temperature profiles were measured with charge-exchange recombination spectroscopy (CXRS). Zeff profiles were calculated from the electron density and electron temperature profiles and Bremsstrahlung. Neutron emission rate ($S_n^{cal}$) were calculated using all the basic plasma data.

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The ratio between the measured neutron emission rate and the calculated neutron emission rate are shown in Fig.1. Within the error bar of 10% the measured and calculated values consistent with each other. Similar results were obtained for a number of other tokamaks[4].

3. Summary

Neutron emission rates were evaluated in P-NB heated plasmas without MHD activity. Experimental neutron emission rates between $10^{14}$ n/sec and $10^{16}$ n/sec were compared with calculated neutron emission rates. In the calculation of the neutron emission rates the measured electron temperature, electron density, ion temperature, Zeff and radiation loss profiles were used. The measured and calculated neutron yields agree well with each other.

4. Future

Now we know that we can calculate accurately the expected neutron yield with the TOPICS and OFMC code. We can apply the same technique to N-NB heated plasmas [5]. In the near future, we will study the heating efficiency and the effect of MHD activity.

Reference

5. Plasma Control and Disruption

5.1 Development of advanced feedback control scheme in JT-60U

T. Fukuda, T. Oikawa and K. Kurihara

1. Introduction

The significance of the real time feedback control of plasma density, temperature and other compound physics quantities, such as the normalized beta, has been recognized particularly in the recent JT-60U and DIII-D experiments, where the sustainment of optimized plasma parameters is intensively sought for. The feedback control tools developed in the early phase of the tokamak experiment, such as the electron density feedback, were inherently versatile in nature and concerned with the spatially averaged quantities. In addition, they were principally meant to improve the efficiency of the tokamak operation, namely to regulate the target plasma density for example. However, the recent development is more focused on the control of local quantities for specific purposes, which is effective for the improvement of global stability or confinement. Furthermore, exploration of the multiple feedback algorithm has been emphasized in JT-60U, aimed at the establishment and sustainment of the integrated plasma performance in the fusion reactor, where more than two real time feedback schemes are simultaneously applied to the plasma. In this section, status of the development of feedback tools as well as the result of extended experiment to establish the multiple feedback scheme performed in the 1999 campaign are described.

2. Development of advanced control tools

In order to maximize the efficiency of the experiment and prepare for the advanced plasma control in fusion reactors, several other tools of the real time feedback control were additionally either developed or designed in 1999. Namely, they are intended to control the outer gap, plasma stored energy, local temperature gradient, central electron temperature, main plasma

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radiation power, divertor pressure, localized temperature fluctuations and current density profile. The list of feedback control tools for JT-60U is shown in Table 1, together with their objectives, field of applications and status in other tokamaks.

The feedback control of the outer gap has demonstrated the sustainment of effective coupling of the LH/ICRF power to a plasma throughout the discharge, even if the value of $\beta_p$ is varied by the additional heating. The stored energy control is more directly relevant to the confinement and stability than the neutron emission rate feedback, as it is not influenced by the impurity induced dilution of deuterium. Therefore, it is used more often than the previously developed neutron emission rate control in the routine operation. The possibility of controlling the temperature gradient at the location of ITB also explored, having in mind that the global $\beta_N$ limit in the reversed shear plasmas may be determined by the local pressure gradient. Here, the $T_e$ profile measured by the Michelson ECE is processed to evaluate the local $T_e$ gradient in real time, and on and off-axis NB are employed as the actuators. As a result, it was demonstrated in a reversed shear plasma that the $T_e$ gradient was successfully controlled according to the set value. However, appearance of the ELM induced spikes in the ECE signal obstructed the feedback, indicating that further improvement is necessary. The main plasma radiation control is intended to regulate the amount of seeded impurities in the main plasma in the radiative improved mode experiment, result of which is described in the succeeding section. The divertor pressure control is intended to sustain the detached plasma, and the relevant report is also provided in the following section. As to the control of temperature fluctuations, which is aimed at the suppression of the neoclassical tearing mode and resulting increase of $\beta_N$, heterodyne ECE data was employed to eliminate the effect of ELMs. The location where the temperature fluctuations is the largest is identified in a few ms and the angle of ECH reflection mirror is controlled so that the incident ECRF wave is directed right into the center of the island. The reflector angle is calculated based on the real time processed magnetic equilibrium information, evaluated with the Cauchy condition surface method. The suppression of neoclassical tearing mode by the application of localized ECCD has been demonstrated in ASDEX-U and JT-60, and implementation of the feedback control software is under progress. In addition, the current density profile control is also being developed. Here, the MSE diagnostic data is fed into a real time processor, and LHCD is presently considered as a primary candidate for the first demonstration of the current density profile control in JT-60U.

3. Development of advanced control algorithm

Although various tools have been developed for the real time control of plasma parameters, as described in the previous subsection, it is anticipated in the fusion reactor that the simultaneous feedback control of more than two plasma parameters is necessary to achieve and sustain the integrated fusion performances. The schematic of the "advanced feedback control" is shown in Fig. 1. In the present experiment, integration process such as the improvement of H factor and

Fig. 1 Schematic of the advanced feedback concept
\( \beta_N \) under the fully noninductive current drive condition involves the deliberate optimization of multiple parameters, and it is performed in shot by shot basis, modifying the preprogramed discharge conditions. Accordingly, multiple feedback experiment was first carried out in JT-60U \(^1\), in order to investigate the seemingly nonlinear interactions among the elements of integration. In the extended investigation, related to the multiple feedback performed in 1999, the stored energy control was employed instead of the neutron emission rate, so that the value of \( \beta_N \) is directly addressed. The waveforms obtained in the multiple feedback experiment is shown in Fig. 2. In this case, the plasma density was kept at around half the Greenwald density, whilst the fraction of radiation power in the divertor to the total heating power was varied from 40 to 50\%. At the respective stage of the radiation power fraction, the plasma stored energy was controlled in the range of (1.2-1.5) MJ. Here, the discharges stayed in the ELMy H mode, and the position as well as the shape of the plasma were kept fixed throughout the period of multiple feedback control from 5.0 s to 11.0 s. The relation between the control variables and feedback actuators is also indicated in the left hand side figure of Fig. 2 and in the diagonal components of the following matrix form.

\[
\begin{pmatrix}
[n_{el} \\
W_{dia} \\
P_{\text{div rad}}
\end{pmatrix} =
\begin{bmatrix}
G_{11} & G_{12} & G_{13} \\
G_{21} & G_{22} & G_{23} \\
G_{31} & G_{32} & G_{33}
\end{bmatrix}
\begin{pmatrix}
\text{Main gas puff} \\
\text{P}_{\text{NB}} \\
\text{Divertor gas puff}
\end{pmatrix}
\]

It is readily observed that the H factor decreases with an increase of \( \beta_N \), and recovers as \( \beta_N \) is stepped down to the original value, even of the radiation power fraction is increased by 25\%. Here, H factor refers to the ITER89P L mode scaling. Thus, investigation of plasma response to the step modulation of specific physics quantity under the condition that other control variables are fixed, provides information directly relevant to the mutual relationship among the control variables. In this case, it is shown that \( \beta_N \) and H factor have antagonistic feature, which is convincing and consisted with the previous experimental results. It has been found that the degree to what extent the improvement of \( \beta_N \) and H factor conflicts to each other is a function of the density. The evaluation of the matrix elements is presently underway.

![Diagram](image)

**Fig. 2** The schematics of the control parameters and feedback actuators (left). Waveforms of the multiple feedback control (right). Here, respective subpanels are for (a) Lin integrated density and its set value, (b) radiation power fraction and its set value, (c) stored energy and its set value, (d) \( \beta_N \) and H89 and (e) NB heating power.
Fig. 3 Waveforms of the multiple feedback control. Here, respective subpanels are for (a) Line integrated density and its set value, (b) radiation power fraction and its set value, (c) stored energy and its set value, (d) $\beta_N$ and H89, (e) NB heating power and (f) recycling flux in the divertor.

...to resolve the mutual relationship among the density, $\beta_N$ and divertor radiation fraction, if they could ever be written using the linear expression.

Further experimental effort was devoted to partly simulate the feedback control the reactor plasma was also produced, related to the multiple feedback control. Having the $Q = 10$ inductive operation in ITER-FEAT, twice as much the controlled heating power was applied steadily as a base to a low q ELMey H mode plasma, during the feedback period of 6 to 10.25 s, as shown in Fig. 3. In this case, the target density was slightly reduced, and plasma shape and positions were modified from the discharge shown in Fig. 2, in order to heat the central part of the plasma intensively. Accordingly, relatively high $\beta_N$ value of 2.5 was obtained as a target plasma, for which the multiple feedback was applied. The value of was reduced stepwise from 6 to 9.25 s, whilst the radiation power fraction was kept at 40%. In the later phase of the discharge, on the other hand, the radiation power fraction was increased to 60% with $\beta_N$ kept fixed at a value slightly lower than it was initially. It was hereby observed that H factor decreases with $\beta_N$, contrary to the case shown in Fig. 2. In addition, due to an increase of the radiation power fraction at 9.75 s, in comparison with the value at 6.25 s, H factor is reduced at a similar value of $\beta_N$. It is noteworthy here that a large amount of heating power is applied to increase $\beta_N$, which resulted in the degradation of the confinement. In order to simulate the burning plasma, it is necessary to modify the control algorithm so that the base heating power, which is not directly controlled, changes according to the value of $\beta_N$.

References
5.2 Feedback control of neutral pressure ratio by divertor gas puffing

H. Tamai, N. Asakura, T. Oikawa, T. Fukuda, and S. Sakata

1. Introduction

Divertor detachment should be controlled to maintain for the reduction of the divertor heat load. However, in a highly radiated divertor plasma the radiation front easily goes up along the separatrix and induces the X-point MARFE, which causes a degradation of the core plasma confinement. Therefore, a detached divertor plasma at both sides of the divertor plates should be maintained without inducing an X-point MARFE.

In order to keep a detached divertor plasma with high recycling and high radiation fraction, several parameters such as averaged density, divertor radiation, and divertor neutral pressure are candidates for the actuator of feedback control. The onset density of the X-point MARFE has been empirically predicted for the various plasma configuration and wall condition. However, at the outer strike point the transition from attachment to detachment and that from detachment to X-point MARFE occurs at almost the same line averaged density.\(^1\) Moreover, onset density depends on the divertor gap and X-point height, so that it is difficult to precisely suppress the X-point MARFE by the feedback control of line averaged density.

Divertor radiation will become a prompt actuator if the effect of the gas puff on the enhancement of carbon emission, which is considered to be a dominant radiation flux, will be identified, and good resolution of viewing chord of bolometer array will be ensured.

In previous campaign of 1998, a rapid change of in/out asymmetry in divertor neutral pressure has been observed during the phase of both-side-detachment and succeeding X-point MARFE\(^1\). In this campaign of 1999 after modification of divertor structure with pumping slot from both private regions, both-side-detachment is observed in a remarkably extended regime of the outside to inside pressure ratio of 0.4-1.3,\(^2\) (as shown in 6.4)

The outside/inside neutral pressure ratio is considered as a good measure for the divertor detachment, so that the sustainment of both-side-detachment by the feedback control with an actuator of the divertor pressure ratio is performed for the first time.

This time divertor gas puff is chosen on the point of view of direct control of divertor plasma, so as to minimise the particle recycling around the main plasma.

2. Feedback control system

Feedback control system for the divertor neutral pressure ratio is illustrated in Fig.1, which is similar to that for the the inside divertor pressure.\(^3\) Measured signal of the ionisation gauge at the inside and outside divertor is fed to the real-time processor to calculate the divertor neutral pressure ratio.\(^4\) From the calculated divertor neutral pressure ratio a command signal of the gas puffing rate, \(Q^\text{com}\), is determined in every 10 milliseconds in the supervisory computer, then the piezoelectric gas valve is switched to puff the proper gas into the divertor region.
Feedback control system by output pressure ratio

Feedback control of the pressure ratio is evaluated by the following equation.

\[
d\frac{R^{obs}}{dt} = \alpha \left( Q^{com} - Q^c \right) \quad \text{Eq. 1}
\]

\[
Q^{com} = G_p \left( R^{ref} - R^{obs} \right) + G_d \frac{d}{dt} \left( R^{ref} - R^{obs} \right) + Q^{pre} \quad \text{Eq. 2}
\]

where \( R^{obs}, R^{ref}, \alpha, Q^c, Q^{pre}, G_p, G_d \) are the observed, and reference pressure ratio, linear constant, threshold, and pre-programmed gas puff rate, proportional, and differential gain, respectively. From Eq. 1 and Eq. 2

\[
d\frac{R^{obs}}{dt} = \alpha \left[ G_p \left( R^{ref} - R^{obs} \right) + G_d \frac{d}{dt} \left( R^{ref} - R^{obs} \right) + (Q^{pre} - Q^c) \right]. \quad \text{Eq. 3}
\]

From the observation of the pressure ratio to the constant pre-programmed gas puff rate, the proportional gain is estimated as \( G_p = 1 / \alpha \tau \), where \( \tau \) is the characteristic time.

Figure 2(a) shows the time response of the pressure ratio \( R^{obs} = (P_{out} / P_{in}) \) to the pre-programmed gas puff rate of \( Q^{pre} = 15 \text{Pam}^3/\text{s} \). From the figure, \( dR^{obs}/dt = 1.73 \text{s}^{-1} \) is estimated and response delay of about 0.2s is suggested. Figure 2(b) shows the \( dR^{obs}/dt \) plotted against the pre-programmed gas puff rate of 15\text{Pam}^3/\text{s}, and 100\text{Pam}^3/\text{s}. Linear constant \( \alpha = 0.198 \), estimated from the gradient, and assumption of \( \tau = 0.01 \text{s} \) for the interval of control step, then, the proportional gain of \( G_p = 504 \) is deduced.

Fig. 2(a) Response of the output pressure ratio to the pre-programmed gas puff. Fig. 2(b) Deduction of linear constant \( \alpha \).
3. Trial of feedback control for keeping divertor detachment

The feedback control of divertor neutral pressure ratio is performed in order to demonstrate to keep the divertor detachment without inducing an X-point MARFE. In the feedback control of pressure ratio there exists certain difficulty resulted from the different response of inside and outside divertor pressure: At the low recycling phase, increase in inside pressure, $p_{in}$, is larger than that in outside pressure, $p_{out}$, then, the pressure ratio, $p_{in}/p_{out}$, decreases by gas puffing at first. So that the feedback loop is considered to not operate well. Therefore, actual feedback control of divertor neutral pressure ratio is started after the divertor recycling is enhanced. For this purpose, the averaged density is maintained at certain amount by a simultaneous feedback control using with main gas puff. No interference is observed between two feedback control, which is the effect of small divertor gap and small X-point height in the present both sides pumping 3).

Figure 3 shows the temporal behaviour of plasma parameters in a typical shot of the feedback control of divertor neutral pressure ratio. Pressure ratio is set constant at 0.5, which is relatively low level in the range of pressure ratio of 0.4-1.3 for both side detachment. Line integrated density is controlled at $4 \times 10^{15}$ m$^{-2}$, which corresponds to 42% of Greenwald density. Around 5.9s observed pressure ratio starts to increase over the pre-programmed one, and gas puff is shut off. Then, the pressure ratio decreases at 6.2s, which results in the increase of the gas puff rate, and reaches the pre-programmed pressure ratio around 6.8s. Those temporal evolution indicate that the feedback loop is anyhow working, but not adequate perhaps because the feedback gain is not properly adjusted.

Pressure ratio decreases after 6.8s despite of the increase of gas puff rate, consequently X-point MARFE occurs at 7.6s. Both inside and outside divertor pressure decrease before the onset of X-point MARFE. Rapid increase in outside divertor neutral pressure around the onset of X-point MARFE, which has been observed in the case of main gas puff, is not observed.

Those results suggest the non-linear response of divertor pressure to the divertor gas puff in a high recycling divertor. In low gas puff rate divertor neutral pressure ratio behaves as predicted, however in high gas puff rate it does not behave as the same manner with that by

Fig.3 temporal behaviour in the feedback control.
main gas puff.

One of the reason for such a different behaviour in the case of divertor gas puff might be the affect of the baffle geometry and the conductance, which results the delay of the response to the divertor gas puff, in consequence followed by the build-up of large volume under the divertor baffle. Difference of the particle fuelling from downstream side (divertor gas puff) and from upstream side (main gas puff) might be another likely reason.

In the case of main gas puff, however, the reproducibility of both side detachment in almost the same range of divertor neutral pressure ratio as previous experiment is still confirmed. For the successful sustaining of divertor detachment without inducing an X-point MARFE, feedback control of pressure ratio by main gas puff should be tried.

Acknowledgement

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References

5.3 Low Voltage Electron Cyclotron Heating
Assisted Startup in JT-60U 1)

K. Kajiwara and R. Yoshino

The development of low voltage startup scenarios for large tokamaks is necessary to reduce breakdown loop voltage. Electron cyclotron heating (ECH) is effective to reduce breakdown loop voltage. In ITER, it is proposed that the electric field which applied for ionization and plasma current ramp-up is limited to E < 0.3 V/m. Reduction of the breakdown loop voltage has been investigated in many tokamak machines. The electric field of 0.15 V/m was achieved in DIII-D using ECH preionization 2). In JT-60U, the breakdown loop voltage with 0.08 V/m was achieved with LHRF and Helium gas injection for prefilling 3).

Study of low voltage startup by electron cyclotron preionization and preheating has been carried out in JT-60U. ECH assisted startup with E ~ 0.24 V/m (loop voltage of 4V) has been demonstrated. The ECH power was 0.75 MW and pulse length was 500 ms. The experiments were carried out in the vicinity of the fundamental cyclotron resonance using an ordinary mode launch from low field side. ECH injection was started 30 ms before voltage applying to ohmic heating (OH)-coil and vertical (V)-coil. The constant voltage was applied to OH-coil and V-coil during first 100 ms. After the constant voltage phase, the feedback control system was started and the plasma current increased according to the preprogramming of plasma current (dIp/dt = 0.2MA/s).

The line density just after ECH injection is 2.2x10^{18} m^{-2}. If the minor radius of 1 m is assumed, the density of preionized plasma is roughly estimated about 1x10^{18} m^{-3} with the prefilling pressure of 6x10^{-6} Torr. The breakdown time is 10 ms. It is much faster than OH breakdown. In the constant voltage phase, the plasma current steeply increases (dIp/dt = 1.36 MA/s). Such a high speed plasma current ramp-up is not desirable. It is necessary for operation of superconductor tokamak to reduce plasma current ramp-up rate. The reduction of the dIp/dt will be achieved by further decreasing loop voltage at the breakdown.

This experimental result is compared with a quasi 0D code which has been developed to analyze burnthrough in ITER 4). This code is applied to JT-60U configuration and estimates the experimental result. We discuss the extrapolation of ITER from JT-60U result.

1) K. Kajiwara and R. Yoshino, to be submitted to Nucl. Fusion.
2) B. Lloyd et al, Nucl. Fusion, 31 (1991) 2031
5.4 Runway current termination in JT-60U

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Runaway electrons generated at the major disruption are considered to significantly reduce the lifetime of the first wall in tokamak fusion reactors, while the first wall should be thin as possible in order to convert the neutron energy in the blanket with high tritium breeding rate. Hence, establishment of methods for avoidance, suppression and termination of runaway electrons is a key issue in present tokamaks. From this point of view, the behaviour of runaway electrons for three types of magnetic turbulence in tokamak discharges has been reviewed: Those are (a) Micromagnetic turbulence, (b) Low-m/n magnetic islands, and (c) Macroscale magnetic turbulence.

The experimental observations of improved confinement of runaway electrons, in the case of micro-magnetic turbulence and in the case of drift islands surrounded by a sea of stochasticity (runaway snake), are well confirmed by theoretical analysis and 3-D particle simulation, respectively. The confinement of runaway electrons is much better than that of bulk thermal electrons. Degraded confinement of runaway electrons in macroscale magnetic turbulence, as observed during major disruptions in JT-60U, can also be explained by the breakdown of toroidal momentum conservation due to the toroidal asymmetry of magnetic perturbations. However, this confinement degradation effect of macroscale magnetic turbulence on runaway electrons is not clear in smaller size tokamaks. Thus critical parameters (e.g. plasma size and toroidal magnetic field strength) that intensify the confinement degradation effect of macroscale magnetic turbulence should be investigated further by 3-D particle simulation.

Fast termination of runaway current was first demonstrated in JT-60U by reducing the safety factor at the plasma surface \(q_s\) (or the effective safety factor at the plasma edge, \(q_{\text{eff}}\) for divertor plasma) by either a controlled inward plasma shift, a VDE (uncontrolled vertical plasma shift) or a rampup of plasma current by 1 MA/s. A sudden decrease in runaway current is always observed at \(q_s\) around 2 or 3 (or \(q_{\text{eff}}\) around 3). This suggests that runaway current may be spontaneously terminated for VDEs in tokamak fusion reactors like ITER due to a natural decrease in \(q_s\). Possible boundary-free MHD activities that cause abrupt termination are low-n external kink modes or surface tearing modes, which should be investigated by experiments and theoretical analyses. The spontaneous and intrinsic termination of runaway current will greatly reduce the energy flux on the first wall and the halo current, which should be also experimentally confirmed.

References
5.5 Effects of the impurity gas puff related to the Halo Current in JT-60U

Y. Neyatani, R. Yoshino, T. Hatae, T. Nakano, H. Tamai

1. Introduction

A current flowing directly into a vacuum vessel from plasma (called halo current) is observed during disruptions in tokamaks [1]. This halo current will produce an intense electromagnetic force on the in-vessel component in ITER, such as the blanket module and the divertor cassette. Thus, the halo current is one of the critical issues.

The halo current data base in JT-60U shows that the maximum TPFxI/Ip0 was 0.52, where TPF is the toroidal peaking factor, I, is halo current and Ip0 is the initial plasma current just before the energy quench [2]. One of the method of the reduction of halo current was proposed to attempt the impurity gas puffing to reduce the electron temperature in the halo region effectively because the magnitude of halo currents depends on resistivity of current path of halo current. In previous experiments, when neon gas of 3.3 Pam^3/s x 0.17 s was applied during VDE, the magnitude of the halo current decreased by about 60% of those for no gas puff case [3]. The reduction of electron temperature is observed with gas puffing during VDE. The electron temperature in the halo region at the time of the maximum halo current is around 10 eV. To confirm the causes of reduction of halo current more clearly, various species of injection gases are attempted.

2. Effects of pulsed impurity gas puff for the reduction of halo current

Pulsed gas puff was tried during disruptions and good results was obtained for the reduction of the halo current by H2 puff in JT-60U [2] and He puff in DIII-D [4]. Based on the measured T_e, reduction method of the halo current has been optimized. The radiation cooling rate of neon and argon in the range of 10-100 eV, which temperature was the same range of T_e during Ip quench, was higher than those of another gases [5]. Thus, both gases are expected to plasma cooling more effective. To estimate the effect of impurity species, neon, argon and helium was used as injected gases.

Typical time traces of discharges with and without neon gas puff was shown in Fig.1. Vertical displacement event (VDE) was simulated to push the plasma downward at 13.84 s with the vertical velocity of 2.5 m/s. In case of no gas puff, plasma current kept constant at 0.7MA until 14.8s. Safety factor of plasma surface (q_e) reached 2. After that, current quench started and the halo current generated. Maximum magnitude of halo current of TPFxI/Ip0 reached 0.2. When neon gas of 16 Pam^3/s was applied during VDE, energy quench occurred before q_e reached 2. After energy quench, plasma current and vertical position rapidly decreased. Halo current generated after the decrease of plasma current. The magnitude of the
maximum halo current of $TPFx_l/l_{po}$ was 0.06, which was 30\% of those for no gas puff case. Neon gas was applied after VDE start. This means that the gas puffing method can be utilized after the detection of vertical displacement when VDE period is long enough to cool down the plasma by impurity gases (in JT-60, order of 10 ms in case of neon gas puff of 16 Pam$^3$/s).

![Fig.1 Time traces of simulated VDE with and without neon gas puff](image)

The reduction rate of halo current depends on total impurity gas injected in vacuum vessel as shown in Fig.2. Total gas puff until the time of which halo current reached maximum during a discharge was estimated to subtract the travel time of injected gas from the gas inlet valve to the plasma. The time interval of this delay was around 40 ms and was estimated to check the delay from the time to open gas inlet valve to the increase of plasma density. In case of argon gas puff, similar halo current reduction was obtained. Magnitude of halo current with argon puff is almost the same within the diagnostic error of 30\% as shown in Fig.2. In case of helium puff, more than 10 times large amount of gas puff (8–10 Pam$^3$) is required to reduce the $TPFx_l/l_{po}$. Thus, we confirm that the neon and argon puff is more effective to reduce the halo current.

![Fig.2 Relation between magnitude of halo current and total amount of gas puff](image)

Amount of helium gas puff indicates 1/10 of injected value.
Cooling of plasma edge by neon injection was confirmed in Fig. 3. Edge electron temperature measured by YAG Thomson scattering diagnostics was reduced with the increase in the total amount of gas puff at the time of YAG Thomson scattering measurement. All data in Fig. 3 indicates the data before the energy quench. The measured radial position was r/a = 0.9-1.1. The electron temperature at the maximum halo current can estimate by extrapolation of the data. It suggests that the significant reduction of halo current can be achieved at the electron temperature below 10eV. From the result, large amount of neon gas puff is effective to reduce the edge plasma temperature even in short time of order 10 ms.

When the initial $q_s$ is high (>5) with gas puffing, $q_s$ at the energy quench was high because the energy quench occurred in the early phase of simulated VDE. After the energy quench, $q_s$ decreased rapidly and the $q_s$ at the time of maximum halo current remained more than 2. Discharge at $I_p = 0.7$ MA with neon gas puff started from $q_s = 12$ and the energy quench occurred at $q_s$ around 10 (Fig. 4). After the energy quench, $q_s$ decreased to 7 during 30 ms. Thus, final $q_s$ at the maximum halo current remained to be high. In case of high $q_s$ without gas puffing, $q_s$ at the energy quench reached 2. After that, final $q_s$ at the maximum halo current decreased around 1. The energy quench may occurred caused by low m (m=2) MHD activities at $q_s$ around 2. On the other hand, in low $q_s$ operation, such as ITER like $q_s$ ~ 3, $q_s$ at the energy quench, $q_s$ was already low even with gas puffing case. In case of $I_p = 1.5$ MA and initial $q_s = 4$, final $q_s$ was around 2 with gas puff. The TPFXI$_B$/I$_{p0}$ was around 0.2, which is larger than that of high $q_s$ case of typically < 0.12. These results suggest that both reduction of electron temperature and high $q_s$ are possible cause of the low halo current for neon gas puffing.
Edge electron temperature was decreased with the increase in total amount of gas puff. Whereas, central electron temperature after the E.Q. was not differs so much. Thus, impurity gas mainly cools down the peripheral plasma where the halo current flowing. This suggests that the edge cooling is essential for the halo current reduction and cooling of plasma center is not necessary. This means that the gas puffing is reliable method for the halo current reduction even in large size plasma, which core plasma does not cooled down by gas puffing only.

We conclude that the neon and argon gas puffing is very effective to reduce the halo current because of low electron temperature in the halo region (<10eV) and a higher q, of more than 2.

References
6. Neutral Particle Control and Divertor/SOL Physics

6.1 Improvement of divertor performance with both-leg pumping

A. Sakasai, H. Takenaga

1. Modification of W-shaped divertor

The W-shaped divertor of JT-60U was modified from inner-leg pumping to both-leg pumping. In the W-shaped divertor of JT-60U, the outer exhaust slot, which has an aperture of 2 cm, was added to the existing inner one (the aperture of 3 cm) as shown in Fig. 1(a). In the case of inner-leg pumping, carbon fiber composite (CFC) tiles were used for divertor plates, top tiles of the dome and baffling tiles at the divertor throat, and graphite tiles were used for the other parts so far. Therefore, heat load to the dome bottom exceeded the limited surface temperature of the graphite tiles in the configuration of the inner/outer separatrix close to the inner/outer slots. The divertor configuration had to be kept the gap-in and gap-out > 3 cm (i.e. distances between the inner/outer separatrix and the inner/outer slots).

Then, all tiles of the dome were switched from graphite to CFC to prevent the problem of heat load to the dome bottom tiles. After the modification, divertor-closure configuration, which means that the gap-in and gap-out are close to 0.5–1.0 cm and the divertor throat become narrow with lower X-point configuration, was enabled. The divertor experiments with both-leg pumping were started from February in 1999. The effective pumping speed for both-leg pumping was estimated to be 15.9 m³/s at about 0.1 Pa by using a gas filling method, which is 25% higher than the one for inner-leg pumping.

2. Improvement of Pumping Rate

Previous studies in W-shaped divertor with inner-leg pumping indicated the private dome and inclined target type divertor functioned to prevent the upstream transport of hydrocarbons generated by chemical sputtering, and to reduce resultant carbon influx to the main plasma [1]. The inner-leg pumping was effective in attached divertor because of inboard-enhanced deuterium flux. On the contrary, it was not effective in detached divertor because of a

Fig. 1. (a) The W-shaped divertor of JT-60U with both-leg pumping. (b) The pumping rate as a function of the gap with both-leg pumping and inner-leg pumping in ELMy H-mode plasmas.
remarkable increase in neutral pressure near the outer strike-point with no pump. Actually, the X-point MARFE onset density was slightly reduced in the W-shaped divertor with inner-leg pumping as compared to the open divertor without pump.

Fig 1(b) shows the pumping rate as a function of the gap with both-leg pumping and inner-leg pumping in ELMy H-mode plasmas at $I_p=1.2$ MA, $B_t=2.5$ T, $P_{NB}=12$ MW. The pumping rate is defined by the ratio of the deuterium particle flux exhausted with pumping to the deuterium particle flux in the divertor. After the modification to both-leg pumping, the pumping rate strongly depends on the gap-in and gap-out in L- and H-mode plasmas. Neutral particles accumulated in the inner private region are exhausted through the space under the dome. In the case of large gap-out, the back-flow to the outer divertor occurred through the outer slot and the pumping rate deteriorated [2]. Actually, the pumping rate at the gap-in and gap-out of 3.0 - 3.5 cm with both-leg pumping was estimated to be about 70% of the one at the gap-in of 3.5 cm with inner-leg pumping. In order to improve the pumping rate, the divertor-closure to prevent the back-flow to the outer divertor is key point. The pumping rate was improved up to 4% with both-leg pumping in a divertor-closure configuration at the gap-in and gap-out of 0.5 cm from 2% with inner-leg pumping at $\bar{n}_e = 3.5 - 4.0 \times 10^{19}$ m$^{-3}$. However, the pumping rate reduced to 1% in the lower density region at $\bar{n}_e = 2.5 - 3.0 \times 10^{19}$ m$^{-3}$. The pumping rate depends on the edge density, which means the particle recycling flux in the divertor [3].

3. Impurity Reduction

Impurity shielding (friction force) with high-density divertor and plasma flow with puff and pump are essential to reduce impurity level in the core plasma. At the gas puff rate of 40 - 70 Pa·m$^{-3}$/s, the carbon impurity level in the core plasmas was reduced from $Z_{\text{eff}}=3.2$ with inner-leg pumping to $Z_{\text{eff}}=2.6-2.3$ with both-leg pumping in ELMy H-mode discharges ($I_p=1.2$ MA, $B_t=2.5$ T, $P_{NB}=18$ MW) as shown in Fig. 2 [1]. The impurity level with both-leg pumping was lower. In the case of inner-leg pumping, the X-point MARFE onset density was found to be $\bar{n}_e/n^{Gr} = 0.50$ [4]. The onset density increased up to $\bar{n}_e/n^{Gr} = 0.63$ with both-leg pumping in a divertor-closure configuration. The onset density strongly depended on the impurity level. The $Z_{\text{eff}} < 1.5$ was obtained in the case of the higher onset density.
4. Conclusions

The W-shaped divertor of JT-60U was modified from inner leg pumping to both leg pumping. The outer exhaust slot, which has a gap of 2 cm, was added to the existed inner one (the gap of 3 cm). The divertor experiments with both leg pumping was started from February in 1999. First of all, the effective pumping speed for both leg pumping was estimated to be 15.9 m³/s at about 0.1 Pa by using a gas filling method, which is 25% higher than one for inner leg pumping. The pumping rate strongly depends on the gap_in and gap_out (i.e. distances between the inner/outer separatrix and the inner/outer slots) in L- and H-mode plasmas. As a result, the main gas puff and pumping in the low Xp discharges was very effective for the reduction of impurity level and the enhancement of impurity shielding.

References
6.2. Particle exhaust in both sides pumping\(^1\)

H. Takenaga, A. Sakasai, H. Kubo, N. Asakura

The pumping scheme in the W-shaped divertor was modified from pumping through the inner side private flux region (IPP) to pumping through both inner and outer side private flux regions (BPP). Figure 1 shows the ratio of the pumping flux ($\Phi_{\text{div}}$) to the total D$\alpha$ emission ($I_{\text{D$\alpha$$\text{total}}}$) integrated in the whole plasma as a function of $n_e$, in the cases with Gap$\text{in}=3.5$ cm for IPP and Gap$\text{in}=3.5$ cm/Gap$\text{out}=3$ cm for BPP, where Gap$\text{in/out}$ is the distance between the inner/outer strike point and inner/outer pumping slot. The ratio is in the range of 0.03-0.27 for BPP, which corresponds to the pumping ratio of 0.2-1.8% to the recycling flux with (ionization events)/(D$\alpha$ emission)=15. The value of $\Phi_{\text{div}}/I_{\text{D$\alpha$$\text{total}}}$ in BPP is smaller than that in IPP, especially for the data without X-point MARFE. The ratio of the integrated D$\alpha$ emission in the inner divertor to that in the outer divertor is estimated to be 3.3-3.8 without X-point MARFE and 1.5-2 with X-point MARFE. This result indicates that the back-flow at the outer pumping slot due to in-out asymmetry could cause a reduction of $\Phi_{\text{div}}/I_{\text{D$\alpha$$\text{total}}}$ in BPP, which is consistent with the UEDGE\(^2\) and DEGAS2\(^3\) simulations.

Figure 2 shows the dependence of $\Phi_{\text{div}}/I_{\text{D$\alpha$$\text{total}}}$ on Gap$\text{in}$/Gap$\text{out}$. BPP data are plotted as a function of Gap$\text{out}$ and IPP data are plotted as a function of Gap$\text{in}$. Gap$\text{in}$ for BPP is 3.5 cm for the case of Gap$\text{out}>3$ cm and 1.5 cm for the case Gap$\text{out}=0.5$ cm. The value of $\Phi_{\text{div}}/I_{\text{D$\alpha$$\text{total}}}$ for BPP steeply increases at Gap$\text{out}=0.5$ cm with X-point MARFE. However, $\Phi_{\text{div}}/I_{\text{D$\alpha$$\text{total}}}$ for BPP without X-point MARFE are low even with the small Gap$\text{in/out}$ due to the strong in-out asymmetry of the recycling flux.

![Graphs showing the dependence of $\Phi_{\text{div}}/I_{\text{D$\alpha$$\text{total}}}$ on $n_e$ and Gap$\text{in/out}$](image)

**Fig. 1** $\Phi_{\text{div}}/I_{\text{D$\alpha$$\text{total}}}$ as a function of $n_e$. Open and closed symbols show the data without and with X-point MARFE.

**Fig. 2** $\Phi_{\text{div}}/I_{\text{D$\alpha$$\text{total}}}$ as a function of Gap$\text{out}$ for BPP or Gap$\text{in}$ for IPP. The meanings of the symbols are the same as Fig. 1.

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1) H. Takenaga, et. al., to be submitted to Nuclear Fusion.
6.3 Pumping effect on the divertor plasma and detachment in the W-shaped divertor[1]

N. Asakura, S. Sakurai, H. Tamai, Y. Koide, Y. Sakamoto, O. Naito, K. Masaki

Control of the plasma flow in the scrape-off layer (SOL) and divertor, using a divertor pumping system, is considered important because of its implications for the exhaust of helium ash and impurity retention in the divertor. Understanding effects of the divertor pumping on the SOL plasma and neutral recycling is a critical issue to design the divertor geometry and pumping system efficient for a tokamak reactor. Experimental results have been obtained in various divertor geometries and pumping slot locations. However, there were few measurements of the SOL plasma flow with changing divertor pumping rate, and the effect on the plasma flow pattern is not understood. In JT-60U, a pumping slot in the outer side of the W-shaped divertor was opened since 1999, and pumping from the private flux region in the inner and outer divertors (both-sides pumping) has been used to increase effective pumping rate for high density and detached divertor operations. The SOL plasma flow for the cases of gas puff at plasma top and pumping at the down-stream side (divertor private flux region) is investigated using the reciprocating Mach probes installed at the outer midplane and the x-point[2]. Effect of bypath between the inner and outer divertors under a dome on the particle recycling is also studied.

Plasma flow in SOL and divertor

Profiles of $T_e$ and $j_e$ in the SOL were measured in the L-mode plasmas, where $I_p = 1.6-1.7$ MA, $B_t = 3.5$ T, $q_{95} = 3.5$ and the ion $\nabla B$ drift direction towards the divertor. The direction of the plasma flow along the field lines is deduced from the ratio of $j_e$ at the down-stream side (divertor side) of the Mach probe to $j_e$ at the up-stream side (midplane side), i.e. $j_{e, \text{down}}/j_{e, \text{up}}$.

The profiles of $j_e$ ratio for the both-sides pumping and inner-side pumping cases are comparable both at the midplane and near x-point. Flow reversal occurs at the midplane, while the plasma flows from the x-point to the divertor plate. A mechanism to produce the flow reversal at the midplane has been proposed based on the in-out asymmetry in the ion poloidal drift in a torus[3]. Plasma flow near the x-point is driven towards the divertor target, which is a plasma sink under attached divertor conditions.

The flow reversal at the midplane decreases with increasing $n_e$ and the plasma flow just below the x-point increases under conditions in which the attached plasma is maintained. For the both-sides pumping, the divertor plasma flow tends to increase at high density of $n_e = (2.0 - 2.56) \times 10^{19}$ m$^{-3}$, i.e. $j_{e, \text{down}}^{X_{f}}/j_{e, \text{up}}^{X_{f}}$ of 0.02–0.03 is smaller than $j_{e, \text{down}}^{X_{f}}/j_{e, \text{up}}^{X_{f}}$ of 0.05–0.06 for the inner-side pumping alone. At the same time, small reduction of the flow reversal flow is observed at midplane. These facts suggest that the increase in the plasma flow at the outer divertor by the both-sides pumping is not large enough to change the plasma flow pattern. The ion poloidal drift rather than the parallel flow may produce net plasma flow at the main plasma edge.

Pumping effect on divertor detachment

An improvement of the divertor operation is observed for the both-sides pumping case: partial-detached plasma at both divertor targets without causing the x-point MARFE (and plasma detached below the x-point) is maintained providing that $n_e$ is kept in the adequate range of (2.3–2.56) $\times 10^{19}$ m$^{-3}$. Greenwald density fraction $n_{e, \text{Gr}}/n_{G_{r}}$ corresponds to 0.46–0.52. For the inner divertor pumping, plasma detachment at both divertor sides is observed only at $n_e \sim 2.45 \times 10^{19}$ m$^{-3}$, and the x-point MARFE appears at higher $n_e$.

With increasing $n_e$ in the above regime, $T_e^{X_{f}}$ decreases from $\sim 60$ eV to 10–30 eV, while $p_e^{X_{f}}$ is maintained at 230–340 Pa both for the inner-side and both-sides pumping.
cases. In contrast, $2p_{\text{e, out}}^{\text{div}}$ for the both-sides pumping decreases, i.e. plasma detachment occurs at the target plate. The ratio of $2p_{\text{e, out}}^{\text{div}}/p_{\text{e}}^{Xp}$ is smaller than that for the inner-side pumping case, which is mostly due to a large reduction in the ion flux at the divertor target, while $T_{\text{e}}^{\text{div}}$ of 5–10 eV is comparable. The larger plasma flow just below the x-point for the both-sides pumping is produced by larger reduction in $p_{\text{e, out}}^{\text{div}}$. The neutral source at the separatrix is presumably reduced by the divertor pumping from the private flux region. The larger reduction in $p_{\text{e, out}}^{\text{div}}$ is also observed during the x-point MARFE, where the detachment region extends to the up-stream (x-point) and the outer flux surfaces.

The neutral density in the private flux region is increased by gas puff from the divertor (divertor-puff). Plasma detached near the separatrix (0–35 mm) during x-point MARFE, where $j_{\text{s, up}}^{Xp}$ and $T_{\text{e}}^{Xp}$ profiles are identical for the two cases. Values of $j_{\text{s, down}}^{Xp}$ in the region of the private flux and separatrix for the divertor-puff are by a factor of two larger than that for the main-puff case, and the flow reversal extended to the x-point. The plasma pressure at the down-stream increases due to the divertor-puff. This is consistent with the neutral pressures measured with fast ion gauges at the private flux region[4]. Neutral pressures at inner-side dome, outer-side dome and dome top are 1.7, 2.5 and 0.32 Pa for the divertor-puff, respectively, which are about a factor of two larger than those for the main-puff (i.e. 1.2, 1.6 and 0.18 Pa). Since plasma detachment extends to the x-point, neutral density at the dome top is increased proportional to those at the dome sides.

**In-out asymmetry in divertor recycling**

Recycling flux is increased with $\bar{n}_{\text{e}}$ provided for attached divertor condition. Due to a larger ion flux to the inner divertor, $\Phi_{\text{D}_{\text{o}}^{\text{in}}}$ is enhanced $2–3$ larger than $\Phi_{\text{D}_{\text{o}}^{\text{out}}}$ in the divertor-puff. Both recycling fluxes in the divertor and the in-out asymmetry are similar for the inner-side and both-sides pumping cases. Neutrals from the inner divertor to outer divertor through the bypass is small so that the in-out asymmetry does not change.

When plasma detachment occurs at the inner divertor separatrix, $\Phi_{\text{D}_{\text{o}}^{\text{div, out}}}$ increases with $\bar{n}_{\text{e}}$, and then outer divertor plasma is detached. For the detached divertor without causing x-point MARFE, in-out symmetries in recycling flux, neutral pressure and ion flux are observed. For the both-sides pumping, larger values of $\Phi_{\text{D}_{\text{o}}^{\text{in}}}$ and $\Phi_{\text{D}_{\text{o}}^{\text{div, out}}}$ are observed in the detached divertor, compared to those for the inner-side pumping and pump-off cases. Here, the radiation fractions in the divertor at the onset of the x-point MARFE are similar ($\sim70\%$ of $P_{\text{SOL}}$, where $P_{\text{SOL}} = P_{\text{abs}} - P_{\text{rad}}$) for the three cases. The both-sides pumping is favorable to increase the pumping rate and to improve impurity retention in the divertor.

**Summary**

With application of inner and outer divertor pumping, neutral flux from the inner divertor to outer divertor through under-dome is so small that the in-out asymmetry of the neutral recycling did not change. The plasma flow pattern did not change. However, in the partial-detached divertor, an increase in the plasma flow at the x-point was amplified by a reduction in the down-stream plasma pressure using pumping from the private flux region. Pumping at both-sides of the divertor was favorable for the partial-detached plasma, which can be maintained without appearance of x-point MARFE and with in-out symmetrical profiles of particle recycling.

**References**

6.4 Behaviour of neutral pressure in a detached divertor plasma

H. Tamai, N. Asakura, S. Higashijima, S. Sakurai, and H. Kubo

1. Introduction

Pumping slot from outer private side has been additionally installed in the end of 1998, in order to ensure the pumping efficiency during the divertor detachment, at which outer private region becomes high recycling. In modified divertor structure, divertor neutral pressure is investigated on the viewpoint of the change of in/out asymmetry during a high recycling divertor condition.\(^1\)

2. In/out asymmetry during divertor detachment and X-point MARFE

Destor neutral pressure is observed at the inner and outer private region (Fig.1) during density increase phase where divertor detachment or X-point MARFE occurs.

In Fig.2 neutral pressure at outer private \(P_{O_{\text{out}}}\), is plotted against that at inner private \(P_{O_{\text{in}}}\), during the density increase phase by main gas puff in the L-mode plasma of \(I_p=1.2\,\text{MA}, B_t=3.5\,\text{T}\), for various X-point height, \(\Delta X_p\), inner and outer private gap, \(\delta_{\text{in}}\), \(\delta_{\text{out}}\) respectively. The tendency of the change in \(P_{O_{\text{in}}}\), and \(P_{O_{\text{out}}}\), is similar to that in the previous pumping scheme from inner private,\(^2\) that is, \(P_{O_{\text{in}}}\) increases faster that \(P_{O_{\text{out}}}\) at the early phase of divertor detachment, in turn, \(P_{O_{\text{out}}}\) becomes larger at the later phase, and then X-point MARFE occurs. However, as clearly seen in the figure, detachment regime becomes wider in narrow gaps with pumping in present pumping scheme. The pressure ratio \(P_{O_{\text{out}}}/P_{O_{\text{in}}}\), in which the divertor detachment appears, extends between 0.4-1.3, on the contrary to narrower region between 0.6-0.8 for the inner-private pumping case.
The result suggests the possibility of feedback control of pressure ratio to sustain the divertor detachment. (see 5.2 in this article)

3. Change of in/out asymmetry in reversed $I_p/B_t$

The mechanism of the change in asymmetry at the onset of the X-point MARFE is not clear yet. Possible mechanism is presumed as the changes in in/out asymmetry of particle recycling and neutral pressure caused by the change of the ion flow in the scrape-off-layer\(^3\), and/or the change in the regions of ionisation and recombination\(^4\) at the onset of an X-point MARFE.

In/out asymmetry itself is observed even in the low recycling condition, and has been observed to be reversed when the direction of toroidal magnetic field and plasma current are reversed. In order to clarify the causality of in/out asymmetry, divertor neutral pressure in the reversed $I_p/B_t$ operation is observed for the first time.

In Fig.3 outside neutral pressure is plotted against inside neutral pressure during the density increase in L-mode plasma with $I_p=1.2\text{MA}$, $B_t=3.5\text{T}$, for normal and reversed $I_p/B_t$ cases. The behaviour of in/out asymmetry of divertor neutral pressure in the reversed $I_p/B_t$ shows the inverse feature respect to that in the normal one as follows: In/out asymmetry is reversed, that is, outer pressure is larger than inner pressure in the low recycling condition. Whereas, in the high recycling condition outer pressure turns to decrease when reaching an X-point MARFE.

Reversed behaviour in the reversed $I_p/B_t$ suggests that the change of the ion flow in the scrape-off-layer might be the most probable mechanism of in/out asymmetry.

4. Difference in pressure evolution depending on gas puff location

Feedback control of neutral pressure ratio in this campaign has not been successfully achieved the both sides detachment, since the outside pressure did not increase by the divertor gas puff (see 5.2 in this article). In order to clarify the reason and to lead the proper control, the response of the divertor neutral pressure to the divertor gas puff and main gas puff is compared.

Behaviour of divertor neutral pressure by the divertor gas puff is compared in low and high X-point height cases as shown in Fig.4. With increasing divertor gas puff up to $35\text{Pam}^3/\text{s}$ in deuterium discharge of $1.2\text{MA}/3.5\text{T}$, divertor detachment and X-point MARFE are attempted. In the high X-point height case of $\Delta X_p=11\text{cm}$, increase of outside neutral pressure prior to the
onset of an X-point MARFE is observed, which was commonly observed by main gas puff. However, the pressure ratio, $P_{out}^o/P_{in}^o$, is small (~0.6) and the range of divertor detachment is quite narrow, as compared to those by main gas puff. On the other hand, in the low X-point height case of $\Delta X_p = 6$cm, X-point MARFE is not observed by the same gas puff rate. However, outside pressure starts to decrease during the increased recycling phase.

By the divertor gas puff, dependence on X-point height is opposite to that by the main gas puff, in which the margin of pressure ratio without X-point MARFE extended in the low X-point height. Therefore, the pressure ratio shows no clear window for the X-point MARFE. Thus, for the sustainment of divertor detachment by divertor gas puff is not a proper method, instead, the pressure ratio must be controlled by a feedback loop with a main gas puff.

![Fig.4 Time evolution of line averaged density, radiation loss from X-point, divertor gas puff rate, inside and outside divertor neutral pressure, and $D_e$ emission from inside and outside divertor for the case of (a) high X-point height, and (b) low X-point height.](image)

Divertor neutral pressure was also observed in the hydrogen plasma during increasing density phase towards an X-point MARFE. Figure 5 shows the temporal evolution of line averaged density, divertor neutral pressure, and the divertor radiation power in the case of gas puff from outer baffle (QIII), NB-heated ($P_{NB} = 4.5$MW) L-mode, 1.2MA/3.5T with the X-point height of 6cm.

As clearly seen in the figure, outside neutral pressure once increased and turned to decrease. On the other hand, increase and decrease of inside neutral pressure occurred earlier and decayed with slower rate.

![Fig.5 Time evolution of line averaged density, divertor neutral pressure, and the divertor radiation power in the case of hydrogen gas puff from outer baffle](image)
From the measurement of the electron temperature in upper stream plasma using the movable X-point probe, it was confirmed that a partial detachment around the strike point occurred at the instance of the decrease of corresponding divertor neutral pressure. X-point MARFE occurred further later phase with the increase of the dome-top neutral pressure. Increase of divertor neutral pressure also coincides with the increase of local radiation power. Such phenomena as increase and decrease of the divertor neutral pressure at the corresponding target detachment and increase of local radiation power were commonly observed in the different location of gas puff from main, divertor, and outer baffle.

On the basis of those observations, a partial detachment would occur due to the increase of impurity radiation in a local area, then the decrease of particle flux from the upper stream would cause the corresponding neutral pressure. Sequential behaviour in divertor neutral pressure is considered as the feature of divertor detachment and also the precursor of X-point MARFE. Time delay between the start of detachment and the appearance of X-point MARFE is much longer in hydrogen plasma than that in deuterium plasma. Such an isotope effect should be precisely analysed.

References
5) N. Asakura, private communication.
6.5 Efficient helium exhaust in divertor-closure configuration

A. Sakasai, H. Takenaga, S. Higashijima

The W-shaped divertor of JT-60U was modified from inner-leg pumping to both-leg pumping. After the modification, the pumping rate was improved up to 4% with both-leg pumping in a divertor-closure configuration from 2% with inner-leg pumping at the high density region. By injecting a neutral beam of helium atoms as central fueling of helium into the ELMy H-mode plasmas (\(I_p = 1.4\) MA, \(B_t = 3.5\) T, \(P_{NB} = 16\) MW, \(V_P = 58\) m\(^3\)), helium exhaust has been studied in the W-shaped pumped divertor on JT-60U[1]. Neutral beams of 60 keV helium atoms were injected for 3 s into ELMy H-mode plasmas with the line-averaged electron density in the main plasma of \(n_e = 3.8 \times 10^{19}\) m\(^{-3}\), which corresponds to 0.57 of Greenwald density limit in the ELMy H-mode plasma. Deuterium gas of about 90 Pa·m\(^3\)/s is puffed to keep the electron density constant by a density feedback control. The He concentration reached 2% of the electron density in the main plasma and was kept constant for 2 s. This indicates that the He source rate (equivalent to 0.6 Pa·m\(^3\)/s) from the He beam injection is balanced by the exhaust rate with He pumping. In steady state, efficient helium exhaust was realized in a divertor-closure configuration with both-leg pumping in ELMy H-mode plasmas. A global particle confinement time of \(\tau_{He}^* = 0.36\) s and \(\tau_{He}^*/\tau_E = 2.8\) was achieved in attached plasmas. As a result, the He exhaust efficiency in the divertor-closure configuration was enhanced by 45% as compared to the one with the inner-leg pumping[2].

In the high \(X_p\) configuration with both-leg pumping, the helium exhaust efficiency and the pumping rate deteriorated because of the back-flow through the outer slot. The global particle confinement time \(\tau_{He}^*\) in the high \(X_p\) configuration became longer about two times as compared to the one in the divertor-closure configuration. The gap dependence on helium exhaust is almost consistent with the deuterium pumping rate.

Impurity shielding (friction force) with high-density divertor and plasma flow with puff and pump are essential to reduce impurity level in the core plasma. At the gas puff rate of 40 - 70 Pa·m\(^3\)/s, the carbon impurity level in the core plasmas was reduced from \(Z_{eff}=3-2.6\) with inner-leg pumping to \(Z_{eff}=2.6-2.3\) with both-leg pumping in ELMy H-mode discharges[3].

References
6.6 Helium removal from core plasma inside the ITB in reversed shear discharges

A. Sakasai, H. Takenaga, T. Fujita

Reversed shear mode with ITB is attractive because of its high performance and a large fraction of bootstrap currents in non-inductive current drive as one of advanced tokamak operation scenarios for future steady-state tokamak reactors, such as ITER. However, helium ash exhaust from the revered shear plasma is a matter of concern. In reversed shear mode of JT-60U, the electron density, electron and ion temperature in the central region are peaked and the confinement is remarkably enhanced inside the ITB, which is formed near the position of minimum q. The improvement of the He particle confinement inside the ITB is also remarkable. A previous study of He exhaust in reversed shear plasmas using He gas puff indicated that helium removal inside the ITB was 2 - 3 times as difficult as outside the ITB [1]. The modification to both-leg pumping enabled helium exhaust in reversed shear discharges in high triangularity configuration. By injecting neutral beams of 60 keV helium atoms into the core plasma inside ITB as central fueling of helium for the first time, helium exhaust characteristics in the reversed shear plasma at \( I_p = 1.7 \) MA, \( B_t = 3.7 \) T, \( \delta = 0.33 \) and \( P_{NB} = 5 \) MW were investigated. The residence time of He density (equivalent to the local \( \tau_{He}^* \)) at \( r/a < 0.4 \) inside the ITB is \( t = 3.7 \) s, while the residence time at \( r/a > 0.6 \) outside the ITB is \( t = 2.8 \) s. The He

\[\begin{align*}
\text{Fig. 1. (a) Time evolution of a reversed shear discharge with He-beam injection and (b) the measured He densities with helium pumping.}
\end{align*}\]
residence time inside the ITB is longer than that outside the ITB. The increase in helium density inside the ITB was faster than outside the ITB and helium density profile was gradually peaked.

In general, the He residence time of $\tau^*_{He}$ is defined as the global particle confinement time from the residence time of total helium particles. The local He residence time is almost the same at the central and peripheral region in ELMy H-modes. However, the local He residence time inside and outside the ITB in reversed shear plasmas is clearly different. A previous study of particle transport indicated that the particle diffusivity around the ITB was reduced by a factor of 5-6 compared with the inside and outside regions in the reversed shear [2]. A transport analysis is required to assess the helium exhaust properties inside ITB. If the local residence time of inside the ITB is assumed as $\tau^{ITB*}_{He}$, the ratio of $\tau^{ITB*}_{He}/\tau_E = 11$ ($\tau_E$ was contributed from the region of inside the ITB) was achieved.

When a partial collapse occurred, helium particles inside the ITB were expelled at the same time of the disappearance of the ITB as previously reported. The reduction of He concentration in the core plasma by periodic partial collapse is very effective. The reversed shear discharge with the L-mode edge by low-power NB heating is unfavorable for helium removal because of the low-density edge. Recently, the reversed shear discharge with H-mode edge by high-power NB heating of $P_{NB}=10$ MW was successfully obtained. In this discharge, the particle recycling flux in the divertor region was a factor of two as high as compared with the L-mode edge. Improvement of the He exhaust efficiency in the core plasma will be attempted with high recycling and ELMs.

References
6.7 Helium Atom and Ion Transport Processes in JT-60U Divertor

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The helium ion and atom transport in JT-60U W-shaped divertor are simulated with IMPMC code which based on the Monte-Carlo technique [1]. The most dominant process is the electron impact ionization, however, the effects of charge exchange processes between the helium ions and the helium atoms are also important in the temperature range of 0.1-10⁵ eV. In order to include the charge exchange processes, the helium density and temperature are necessary. Thus we calculate the helium distribution iteratively to be self-consistence. We also treat the helium reflection, desorption, and absorption on the graphite wall. In this calculation, the integrated helium density is assumed to be 10% of the integrated electron density.

The analyzed shot was an L-mode discharge and neutral particles were pumped from the inner divertor slot. In the inner divertor region, ne=2-4x10¹⁹ m⁻³ and Te=10-20 eV, and in the outer divertor region, ne=0.5-1x10¹⁹ m⁻³ and Te=30-40eV. Fig.1(a) and (b) shows the neutral helium atom density with pumping and without pumping case, respectively. This figure shows that the helium atoms are effectively pumped from the inner divertor region. Fig.1(c) is the same conditions as 1(b) but the charge exchange processes are not considered. The helium atom density in the fig.1(c) is larger than in the fig.1(b). This is explained as follows. The dominant helium atom sources are: (1) the divertor plates by desorption, (2) the divertor plates by reflection, (3) the charge exchange processes. The sources (2) and (3) produce the fast helium atoms, and those escape from the divertor region readily. Thus the charge exchange processes reduce the helium atom density. Fast helium atoms that have a long mean free path can easily penetrate into the core plasma. The penetration probability is 3.9% and 2.9% in the case of fig.1(b) and 1(c), respectively. We also calculate the helium ion and atom flux onto the divertor plates. The maxima of helium atom flux are found in the vicinity of the strike point. However, these maxima are not found when the charge exchange processes are not included. The charge exchange reactions between the helium ions and the helium atoms released from the divertor plates around the strike points make the helium atoms traveling to the divertor plates. Thus the maxima of the helium

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atom flux on the divertor plates appear around the strike points. The He$^+$ ion flux is larger than the He$^{2+}$ ion flux around the strike points: the ionization of He$^+$ ions is not important in the recycling processes.

Fig.1. Neutral helium atom density in the divertor region. (a) with charge exchange processes and with pumping, (b) with charge exchange processes and without pumping, (c) without charge exchange processes and without pumping.

Reference

6.8 Chemical sputtering in the divertor plasma

1. Introduction
Carbon materials have widely been used for divertor tiles, limiters and first walls in many tokamaks because of their high thermal shock resistance, low atomic number, high melting point and good thermal conductivity. Carbon, however, has the property of strong chemical reaction with hydrogen. The chemical sputtering process is more prominent than the physical one in cold and dense divertor plasmas assumed to be applied to fusion reactors. The chemical erosion is one of the most important factors to determine the lifetime of carbon divertor tiles. In terms of the carbon impurity release and transport, it is necessary to evaluate the carbon source quantitatively. It is therefore important to measure the chemical sputtering yield at the divertor plate exposed to the high particle flux in large tokamaks. In this work, chemical sputtering yield of \( C_2D_2 + C_2D_4 \) is reported.

2. Experiment
Spectroscopic measurements have been applied to the estimation of the molecular outflux from the divertor plate. Figure 1 shows a schematic diagram of the viewing codes of the diagnostics for this study. A 60-channel optical fiber array with an interference filter measures the CD spectral band \( \left( A^2\Delta - X^2\Pi \right) \) of the wavelength of the \( v=0-0 \) head is around 431.4nm) with a spatial resolution of about 1cm \(^1\). The CD spectral band intensities observed by the viewing codes from 33ch to 53ch are summed up to evaluate the total molecular outflux from the outer divertor plate. A 0.5m visible spectrometer \(^2\) with a grating of 2400g/mm measures \( C_2 \) spectral band (Swan system \( A^2\Pi_g - X^2\Pi_u \), the wavelength of the \( v=0-0 \) head is around 516.5nm) with spatial resolutions of 1cm and partly 2cm. It has 16 viewing codes selected between 33ch and 53ch of another 60-channel fiber array. The \( C_2 \) spectral band intensities observed with the 16 viewing codes are also summed up. The 60-channel fiber array for the CD spectral band are calibrated using the calibrated spectrometer by two similar discharges described below. Consequently, CD and \( C_2 \) spectral bands intensities are measured simultaneously by the filter array and the spectrometer. Figure 1 also shows other diagnostics and a typical plasma configuration in the divertor region of this experiment. The total ion flux to the outer divertor plate is measured by 9 Langmuir probes on the divertor plate and the surface temperature of divertor tiles is measured by IRTV.

L-mode discharges have been analyzed to measure the chemical sputtering yields. Figure 2 shows typical time traces in this experiment. The discharges were carried out
in deuterium at 1.2MA and 3.5-3.8T with additional neutral beam heating powers of 4-6MW. The line averaged central plasma density increases up to about 2.5x10^{19}/m^3 by gas fuelling. The vessel is baked at the temperature of 290°C during the experiments. The surface temperature of divertor tiles near the outer strike point is almost constant around 300°C during the discharges. The ion saturation current starts to decrease when the plasma becomes cold and dense enough to detach with the increase of the main plasma density. It is difficult to measure the accurate ion flux to the divertor plates by Langmuir probes in detached plasma conditions. Therefore, the estimation of the yields is restricted to attached conditions.

3. Analysis

The CD$_4$ outflux is not proportional to the observed intensity of the CD spectral band. The CD spectral bands are emitted from CD radicals originating not only from CD$_4$ but also from C$_2$D$_2$, C$_3$D$_4$ and other heavier hydrocarbon molecules at the end of a long break-up chain. Therefore, the observed CD spectral band intensity is expressed as follows,

$$I_{CD_{\text{observed}}} = I_{CD_{\text{CD}}^4} + I_{CD_{\text{C2D4}}} + I_{CD_{\text{C3D4}}} + \ldots$$

According to the laboratory experiment \cite{3}, the chemical sputtering yields of C$_2$D$_2$ and C$_3$D$_4$ are almost comparable and the yields of C$_2$D$_4$ and other heavier hydrocarbons are about one order of magnitude smaller than those of CD$_4$, C$_2$D$_2$ and C$_3$D$_4$. Therefore only CD$_4$, C$_2$D$_2$ and C$_3$D$_4$ are taken into account to evaluate the yields.

$$I_{CD_{\text{observed}}} = \frac{D/XB_{(CD4)}}{\text{CD}} \cdot \Gamma_{CD4} + \frac{D/XB_{(C2D4)}}{\text{CD}} \cdot \Gamma_{C2D4} + \frac{D/XB_{(C3D4)}}{\text{CD}} \cdot \Gamma_{C3D4} + \ldots$$

where $D/XB_{(CD4)}^\text{CD}$ is the number of CD photons emitted per a dissociating CD4 molecule and $D/XB_{(C2D2,4)}^\text{CD}$ is the number of CD photons per either a C$_2$D$_2$ or C$_2$D$_4$ molecule on the assumption that the same number of C$_2$D$_2$ and C$_3$D$_4$ are sputtered by the chemical reactions. $\Gamma_{\text{ion}}$ is a total ion flux to the divertor plate measured by Langmuir probes. By arguments similar to the CD emission, the following equation about the C$_2$ emission is obtained.

$$I_{C2_{\text{observed}}} = I_{C2_{\text{C2D2}}} + I_{C2_{\text{C3D4}}} + \ldots = \frac{D/XB_{(C2D2,4)}}{\text{C2}} \cdot \Gamma_{C2D2} = \frac{D/XB_{(C2D2,4)}}{\text{C2}} \cdot \Gamma_{C2D2}$$

where $D/XB_{(C2D2,4)}^\text{C2}$ is the number of C$_2$ photons per a C$_2$D$_2$ or C$_2$D$_4$ molecule.

The D/XB values used in this work are reported in the reference paper \cite{4}, which also gives the intensity definitions of the CD and C$_2$ spectral bands as shown in Figure 3.

4. Results

Figure 3 shows the spectral bands of CD and C$_2$ ($\Delta v=0,1$) observed along the
viewing code (52ch) closest to the outer divertor plate. The CD spectral band of $A^2\Delta-X^2\Pi$ ($\nu=0-0$) system in Figure 3 (a) shows a well-known structure and is often used to evaluate CD$_4$ outflux. Another vibrational transition ($\nu=2-2$) of the same electronic transition has a line like feature around 432nm. The C$_2$ spectral bands around 516.5nm and 563.5nm in Figure 3(b) and 3(c) respectively, are identified as Swan system ($A^2\Pi_u-X^2\Pi_u, \Delta \nu=0,1$). In particular, the sequence with $\Delta \nu=1$ has such an obvious structure that one can identify it with ease. The sequence with $\Delta \nu=0$ however, is preferred for this study because the spectral band intensity of the $\nu=0-0$ head is factors of magnitude stronger than that of the $\nu=0-1$ one.

Figure 4 summarizes the chemical sputtering yield measured at the outer divertor plate. When the electron temperature is regarded as 20eV, D/XB$_{\text{CD4}}^{\text{CD}}$~100, D/XB$_{(\text{C2D2A})}^{\text{C2}}$~200 and D/XB$_{(\text{C2D2A})}^{\text{C2}}$~435. As a result, both yields of CD$_4$ and C$_2$D$_2$ + C$_3$D$_4$ are around 1% to 2%. This result reveals C$_2$D$_2$ and C$_2$D$_4$ are also sputtered by chemical reaction to an extent comparable to CD$_4$. Hence C$_3$ hydrocarbons can play some roles in atomic and molecular processes in divertor plasmas, for example, a MARFE phenomenon.

References

2) T. Nakano, et al., Section 8.7 in this report.

Fig.1 Diagnostics and plasma configuration in the divertor region. CD and C$_2$ emissions are integrated between 22 and 16 viewing codes respectively. The spatial resolution is about 1-2cm. The ion flux to the plate is measured by Langmuir probes. The surface temperature of tiles is measured by IRTV.
Fig. 2 A typical waveform of line averaged electron density, ion flux measured with Langmuir probe, CD and C2 band intensity. Divertor plasmas in attached conditions and Te~20eV are analyzed (from 7.7s to 8.5s in this shot).

Fig. 3 a) CD spectral band b) C2 spectral band (v=0-0) c) C2 spectral band (v=0-1,1,2,...) observed with the spectrometer. In a) and b), the hatched areas define the integrated wavelength range (1.5nm) according to [4].

Fig. 4 Chemical sputtering yield measured in JT-60U divertor plasma. Open squares and closed diamonds represent CD₄ and C₂D₆ + C₃D₄ sputtering yield respectively. Open circles indicate CD₄ sputtering yield without any consideration of other heavier hydrocarbons as a reference.
6.9 Effect of B₄C tiles in gas puffing ports on carbon impurity

S. Higashijima and K. Itami

1. Introduction

Carbon impurities in the main plasma can be reduced by the effect of puff and pump after the divertor modification, but in some discharges carbon impurity in the main plasma is not reduced when the strong main puff is applied. We thought that neutral particles added by the strong gas puff generated chemically hydrocarbon around the gas puffing port. So we replaced normal CFC tiles with B₄C-coated CFC tiles around the gas puffing port, because chemical sputtering yield of B₄C-coated CFC tile is smaller than that of CFC tile. The effect of B₄C tiles in gas puffing ports on carbon impurity is estimated in this section.

2. Carbon dust around the gas puffing port after the divertor modification

Divertor pumping and gas puffing system were installed in the divertor modification. Three puffing inlets were installed at the top of main plasma (main gas puffing). Particles in the vacuum vessel are exhausted through the toroidally-continuous pumping aperture in the inner leg side in the private flux region. And then, particles are conducted under the divertor dome and outer baffles and led to three cryo-pumps.

After about five months of operation, the vacuum vessel was opened for the maintenance. It was found that carbon dusts adhered to the carbon tiles around a gas puffing port which was applied most frequently and strongly. Since the adhering area was a concentric circle around the port, it seemed that it had no relation to magnetic structure. We thought as follows; Neutral particles added by puffing generate chemically hydrocarbons around the port. The hydrocarbons generate carbon dusts. And then the dusts stick on the tiles. Therefore, we tried to decrease hydrocarbon sputtered chemically because it was impossible to obtain high density plasmas without the strong gas puffing.

3. Characteristics of chemical sputtering in B₄C-coated CFC tiles

In the open divertor configuration of JT-60U, the material near the inner strike point was the carbon fiber composite (CFC), and that near the outer strike point was the B₄C-coated CFC with the original thickness of B₄C layer of ~300 μm. Although these B₄C-coated CFC tiles have been eroded through previous four years of operation, chemical sputtering yield of methane of normal CFC tiles was about three times larger than that of B₄C-coated CFC tiles. This result shows B₄C layer suppresses the chemical sputtering of methane even if the thickness of B₄C layer were thin. And it is also reported that boron can suppress chemical sputtering in the laboratory experiments. So we experimentally installed B₄C-coated CFC tiles around one of three gas puffing ports.
The characteristics of B₄C-coated CFC tiles are as follows; The size of the tiles is about 5 cm height x 10 cm wide x 2 cm thickness. The number of the tiles is twenty. The thickness of B₄C layer is ~300 μm.

4. Experiments

The aim of this experiment is to estimate the effect of B₄C tiles in gas puffing port on carbon impurity. We compare carbon contents between the gas puffing from the port (U2 port of P-14 section) of B₄C tiles and the puffing from the port (U2 port of P-11 section) of normal tiles in the same discharge condition.

The plasmas are ELMy H-mode (NB power ~15 MW) because the strong gas puffing is applied. Plasma configuration, CXRS diagnostics and the location of gas puffing are shown in figure 1. The discharge conditions are as follows; Iₑ = 1.45 MA, Bᵣ = 3.0 T, plasma volume ~80 m³, height of x-point ~0.07 m from the top of divertor dome, a distance (gap-in) between the inner leg of the separatrix and the divertor dome is ~1 cm, a distance (gap-out) between the outer leg of the separatrix and the divertor dome is ~5 cm, the working gas is deuterium, and pumping is on.

Figure 2 shows the time history of line-averaged electron density \( \bar{n}_e \), injected NBI power \( P_{NB} \), gas puffing rate, \( Z_{eff} \), C VI intensity with charge exchange recombination spectroscopy at 10 ch shown in fig. 1 \( (I_{CXRS(10ch)}) \) and averaged intensities of \( D_α \) C II, CD-band and C IV in the divertor \( (D_α^{dv}, C II^{dv}, CD-band^{dv}, C IV^{dv}) \). The solid line is a case of gas puffing from B₄C port and the dotted line is a case of gas puffing from CFC port in fig. 2. The \( \bar{n}_e \) is gradually increased up to the density of the x-point MARFE onset by the feedback control. Shadow area after ~8.5s shows that the x-point MARFE appears.

It is found that \( \bar{n}_e \) is well-controlled by the feedback control and the gas puffing rate is the same in the cases of gas puffing from B₄C port and from CFC port, and that injected NBI power is almost the same in both cases. According to \( Z_{eff} \) and CXRS measurements, the content of carbon impurity in the main plasmas has no difference in both cases. Moreover, there is no large difference in divertor intensities in both cases.

5. Discussion

In this experiment, there is no difference in carbon impurity in the main plasma. However, in an experiment of pulsed methane gas puffing from the main chamber (puffing rate: ~8.5 Pam³/s, duration: 0.5s) in JT-60U, carbon impurity in the main plasma did not change, but carbon impurity in the divertor increased. It suggests that the shielding of neutrals and impurities is strong in the SOL around the main plasma. Since B₄C tiles suppress the formation of hydrocarbon, we think that methane around the main plasma must decrease in the gas puffing from B₄C port in comparison with the gas puffing from normal port. So we are planning to measure directly the emission of CD-band in both the B₄C-coated CFC port and normal CFC
port.

Acknowledgment

The authors would like to thank Mr. T. Sasajima, Mr. K. Masaki, Mr. Kodama and Dr. N. Hosogane for experimentally installing the B₄C-coated CFC tiles.

Reference

Fig. 1 Plasma configuration and CXRS diagnostics

Fig. 2 Time history of discharges. solid line: gas puffing from the port of B₄C tiles, dotted line: gas puffing from the port of normal CFC tiles.
6.10 Wall conditioning to control impurity and hydrogen isotope recycling in JT-60U

S. Higashijima and the JT-60 Team

To obtain the high performance plasma, it is necessary to control impurity and hydrogen isotope recycling. For this purpose, various methods of wall conditioning have been adopted in JT-60U. The experience of wall conditioning in JT-60U is summarized.

Wall conditionings of JT-60U are baking of the vacuum vessel, helium Taylor discharge cleaning (TDC), helium glow discharge cleaning (GDC), tokamak discharges and boronization.

Baking of the vacuum vessel is effective in decreasing water generated from the vacuum vessel, and this is the first conditioning after the ventilation. The baking temperature is 300°C and is usually adopted in JT-60U. He-TDC is the 2nd conditioning after the ventilation. The aim of this conditioning is to decrease the recycling and impurities. This is also applied for ~7 minutes after the disruption to make the next discharge. He-GDC is the 3rd conditioning after the ventilation. In an experimental period, He-GDC is executed overnight after the experiment. But it is rare that He-GDC is applied between the discharges in JT-60U because the gate valves of NBI become high temperature just after the injection and cannot be closed. When tokamak discharges are carried out, we often use the discharges itself for the conditioning. Tokamak discharges are effective in cleaning the position where the plasma-wall interaction is strong, e.g. the divertor plates and the limiter tiles. When the quantity of oxygen impurity in ohmic plasmas is decreased to less than 1%, we finally execute boronization using the decaborane in wall temperature of 250°C that is supported by glow discharge of helium or the mixture of helium and deuterium. Boronization is effective in reducing the recycling and oxygen impurity. Depending on the experimental plan, boronization is done once or twice in a year. In JT-60U the procedure of these wall conditionings is decided by the experimental criterion.

By using the above-mentioned wall conditioning, tokamak discharges can be obtained within a week after the ventilation. In the experimental campaign these conditionings can also make such a good wall condition that recycled particles in ohmic plasmas are so low as to correspond to injected NB particles. Consequently, an equivalent $Q_{\text{DT}}$ up to 1.25 can be transiently achieved in reversed shear plasmas because of both the optimization of wall conditioning and the improved confinement.

The present problem of the wall conditioning of JT-60U is to reduce carbon impurity that is the material of the first wall. Typically, $Z_{\text{eff}}$ value is ≥ 3.2 and $n_c/n_e$ ≥ 4% in the reversed shear plasmas. The wall conditioning of low Z material coating, e.g. lithium, might be necessary to decrease carbon impurity.

Reference
7. Radiative Divertor and High Density Plasma

7.1 Improved confinement and radiation enhancement of Ar seeded high density ELMy H-mode discharges

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1. Introduction

Improved confinement $HH=1.0$, low dilution $Z_{eff}<1.8$ and radiation enhancement for reduction of heat load on target plates should be satisfied simultaneously at 85% of Greenwald limit density ($n^{GW}$) for ITER-FEAT. These conditions have not been demonstrated in large tokamaks such as JT-60U$^1$ and JET$^2$, since energy confinement degrades at high density region. The effect of Ar seeding for radiation enhancement and confinement improvement has been investigated in ELMy H-mode. The radiated power inside the separatrix remains $\leq30\%$ of the absorbed heating power, although large radiation fraction ($P_{rad,total}/P_{net,\geq70\%}$) and detached divertor plasma are obtained at high density ($\geq0.6n^{GW}$) by Ar seeding. Higher temperature and improved energy confinement ($H^{89} \geq1.4$) are maintained up to $0.7n^{GW}$. On the contrary, low edge temperature in strongly $D_2$ fueled case degrades core temperature and thermal component of stored energy.

2. Experimental setup

Radiation enhancement experiments have been carried out by applying weak Ar seeding and moderate $D_2$ fueling to ELMy H-mode plasmas with neutral beam power $P_{NB}=17-21$ MW, plasma current $I_p=1.2-1.7$ MA, toroidal magnetic field $B_T=2.5-3.5$ T and high triangularity $\delta=0.35$. Ar gas was seeded to the main plasma from outer side and $D_2$ gas was fueled from top fueling ports. An outer pumping slot is opened between a dome and an outer target plate in 1999. Therefore, the outer pumping can exhaust particles through outer divertor for high triangularity configuration, in which the inner strike point is too far from the inner slot for effective pumping.

3. Comparison between Ar seeding and strongly $D_2$ fueling

Typical wave forms of Ar seeded case and the case of strongly $D_2$ fueling are compared in Fig. 1. Target plasma conditions were $P_{NB}=17$ MW, $1.2$ MA, $B_T=2.5$ T, $q_95 \sim 3.3$ ($q_{eff} \sim 4.0$) and confinement improvement $H^{89}=1.6-1.8$ at the density of $\sim0.45n^{GW}$. Ar gas was seeded at the density of $0.45n^{GW}$, just before significant degradation of energy confinement.

The top and second boxes of Fig. 1 show the fraction of the total radiated power $P_{rad,total}$ and the radiated power from the main plasma $P_{rad,main}$ to the net heating power $P_{net}$. $P_{rad,main}$
includes radiation losses inside the separatrix and the SOL around the main plasma. In the Ar seeded case, the fraction of $P_{\text{rad}}^{\text{main}}$ enhanced from ~20% to ~40%, and then it exceeded 50% and caused radiative collapse after 8.5s. The larger stored energy $W_{\text{dia}>2MJ}$ was maintained up to ~0.7$n^{\text{GW}}$ by Ar seeding. The effect on radiation, core confinement and edge pedestal are described in the next section.

4. Radiation enhancement and effect on global energy confinement and pedestal

The H factor and $P_{\text{rad}}^{\text{total}}/P_{\text{net}}$ in low toroidal field ($B_T=2.5\text{T}$) cases are plotted against the normalized density in Figs. 2. Improved confinement ($H^{89} 1.4$) is maintained at high density (0.7$n^{\text{GW}}$), although $P_{\text{rad}}^{\text{total}}/P_{\text{net}}$ reaches ~80% of the heating power.

Radial profiles of electron density, ion and electron temperature and radiated power density at 0.6$n^{\text{GW}}$ in Ar seeded and strongly D$_2$ fueled cases are shown in Figs. 3. Electron density profiles are flat and similar in both cases. However, the ion and electron temperature profiles in the Ar seeded case are higher than those in the strongly D$_2$ fueled case. Since stiff temperature profiles, which are the evidence for an edge-core relationship, have been also observed in only D$_2$ fueled cases\textsuperscript{3}, it has not been clear that high center temperature in the Ar seeded case is caused by confinement improvement at the core region due to Ar effect such as RI-mode.

Ar seeding enhances edge (0.7a—a) radiation more than several times. However, the total radiated power inside the separatrix remains 30% of the absorbed heating power. Therefore, the radiative mantle does not degrade energy confinement significantly.

Ion and electron temperature and electron density at the edge pedestal are plotted against normalized density in Fig. 4(a). The Ar seeded case shown by closed symbols indicates smaller degradation of edge temperature. On the other hand, not only the height but also the width of pedestal decreased in the strongly D$_2$ fueled case. The thermal component of stored energy has been evaluated from the result of transport analysis which agrees with measured temperature and density profiles and neutron yield. Improvement in the total stored energy is mainly caused by the improvement of thermal energy confinement as shown in Fig. 4(b).

On the other hand, radiation losses in the SOL and the divertor are dominantly enhanced in high toroidal magnetic field cases of $B_T=3.5\text{T}$. However, the electron density and the radiated power inside the separatrix hardly increase in high $B_T$ cases. Since the high toroidal field and safety factor make the SOL thick and high temperature, the fueling efficiency for D$_2$ and Ar gases are probably reduced. Nevertheless, ELMs and the edge pedestal lose when Ar penetrates into the main plasma and enhances radiated power inside the separatrix. Therefore, Ar concentration and radiated power cannot be controlled in high $B_T$ cases.

5. Conclusion

The effect of Ar seeding to the main plasma has been investigated in ELMy H-mode plasmas with high triangularity. The radiation fraction of ~80%, the divertor detachment, improved
confinement $H^{89} = 1.4$ and the clear edge pedestal are simultaneously maintained up to $0.7n^{GW}$ by Ar seeding.

The radiation losses are enhanced at not only in the divertor but also in the edge plasma (0.7a) and the SOL. Nevertheless, the total radiated power inside the separatrix remains 30% of the total absorbed heating power before the confinement improvement loses at $\geq 0.7n^{GW}$.

Ar seeding maintains high plasma temperature in the main plasma up to $\sim 0.7n^{GW}$, but clearly peaked density profile such as RI-mode has not been observed yet. It has not been clear that high center temperature in the Ar seeded case is caused by confinement improvement at the core region due to seeded Ar. On the contrary, low and narrow edge pedestal in strongly D$_2$ fueled case degrades core temperature and thermal component of stored energy. Transport analysis including argon seeding effect on ion temperature gradient (ITG) mode is in progress under the collaboration with Princeton Plasma Physics Laboratory.

Improvement of energy confinement time ($\sim 30\%$) and temperature ($\sim 40\%$) by Ar seeding has an advantage, although peaked density profile is not obtained and deuterium density decreases $\sim 15\%$ in comparison with that in strongly D$_2$ fueled cases. Therefore, the optimization of energy confinement and dilution should be investigated now.

Reference

3) Urano H et al.: Section 3.4 in this review.
Fig. 1 Typical wave forms of Ar seeded and strongly D₂ fueled cases. Thick and thin lines show the Ar seeded case and the strongly D₂ fueling case, respectively. Radiation fractions are shown in top and second boxes. The line averaged electron density normalized by Greenwald limit and the stored energy W²ₐ are shown in the fourth and fifth boxes, respectively.

Fig. 2. (a) H-factor and (b) the ratio of total radiated power to heating power as functions of line averaged electron density normalized by the Greenwald limit in ELMy H-mode plasmas (Iᵰ = 1.2 MA, Bᵢ = 2.5 T, and Pᴺᴮ = 17–20 MW). Divertor plasma detachment is shown as closed symbols.

Fig. 3 ¹⁾ Profiles of (a) electron density, (b) electron temperature, (c) ion temperature, and (d) radiated power density in ELMy H-mode (nₑ/nₑGW = 0.6 and Pᴺᴮ = 16 MW). Solid and broken lines indicate the cases Ar seeded and strongly D₂ fueled cases, respectively.

Fig. 4 ²⁾ (a) Ion and electron temperature and electron density, and (b) The total and thermal components of stored energy vs. line averaged density normalized by Greenwald limit.
7.2 ELMy H-mode hydrogen plasmas seeded with Ar and Kr

H. Kubo, S. Higashijima, K. Hill, S. Sakurai, and N. Asakura

1. Introduction

In JT-60U, confinement improvement and radiation loss enhancement has been obtained by Ar seeding in ELMy H-mode deuterium discharges [1]. Similar results have been also obtained by Kr and Xe seeding in TFTR [2]. These heavy impurities have two advantages over Ar: (1) they are still good radiators at reactor temperature range above 15 keV, (2) it results in lower dilution, because the radiation loss coefficient is approximately proportional to the cube of the charge number. In order to study the effect of heavy impurity on confinement and radiation loss, ELMy H-mode hydrogen plasmas were seeded with Ar and Kr.

2. Experimental setup

ELMy H-mode plasmas with the configuration (volume: 64 m$^3$, elongation: 1.28, triangularity: 0.39) shown in Fig. 1 were seeded with Ar and Kr. Hydrogen gas was puffed from the top, and the impurity gases were puffed from the lower outside. Neutral particles were pumped from the private flux region through the two slots between the divertor dome and the divertor target plates. In the Ar case, the sum of radiation loss power measured with the bolometers viewing the edge region was controlled with a feedback technique by Ar puffing, since the radiation losses are dominantly enhanced in the edge region. In the Kr case, the sum of radiation loss power measured with the bolometers viewing the core region was controlled by puffing Kr gas mixed with H$_2$ gas, since the enhancement of the radiation losses in the core region can be attributed to radiation losses by Kr. The mixing ratio was 1:1; the ratio of the injection rate was expected to be 1:6.3, since the injection rate is inversely proportional to the mass. The plasma current was 1.2 MA, and the toroidal magnetic filed was 2.5 T. Conditions in the discharges were almost similar to those in the ELMy H-mode deuterium discharges with Ar seeding presented in Ref. 1.

3. Results and discussion

Waveforms of an ELMy H-mode plasma seeded with Kr are shown in Fig. 2. The NBI power is 13 MW, and hydrogen gas is puffed at a constant rate. Although the measured core radiation loss power waves, it is controlled around the reference. Smoother control may be possible by further adjustment of the control gains.

The H-factor and radiation loss fraction are shown as functions of the electron density in Fig. 3. Confinement improvement is degraded, as the density increases. In low-density range, the H-factor is higher in the no impurity case than in the other cases. However, the H-factor in the Ar case is about 10% higher than that in the no impurity case in the density range above 70% of the Greenwald density limit. Confinement improvement is lower in the
Kr case than in the other cases. The total radiation loss power is enhanced up to around 80% of the net heating power by the impurity seeding, as seen in Fig. 3 (b). As shown in Fig. 3 (c) and (d), enhancement of radiation loss is dominant in the main plasmas in the impurity seeding cases, particularly in the Kr case. In the all three cases, the divertor radiation loss fraction is around 50% in high-density range. Radiation loss in the divertor plasmas is not enhanced by the impurity seeding in the high-density range.

The confinement improvement by Ar seeding in high-density range is similar to that observed in the deuterium discharges [1]. In the Kr case, confinement degradation is observed when the radiation loss power is high. The tendency can be seen in Fig. 2. The stored energy oscillates off phase with the radiation loss power. In the Kr case, the electron temperature is around 2 keV at the plasma center, when the electron density is 60% of the Greenwald density limit. Then, it is calculated with an impurity transport code (MIST) [3] that the dominant Kr ion is Kr$^{+26}$ at the plasma center and the profile of the radiation loss power density is almost flat in the main plasma. Therefore, the confinement degradation can be attributed to the radiation loss due to Kr. In TFTR, confinement improvement by Kr seeding was observed in super shots, where the electron temperature was 5 - 9 keV at the plasma center. It is probably necessary to increase the electron temperature in order to reduce the radiation loss power in the core plasma and improve the confinement. In order to obtain high electron temperature, high power heating is necessary. In JT-60, while the maximum heating power of the hydrogen NB is around 13 MW, that of the deuterium NB is around 22 MW. Using the high power heating with the deuterium NB, the confinement improvement observed in TFTR may be obtained. Further experiment with high power heating is necessary to investigate effect of Kr seeding in reactor regime plasmas with high temperature.

4. Summary

Hydrogen ELMy H-mode plasmas were seeded with Ar and Kr using feedback control of radiation loss power. By the Ar seeding, the energy confinement was improved by about 10% in the density range above 70% of the Greenwald density limit, and the total radiation loss power was enhanced up to 80% of the net heating power. On the other hand, by the Kr seeding, the confinement was degraded due to enhancement of radiation losses. Further experiment with high power heating is necessary to investigate effect of Kr seeding in reactor regime plasmas with high temperature.

References
[1] Sakurai S. et al., in this report.
Fig. 1. Schematic diagram of JT-60U cross section for explanation of radiation loss power control by a feedback technique using impurity gas puffing and divertor pumping.

Fig. 2. Waveforms of an ELMy H-mode plasma seeded with Kr. The first row shows the NBI power and the stored energy, and the second row shows the line-averaged electron density and the puff rate of hydrogen gas. The measured and refereed core radiation losses are shown in the third, and the puff rate of the Kr + H₂ gas and the intensity of Kr XXVI 17.9 nm are shown in the forth. The bottom row shows the radiation losses.

Fig. 3. (a) H-factor and radiation loss fractions (b): total, (c): main and (d): divertor against the electron density normalized by the Greenwald density limit. The Greenwald density limit is 5.2 x 10¹⁹ m⁻³.
7.3 Reduction of gas puff amount with near-limiter configuration for the Ar seeding radiative edge plasma
N. Asakura, S. Sakurai, H. Kubo, H. Tamai and S. Konoshima

Maintaining good energy confinement at high density is the most important issue for a tokamak reactor. However, degradation of the enhancement factor of the energy confinement is observed in the high density operations, where deuterium gas puff rate is required. One of the candidates is due to an increase in the neutral density at the plasma edge, and pedestal width of the plasma pressure profile is decreased with enhancing the neutral recycling flux[1,2].

Recycling impurity seeding such as neon (Ne) and argon (Ar) has been found as enhancement of the radiation fraction at the plasma edge and improvement of the energy confinement. At the same time, deuterium gas puff rate to obtain high density plasma was reduced and the relatively good energy confinement was observed, in particular, for the Ar seeding case. Understanding the improvement of the energy confinement in the high density regime is very important for the large tokamak experiments, and the knowledge is directly extended to the reactor operation.

Gas puffing in near-limiter plasma was proposed in order to reduce the deuterium gas-puff rate during increasing the plasma density. Here, limited plasma discharge during long NBI is prohibited in JT-60U, while plasma detachment in the divertor occurs at relatively low density for the case of OH plasma. It may be also useful to arrange future experiment plans of the radiative improved mode, which has been obtained in the limited plasma discharges of TEXTOR, DIII-D and JET. In 1999, (1) gas puff rate during increasing the plasma density, (2) Ar-gas feedback control were investigated as first step after the adjustment of the plasma configuration. Other parameters such as NB units, pumping rate and strike point locations, Ar gas puff rate, \( I_p \) and \( B_t \) were fixed. Systematic scan of the parameters were postponed.

Plasma configurations

Plasma configurations of near-limiter, Fig. 1(a), and divertor, Fig. 1(b), were used during increasing plasma density with low \( P_{NB} = 4.3 \text{ MW} \), and during Ar gas puff with high \( P_{NB} = 17 \text{ MW} \), respectively. Here, for the case of near-limiter configuration, the field line interacted at the inner upper first wall corresponds to that mapped to midplane radius of 1 cm outer the separatrix. Larger fueling efficiency into the main plasma was expected for the near-limiter plasma compared to the divertor plasma. Distances between the outer stripe point and the pumping slot were 2 cm (near-limiter) and less than 1 cm (divertor). The pumping rate of the deuterium gas was increased for the latter case, which might be beneficial for high density operation.

Gas puff rate during increasing density

Fueling efficiency was compared in Fig. 2 for the two cases of the near-limiter configuration: (1) ELMy (type III) H-mode with low \( B_t = 2.4 \text{ T} \), and (2) L-mode plasma with high \( B_t = 3.5 \text{ T} \). Plasma current of \( I_p = 1.2 \text{ MA} \) was the same. L-mode plasma with higher \( B_t = 3.5 \text{ T} \) and \( I_p = 1.6 \text{ MA} \) in the divertor configuration (the same safety factor as (1)) is also shown. \( P_{NB} = 4.3 \text{ MW} \) was the same for the three cases. Plasma densities of the target OH plasma of \( \bar{n}_e = (1.1 - 1.4) \times 10^{19} \text{ m}^{-3} \), were similar, and they were increased to \((2.5 - 2.8) \times 10^{19} \text{ m}^{-3}\), which correspond to 0.50–0.55 of Greenwald density fraction, \( \bar{n}_e/n^G \), for \( I_p = 1.2 \text{ MA} \). Horizontal axis shows the value integrating the gas puff rate using the divertor pumping, and the these experiments were done in the same day. However, absolute values changed depending on wall pumping. For the case of (1), \( \bar{n}_e \) is increased during NBI due to H-mode transition (good particle confinement), but \( \bar{n}_e \) becomes saturated during ELMs. For the case of (2), smaller fueling efficiency is
observed since SOL width for the high safety factor (2) is larger than that for lower one (1). However, the fueling efficiencies for (1) and (2) are about 3 and 2 times better than that for the divertor plasma. As a result, smaller deuterium gas puff for the lower $B_t$ case to obtain the relatively high $\bar{n}_e$ under the attached divertor condition (i.e. $\bar{n}_e/n^{Gr} = 0.50 - 0.55$). Low $B_t = 2.4$ T plasma was used for the energy confinement studies.

Preliminary results

Ar gas of 1 Pam$^3$/s was injected during changing plasma configuration from the near-limiter to the divertor (0.3 s), where the plasma center and X-point height were simply shifted about 5 cm, 1 cm downward, respectively, as shown in Fig. 1. Then, during high $P_{NB}$ injection, Ar-gas puff rate was adjusted to follow programmed edge radiation power using feedback control as shown in Fig. 3 (4th column). Feedback control of deuterium gas puff was also used to increase $\bar{n}_e$ slowly, where the gas puff rate (maximum rate of less than 10 Pam$^3$/s) was smaller than that in the near-limiter phase. The particle recycling flux in the divertor was not enhanced (2nd column), and the stored energy was maintained at constant level until 9.4 s (2 NB units was broken down and recovered). Radiation power fraction from the main plasma edge was, in particular, increased after Ar-gas injection (3rd column). Divertor radiation power fraction was also increased to 36% of $P_{abs}$ at 8.4 s, where divertor plasma both at the inner and outer strike points was detached since $P_{sol}/P_{abs} = 1 - P_{main}/P_{abs} \sim 0.6$. Greenwald density fraction was 0.65 at 9.4 s.

After 9.4 s, the plasma stored energy was decreased and was not recovered although total $P_{NB}$ was recovered. In particular, ion temperature (and electron temperature) in the plasma center region was decreased, and the energy confinement was degraded.

Figure 4 shows the enhancement factors of the energy confinement (H-factor) and total radiation fractions for the similar discharges ($P^{tot}_{rad}/P_{abs}$). Gas puff duration in the near-limiter configuration for 34348 and 35349 was larger than 34344, and breakdowns of only 1 NB unit were observed. Relatively high H-factors of 1.4 - 1.45 was maintained during 0.5 - 1.5 s with large radiation fraction of $\sim 0.8$, where Greenwald density fraction was 0.7. This obtained parameters were reproducible for Ar gas seeding cases (normal start-up with divertor). Exploring the higher radiation and density operations was not studied further more since an improvement of the confinement at high density nor density peaking (which are synopsis of so-call RI mode) was not observed in the first trials.

It should be noted that week internal transport barrier was observed in $T_e$, $T_i$ and $n_e$ profiles during early period of high $P_{NB}$ in a few discharges with the Ar gas puff as shown in Fig. 5, and that, at the same time, amplitude of type I ELM was reduced. Although the ITB was not observed at high density ($\bar{n}_e/n^{Gr} \geq 0.65$), it may be important to find out the reason to form the ITB plasma for the further study of RI-mode trials.

Summary

Degradation of the energy confinement at high density has been often coincident with large deuterium gas puff rate or amount and enhancement of the edge particle recycling. Gas puffing in near-limiter plasma was proposed in order to reduce deuterium gas-puff rate during increasing the plasma density. This may affect the later energy confinement during high power injection with Ar gas puffing. In 1999, (1) gas puff rate during increasing the plasma density, (2) Ar-gas feedback control were investigated as first step after the adjustment of the plasma configuration. Although H-factor and radiation fraction were comparable and RI mode has not been obtained, small ITB was observed in the density and temperature profiles in the early phase of high power injection. In 2000, density peaking was observed in a few discharges. Systematic study of these plasmas will
be important to understand the confinement improvement and to extend the operation regime to the higher density.

References

Fig. 1 Plasma configurations, (a) near-limiter during increasing density with low NB power injection, (b) divertor during Ar gas puff with high NB power injection.

Fig. 2 Fueling efficiencies for near-limiter and divertor plasmas.

Fig. 3 Time evolutions of plasma parameters using feedback controls of Ar gas and D₂ gas puffing.

Fig. 4 Enhancement factors of energy confinement, H₈₉L, and total radiation fractions, $P_{rad}^{tot}/P_{abs}$.

Fig. 5 Profiles of electron temperature and density during high NB power injection at low density.
7.4 Preliminary estimation of divertor heat flux during enhanced radiative discharges

H. Tamai, S. Sakurai, H. Kubo, S. Konoshima, N. Asakura, and K. Itami

1. Introduction

In order to achieve the high density and high radiation fraction with an improved core confinement, intense deuterium gas puff and/or additional impurity seeding with argon/neon are performed in JT-60U\(^1\). Reduction of heat load on the divertor plates and succeeding divertor detachment is expected. It is important to clarify the dependence of seeding gas and plasma configuration on the profile and absolute value of heat flux, and to estimate the power balance including the main and divertor radiation power. Preliminary analysis of heat flux on the divertor plates in the enhanced radiation plasma is presented.

2. IRTV camera

Heat flux is estimated from the temporal evolution of the divertor plates measured by the IRTV camera\(^2\). Available temperature range by the IRTV is 100-1200°C. Viewing area of IRTV is illustrated in Fig.1. Whole the area between inside and outside divertor plates is poloidally scanned by the rotating polygon mirror with a cycle of 4kHz in front of the IR detector. Therefore, the maximum time resolution is 0.25ms for one fixed position on the divertor plates. Temperature data are recorded in a CAMAC-AD convertor with the sampling rate of 1μs. By the limitation of CAMAC memory, data acquisition has been done in 1 scan (0.25ms) per 12.5ms, so that the time resolution in this experiment is entirely 12.5ms for one fixed position.

[Diagram of IRTV camera]

Fig.1 Viewing area of IRTV and location of gas puff

3. Deduction of heat flux

Temperature increase at time \(t\), \(T(t)\), of the divertor plate is a integrated function of heat flux, \(q(\tau)\), as

\[
T(t) = \left(\frac{\pi \rho C \kappa}{t}\right)^{-\frac{1}{2}} \int_{-\infty}^{t} dt (t - \tau)^{-\frac{1}{2}} q(\tau), \tag{1}
\]

where, \(\rho\), \(C\), and \(\kappa\) are weight density, specific heat, and thermal conductivity of divertor plate (CFC), respectively. Heat flux is numerically calculated by the differential approximation deduced from the Eq.1 as,
\[ q(t_n) = \left( \frac{\pi \rho \kappa}{\Delta t} \right)^{1/2} T(t_{n+1}) - (\Delta t)^{1/2} \sum_{i=0}^{n-1} (t_{n+1} - t_i)^{1/2} q(t_i) \]  \hspace{1cm} \text{Eq.2}

Heat conduction loss of deposited power is assumed in the perpendicular direction to the surface of divertor plates. For exact values of divertor plates of carbon-fibre-composite (CFC), \( \rho = 1.65 \times 10^3 \text{kg/m}^3 \), \( C = 6.69 \times 10^2 \text{J/(kg*K)} \), and \( \kappa = 200 \text{W/(m*K)} \) are adopted in the estimation using with Eq.2. Since \( \kappa \) of CFC is a decreasing function with temperature, the adopted value is at 500°C of the most likely temperature during discharge.

4. Heat flux during gas puff

High density and high radiation fraction has been achieved in ELMy H-mode by the assist of argon gas puff. Here the heat flux on the divertor plates in ELMy H-mode with \( I_p = 1.2 \text{MA} \), \( B_t = 2.5 \text{T} \), \( P_{NB} = 16 \text{MW} \) is compared between two cases of radiative discharges; one is by deuterium gas puff (E033678), and the other by additional argon gas puff (E033695). Location of gas puff is indicated in Fig.1.

Time evolution of typical plasma parameters for each discharge is shown in Fig.2. For the case of \( D_2 \) gas puff, increase of divertor radiation loss is observed after the start of \( D_2 \) gas puff. Since the main radiation loss decreases, the enhanced radiation is localised in divertor. Measurement of IRTV shows that the temperature at inner strike point gradually decreases, which suggests the divertor detachment. On the other hand, large main radiation loss rather than divertor radiation loss is observed for the case of Ar injection, which indicates the formation of radiative mantle around the main plasma. Fast decrease in temperature at inner strike point is observed.

![Fig.2](image.png)

Fig.2  Time evolution of typical plasma parameters for \( D_2 \) puff (E033678), and \( Ar+D_2 \) puff (E033695). From the top box, NB power, Ar puff rate, averaged density \( D_2 \) puff rate, divertor and main radiation loss, and temperature at inner, outer strike points.
Fig. 3 shows the profile of heat flux on the inner and outer divertor plates in the case of intense D₂ gas puff. After D₂ gas puff starting at 6.0s heat flux on inner divertor plate gradually decreases and at around 9.0s reaches detachment, which is defined as the elimination of heat flux at the strike point. On the contrary, on the outer divertor plate, decrease of heat flux is delayed at around 8.0s, and detachment never occurs.

![Diagram of heat flux profiles](image)

**Fig.3** Profile of heat flux on the divertor plates after intense D₂ gas puff (E033678) plotted against the major radius. Viewing area is illustrated in the lower box.

On the other hand, in the case of combined gas puff of Ar+D₂, the reduction of heat flux is more drastic as shown in Fig. 4. After Ar gas puff starting at 6.5s heat flux on inner divertor plate immediately decreases and detachment occurs at around 7.5s. Also outer divertor plate detachment is observed at round 7.5s. By Ar gas puff, fast reduction of heat flux on divertor plates is attained rather than that by only D₂ gas puff, thus detachment on both side divertor plates is achieved.

![Diagram of heat flux profiles](image)

**Fig.4** Profile of heat flux on the divertor plates after Ar+D₂ gas puff (E033695)
From those profiles, heat flux at the midplane is deduced by the magnetic flux mapping. As a result, effective e-folding length of heat flux at the midplane is roughly 2-2.5cm for D₂ gas puff and 1.5-2cm for Ar+D₂ gas puff.

5. Discussion

Total deposited power on the divertor plates is calculated by the integration of the heat flux profile. Figure 5 shows the power balance for both discharges. Deposited power is expressed in an arbitrary unit. The decrease in the deposited power qualitatively agrees with the increase of the radiation loss by the gas puff.

Fig. 5 Power balance for (a) D₂ and (b) Ar+D₂ gas puff

At present, the value of heat flux is expressed in an arbitrary unit, since the estimated deposited power exceeds the total input power during early phase before gas puff. Possible reason of this discrepancy would be the change in thermal conductivity of the divertor plates. According to an accurate inspection of the in-vessel during shut-down phase, surface of the divertor plates around the strike point has been observed like as amorphous. In such a case the thermal conductivity might be reduced, so that the heat flux would be over estimated in present calculation. In a trial calculation, the summation of heat flux and radiation loss never exceeds the total input power if the thermal conductivity would be reduced to a quarter, κ=50W/(m·K). Further investigation is required to estimate the absolute value of heat flux and deposited power on the divertor plates.

References
1) S. Sakurai, et al., IAEA TCM (October 1999, Kyushu Univ.)
7.5 Radiation and impurity transport in the hot plasma core of JT-60U reversed shear discharges


In reversed shear discharges with ITB (internal transport barrier), electron density, temperature and radiation power strongly increase inside the ITB. The core radiation is analyzed by bolometry, VUV spectrometer and CXRS (charge exchange recombination spectroscopy). The core radiation is evaluated with the spectroscopic method in the reversed shear regime.

During high performance discharges with strongly reversed shear, the radiation profile strongly peaks inside the ITB radius. 97% of radiation from inside the ITB radius is bremsstrahlung and this estimation is consistent with the bolometric measurement within 10%. The measured radiation is under 15 % of absorption power inside ITB (4.9MW) and the radiation loss from the core plasma is small. The carbon fraction estimated from $Z_{\text{eff}}$ and VUV spectrometer is consistent with CXRS measurement.

The tendency of carbon density increment is explained by neoclassical transport of banana-banana collision regime, although the magnitude of carbon particle flux derived from the experimental results is about 5 times smaller than the neoclassical theory. However, spectroscopy study shows that $n_c/n_e$ is flat inside ITB during the electron density increase. It implies that the carbon does not selectively accumulate inside the ITB. These apparent discrepancies may be explained by the transient nature of the discharge. If the reversed shear plasma is sustained for $\sim 1.5$ s longer, the carbon accumulation may become a problem to the core plasma performance. On the other hand, there is no carbon accumulation in the long pulse($\sim 2.6$s) weakly reversed shear plasma. The carbon density profile is flat over the radius in weakly reversed shear plasma, suggesting that the q-profile is important for impurity control.

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To be submitted
7.6 Progress on Transport and Microinstability Analysis of JT-60U Discharges at PPPL

K. Hill, N. Asakura, H. Kubo, D. Mikkelsen, R. Nazikian, G. Rewoldt, H. Shirai, and K. Young

Introduction

In 1998 the collaborative effort between physicists at JT-60U and the Princeton Plasma Physics Laboratory (PPPL) was significantly increased and placed on a more formal basis. PPPL researchers, jointly with their JAERI counterparts, wrote about 8 different experimental proposal document sheets (PDS) to study tokamak physics in several areas. These included studies of transport and confinement in reversed shear (RS) and other types of discharges, MHD stability studies, studies of fast particle physics and TAE modes, comparison and benchmarking of JAERI computer codes against PPPL codes, and analysis of radiative improved (RI) modes in JT-60U and comparison with RI experience in TFTR.

Some goals of the increased collaboration were to provide a richer physics experience to both JAERI and PPPL researchers by sharing ideas on physics understanding and by comparing results between JT-60U and TFTR, help to reduce the burden of analysis of JT-60U data on JAERI through increased hands-on participation by PPPL physicists, and to provide broader access of JAERI physicists to PPPL theoretical code results. Also, the PPPL impurity transport code MIST was installed on the JAERI computers for JAERI personnel to use.

Development of PPPL capabilities for analysis of JT-60U data

As one part of the development of capabilities at PPPL for analyzing JT-60U data, some PPPL physicists learned to use the JAERI computer programs for studying JT-60U data
and for extracting data files for further analysis with PPPL codes. The starting point for
data analysis at PPPL is the TRANSP transport code. This code both enables improved
understanding of tokamak physics and provides output which can readily be input to
additional PPPL theoretical codes to provide analysis of microinstability behavior, ExB
shearing rates, MHD stability, and fast particle transport and loss.

To facilitate TRANSP analysis of JT-60U shots, some existing computer codes were
ported to the CERES computer, and additional codes were written to convert SLICE
profiles to the TRANSP input format. The existing codes include those to read the JT-
60U equilibrium databases and provide boundary data for TRANSP; to read the NBI
powers, voltages, and fractions and write them to the TRANSP input file; and to read
scalar information such as stored energy, surface voltage, neutron source strength, plasma
radius, etc. for TRANSP input data. Also codes were written to convert SLICE profile
data to shot input files for running the PPPL MIST impurity transport code, and for
writing hydrogenic neutral density spectra calculated by TRANSP into MIST input files;
these hydrogenic neutrals affect the MIST calculated charge-state distribution through
charge exchange with impurity ions.

**Progress on Analysis of JT-60U Data at PPPL**

Several TRANSP runs have been done, mostly to serve as a starting point for
microinstability analysis, in order to correlate confinement with microinstabilities and
understand the reasons for improved confinement in RS and RI regimes. In a few cases,
the TRANSP runs were used to provide profiles and hydrogenic neutral spectra for MIST
runs on discharges with impurity puffing. In addition, profiles and other data files
required to perform TRANSP runs have been extracted for about 30 other shots-times.
These shots are mostly discharges with argon or krypton puffing, which were done as a
JAERI-PPPL collaboration. Queueing of the TRANSP runs is proceeding.
A preliminary comparison of the argon ion charge-state distributions predicted by IMPACT and MIST for an ELMy H-mode discharge with argon puffing, showed reasonable agreement between the peak locations and ion densities for the fully-stripped, hydrogen-like, and helium-like ions. Some differences in the predicted densities for lower charge states are being investigated. Efforts are underway to insure that both codes have the same and latest and best ionization and recombination rates included in their atomic databases. Also, differences between the measured radiated power profiles and the predictions of the two codes are being investigated. Improvements in the atomic physics databases and/or understanding of the impurity transport may result from these studies.

TRANSP and FULL code analyses were performed for a RS discharge, 29728, at two different times, including measured toroidal rotation. The radial profile of growth rate of the electrostatic toroidal drift mode calculated by the FULL code is illustrated in Fig. 1. This particular TRANSP run was made mainly using ufiles (universal file format of PPPL) of profiles provided by JAERI personnel.
Fig. 1 Growth rate as a function of normalized minor radius for electrostatic toroidal drift modes calculated for shot 29728 at 6.0 and 6.5 seconds, as calculated by the PPPL FULL microinstability code.

These calculations show that the growth rate for the electrostatic toroidal drift mode decreases in going from 6.0 to 6.5 s, and that the rate is further decreased by rotation, presumably by increased ExB shearing.
8 Plasma Diagnostics and Heating Systems

8.1 Development of collective Thomson Scattering Based on Pulsed CO\textsubscript{2} Laser\textsuperscript{1,2)}

T. Kondoh, S. Lee

1. Introduction

In fusion reactor, it is important to measure density and velocity distribution of the alpha particle created by fusion reaction in order to clarify the physics in alpha heated plasmas. However, the technique that measures confined alpha particle has not been established yet. Collective Thomson scattering measurement in JT-60U has been developed to measure the ion temperature using pulsed CO\textsubscript{2} laser, and to demonstrate alpha particle measurement for ITER.

2. Instrument

The conceptual scheme of collective Thomson scattering measuring system is shown in figure 1. Heterodyne receiver system and high power carbon dioxide gas laser and neutron shield have been installed in FY1998. In 1999, laser guide tube and optical system were installed in the torus hall. The laser beam is transmitted through the tube that is filled nitrogen or dry air in order to prevent the attenuation of the laser beam. Monitor cameras always watch the optic axis of laser beam and scattered light using helium neon laser. The laser is converged to the plasma center using the spherical mirror made of molybdenum. Position of the CO\textsubscript{2} laser is monitored behind the beam dump made of fused silica by video camera. Shift
of the optic axis which originates from the vibration of building and the diagnostic platform can be neglected.

3. Measurement limit

The region where the measurement of the ion temperature is possible is shown in figures 2. Since ion contribution to the scattered light must be larger than electron contribution, condition of the collective scattering \((\alpha_{e}>1)\) are necessary for the ion temperature measurement. Where \(\alpha_{e} = 1/(2\lambda_{D}/k_{laser}\sin(\theta/2))\), \(\lambda_{D}\) is the Debye length, \(k_{laser}\) is the input wave number, and \(\theta\) is scattering angle. Signal to noise ratio \((S/N)\) decides the lower limit of ion temperature measurement. \(S/N\) is given here in following equations.

\[
S/N = P_s/(P_n + P_s)\sqrt{B\tau + 1}
\]

Where \(B\) is the receiver bandwidth, \(\tau\) is the pulse length, \(P_s\) is the scattered power, and \(P_n\) is noise equivalent power of the detector.

Figure 2 Ion temperature measurable region.

Lower-limit of detection density of the fast ion in NNB heated plasma is shown in figure 3. In high \(T_e\) and low \(n_e\) region where slowing-down time of fast ions is long, the lower detection density limit of the fast ions is high. Therefore, careful optimization of the experimental condition is necessary to detect fast ions in NNB heated plasmas.

4. Summary

Installation of the CTS system has been finished, and the incidence of the CO\(_2\) laser was started in FY1999. The ion temperature measurement will be carried out in comparison with the result of the charge exchange recombination spectroscopy measurement. Detection of fast ions in NNB heated plasma will be tried in FY2000.

References
8.2 Advanced Impurity Measurement for DT Burning Plasmas using a Pulsed CO₂ Laser Collective Thomson Scattering

S. Lee and T. Kondoh

Advanced diagnostic technique to measure the impurities, thermalized helium density and the tritium to deuterium density ratio in DT burning plasmas is proposed. The measurement is made by collective Thomson scattering (CTS) using a high power pulsed CO₂ laser. In the condition of the scattered wave vector must be nearly perpendicular to the magnetic field direction, the spectral density function exhibits the characteristic modulation at the ion cyclotron frequency. The expected scattered power spectrum from pure deuterium and include argon (0.12%) plasmas in ITER are given in Fig. 1. The modulation occur at the deuterium cyclotron frequency of 40.4MHz and the spectrum extends to approximately 1 GHz. Although the spectrum becomes more complicated with impurity (Ar, Be, He ash) ions, spectrum components of main plasma and impurities can be distinguished because the two ion species have different Doppler widths and the impurity modulations are larger at the lower frequencies. In order to determine the helium ash density and D-T ratio, the scattered spectrum for D-T and impurities plasma is surveyed (Fig. 2). The modulations of main plasma and helium ash are overlapped with strong scattered spectrum from impurities. However, several modulation peaks represent the contribution of main plasma and helium ash. The helium ash density can be evaluated by the comparison with relative fitting curves which based on the scattered spectrum at the higher frequencies. Consequently, it is considered that the impurity measurement including helium ash using a $k \perp B$ scattering method can be expected as promising diagnostic technique in advanced fusion plasmas.

Fig. 1: Expected scattered spectrum for ITER, scattered $\theta =0.5$deg, $B \perp k=89.5$deg, with deuterium plasma and including Ar 0.12%.

Fig. 2: Expected scattered spectrum of D-T angle plasmas with impurities (Ar, Be and He ash).

Reference

8.3 Electron Density Measurement for Steady State Plasmas

Yasunori Kawano, Shin-ichi Chiba, and Akira Inoue

( Abstract )

Electron density of a large tokamak has been measured successfully by the tangential CO₂ laser polarimeter developed in JT-60U. The tangential Faraday rotation angles of two different wavelength of 9.27 and 10.6 μm provided the electron density independently. Two-color polarimeter concept for elimination of Faraday rotation at vacuum windows is verified for the first time. A system stability for long time operation up to ~10 hours is confirmed. A fluctuation of a signal baseline is observed with a period of ~3 hours and an amplitude of 0.4–0.7°. In order to improve the polarimeter, an application of diamond window for reduction of the Faraday rotation at vacuum windows and an another two-color polarimeter concept for elimination of mechanical rotation component are proposed.

Fig. 1 Line-averaged electron density measured by the 9.27-μm polarimeter, the 10.6-μm polarimeter, and the dual CO₂ laser interferometer, respectively.

Fig. 2 Line-averaged electron density obtained under the two-color polarimeter concept.

1) Y. Kawano, S. Chiba, and A. Inoue, accepted for publication in Journal of Plasma Fusion Research Series.
8.4 Development of O-mode reflectometer for density fluctuation measurement on JT-60U

N. Oyama and K. Shinohara

1. Introduction

The X-mode core correlation reflectometer, which has been developed in collaboration with Princeton Plasma Physics Laboratory, successfully measured the decay length of density fluctuations as shown in Ref. 1-2. The frequency range of the system is optimized for toroidal magnetic field, $B_T$, from 3.3 to 4.0 T in a consideration of typical density profile. Therefore, the system cannot measure in lower $B_T$ discharges due to the disappearance of cut-off layer. In typical density case, there is no cut-off layer in a discharge of $B_T < 3.0$ T. On the other hand, using O-mode propagation, the cut-off layer depends only on the electron density. In order to measure various case of plasma configurations, 3ch O-mode reflectometer has been designed for density fluctuation measurement to complement X-mode system.

2. Instruments

The targets of O-mode reflectometer are the change of fluctuation level and correlation during formation of the transport barrier, $H$-$L$ and $L$-$H$ transition, ELM and MARFE activity and so on. These targets are varying from edge to core plasma region. To cover these region, we chose 3 frequencies, 34GHz fixed, 34 to 40GHz selectable and 50GHz fixed, which correspond to the cut-off density of $1.43$, $1.43$-$1.98$ and $3.10 \times 10^{19}$ m$^{-3}$, respectively. Examples of target plasma are shown in Figure 1.

The transmission line which consists of corrugated waveguide was shared between X-mode and O-mode system to measure the same location of a plasma and to avoid the limitation of diagnostics port. Two separate lines for the incident wave and the received wave are used to reduce the cross talk and each line has focusing antenna with steering system which can control poroidal and toroidal mirror angle independently. The transition from rectangular waveguide in Q-band to corrugated waveguide for O-mode system using horn with lens is directly connected to the wire-grid beam splitter which acts as the polarization separator.

Figure 2 shows the schematic of millimeter-wave section of the O-mode reflectometer. There is some difference among each channel, but basically each channel is configured as a heterodyne system having two separate sources. They produce two intermediate frequency (IF) signals, one comes from the plasma, IF$_{SIG}$, and the other is used as a reference, IF$_{REF}$, with the frequency of 140 MHz. The optical fiber link system, which can transfer the signal in the frequency range from 10 MHz to 1.5 GHz, carries these IF signals from Torus Hall to
Diagnostic Hall. The quadrature-type phase detection system provides sin- and cosine-components, $E_{\sin \theta}$ and $E_{\cos \theta}$, in each channels. The outputs are acquired by PC-based 12 bit digitizer with 4 M words memory and then stored on the magneto optical (MO) jukebox via network file system (NFS) as shown in Fig. 3.

Fig. 1 Typical density profile measured with the YAG Thomson scattering system together with the cut-off position of each frequency. Left figure is high $\beta_p$-$H$ mode plasma in relatively high density case and right one is steady-state reversed magnetic shear plasma with $L$ mode edge.

Fig. 2 Schematic of millimeter-wave section of the O-mode reflectometer.
3. Experimental results

The 34 GHz fixed frequency reflectometer and 34 to 40 GHz selectable reflectometer have been started to measure the density fluctuation in JT-60U routinely from July, 1999 and March, 2000, respectively. These two channels of reflectometer are also applicable to correlation measurement.

First, the response of the reflectometer system was checked by reconstruction of the plasma motion from the phase variation of the reflected signal, since phase variation corresponds to the movement of cut-off layer directly assuming the density profile unchanged. The movement of cut-off layer calculated using reflectometer signal is in good agreement with that of plasma center as shown in Fig. 4. In lower density case that there is no cut-off layer, we also confirmed that the system acts as a interferometer. Then, the reflectometer system has been applied to the edge fluctuation measurement.

An example of the fluctuation spectrum measured with the 34 GHz fixed frequency reflectometer at the H-L and L-H transition is shown in Fig. 5. The time evolution of $D_\alpha$ signal measured at divertor region and spectrogram of the complex amplitude of reflectometer signal in degradation phase of high $\beta_p$-H mode plasma are shown. Increase of the fluctuation with the frequency lower than 80 kHz is clearly observed during L-mode phase. One feature of time slice of the spectrum is the reduction of zero frequency component, but total reflected power in low frequency region is almost same in both case. After L-H transition at $t = 6.53$ and 6.7 s, the fluctuation level decreased same as before H-L transition. On the other hand in high frequency region higher than 100 kHz, the amplitude of the fluctuation in H-mode phase become higher than L-mode phase. This features resemble the other tokamak device such as JFT-2M. According to YAG Thomson scattering measurement, the electron density at edge pedestal is $2.5 \times 10^{19} \text{m}^{-3}$ and measured point which corresponds to the cut-off density of $1.43 \times 10^{19} \text{m}^{-3}$ is kept almost constant even in the L-mode phase.
4. Summary

The 3ch O-mode reflectometer in Q-band has been designed for density fluctuation measurement. The 34 GHz fixed frequency and 34 to 40GHz selectable reflectometer successfully measured the density fluctuation and correlation of JT-60U plasmas. The 50 GHz reflectometer will be constructed in July, 2000.

References

2) Nazikian R. et al., "Core density fluctuations in reverse magnetic shear plasmas with internal transport barrier in JT-60U", in Plasma Physics and Controlled Nuclear Fusion Research (Proc. 17th Int. Conf. Yokohama), Post Deadline, Vienna, 1998, IAEA.

Fig. 4. Reconstruction of the plasma motion (dotted line) together with the motion of the plasma center (solid line). The plasma position was swung radially in 3 Hz with keeping plasma shape. Raw data of reflectometer was filtered by using low-pass filter with cut-off frequency of 500 Hz before reconstruction.

Fig. 5. Time evolution of Dα signal at divertor region and spectrogram of the reflectometer signal. H-L back-transition occurred at t = 6.49 and 6.60 s.
8.5 ECE Measurements in JT-60U

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N.Iwama* and JT-60 team
*Toyama Kenritsu University

1. ECE Measurement System in JT-60U

An ECE measurement system has been developed to obtain electron temperature with high time and spatial resolution for JT-60U plasmas. The system consists of three different instruments: a Fourier transform spectrometer system (FTS), a 20-channel grating polychromator system (GPS) and a 24-channel heterodyne radiometer system (HRS). These systems have different advantages and have been used complementarily. The FTS is absolutely calibrated and measures a wide frequency range of the ECE, and the GPS has a high time resolution with a flexibility of frequency range by the change of grating angle. The HRS has high time and spatial resolution with a better signal-to-noise ratio than the GPS, although the frequency range is limited.

2. Corrugated Waveguide Transmission System

The S-band rectangular waveguides with smoothed wall had been used for the transmission line of ECE diagnostics in JT-60U. In order to reduce the transmission loss, corrugated waveguide transmission system was installed from the antenna to just before the shielding wall in the Torus hall. After the installation, the total transmission loss was measured by the FTS. The reduction of the loss of ~3dB was attained by the corrugated system.

3. Preliminary Applications toward the Operation of Future Reactor

3.1 Feedback Control of Electron Temperature Profile in Real Time

A feedback control system on $T_e$ profile was developed in JT-60U, in order to avoid violation of MHD stability limit by controlling the gradient of pressure at the ITB in reversed shear plasma. As the actuator, several units of NB with central and peripheral heating depositions are adopted. The FTS is used for a real time monitor of electron temperature, where a real-time data processing by FFTP is used. The feedback system was applied to the steady-state reversed shear scenario.

3.2 Automatic Determination of the Sawtooth Inversion Radius with an Adaptive Neural Network

A neural network algorithm is applied to the ECE data of JT-60U plasmas. The automatic determination of the sawtooth inversion radius in a tokamak plasma was demonstrated. To detect the time of sawtooth collapse, a method of signal prediction with an adaptive neural network was applied to the time series of ECE data from the GPS, and the collapse time is determined from the time when the prediction error is at its maximum.

Reference
1) N.Isei et al., to be published in Fusion Engineering and Design.
8.6 Recent Development of ECE Diagnostic Systems

A. Isayama, N. Isei, S. Ishida and M. Sato

1. Introduction

In JT-60U, there are three electron cyclotron emission (ECE) diagnostic systems \(^1\), which are the Fourier transform spectrometer system \(^2\), the 20-channel grating polychromator system \(^3\) and the 24-channel heterodyne radiometer system \(^4, 5\). In this section, development of ECE diagnostic systems in FY1999, which is not included in the previous section, is described.

2. Installation of a highpass filter in the ECE transmission line

Electron cyclotron wave injection system has been installed in 1999. Since the frequency range of the EC wave overlaps with that of the ECE measurement, a high-pass filter was installed to reject the 110 GHz EC wave. At first the high-pass was to be installed in the transmission line for the FTS. However, it was found that EC wave from the ECH/ECCD system can reach the detectors of the GPS as stray light even when the measurement frequency region does not overlap with the EC wave frequency. Thus the highpass filter was installed before the power divider (See Fig. 1). As shown in Fig. 2, transmittance of the highpass filter is -60 dB at 110 GHz, which is expected to be enough to reject the EC wave from the gyrotron. Time evolution of ECE signal measured by the GPS before and after the installation of the highpass filter is shown in Fig. 3. As is shown in this figure, the EC wave is completely rejected, and electron temperature during the EC wave injection can be measured.

3. Upgrade of the heterodyne radiometer system

3.1 The 24-channel heterodyne radiometer system

A heterodyne radiometer is advantageous in that electron temperature perturbations can be measured with high signal-to-noise ratio. In JT-60U, the heterodyne radiometer system is mainly used for measurement of structure of electron temperature perturbations. In 1999, the number of channels was doubled to 24 by adding a new heterodyne radiometer. By using this system, ECE ranging from 176 to 200 GHz can be measured. Schematic diagram of the new heterodyne radiometer is shown in Fig. 4. It is composed of two components: ‘front end’ which converts the high frequency ECE to intermediate frequency (IF) and ‘back-end’ which detect, divide and filter the IF signal. A Gunn oscillator with 91 GHz and a doubler is used for the local oscillator. Inputted ECE signal is firstly filtered to reject the heterodyne image band ranging from 164 GHz to 176 GHz. After then, the ECE signal is mixed with the signal from the local oscillator and converted to intermediate frequency (6-18GHz). At the back-end, the IF signal is divided into the 12 channels. Each channel contains a bandpass filter and a video detector. Band width of the filter is \( f_0 \pm 0.25 \) GHz. Here, \( f_0 \) is the center frequency of the bandpass filter which ranges from 6.5 to 17.5 GHz at 1 GHz intervals. The system noise of the heterodyne radiometer is 12 dB, and the noise temperature is about 4300 K. Linearity of the output is kept within \( \pm 5\% \) in the range from -50 dBm to -20 dBm. After the detection, the signal of each channel is sent to isolation amplifier and analog-to-digital converter, and it is finally sent to the data acquisition computer through optical fibers.
After the upgrade of the heterodyne radiometer system, measurement range is extended as follows: $R=3.7 - 4.2 \text{ m}$ for $B_t=4.0 \text{ T}$, $R=3.3 - 3.7 \text{ m}$ for $B_t=3.5 \text{ T}$ and $R=2.8 - 3.2 \text{ m}$ for $B_t=3.0 \text{ T}$ (See Fig. 5). By using the upgraded heterodyne radiometer, the whole structure of various MHD modes, such as a tearing mode in the steady state high $\beta_p$ mode discharges (See section 4.9) can be measured.

3.2 ECE measurement using the heterodyne radiometer for JFT-2M

One of the disadvantages of the heterodyne radiometer is that the measurement points can not be chosen freely but uniquely determined by the value of the toroidal field. Thus, if the heterodyne radiometer for JT-60U is used in low field discharges such as $B_t=2.1\text{ T}$, the measurement points locates outside the plasma. Furthermore, in the low field region, the grating polychromator is not available because of low signal-to-noise ratio. In order to measure the electron temperature in the low field region, the heterodyne radiometer for JFT-2M is installed. Since the design of the IF component is similar in both radiometers, only the RF component is replaced. The schematic diagram of the radiometer for JFT-2M is shown in Fig. 6. The radiometer contains a 96 GHz and a 34 GHz Gunn oscillators, and can measure ECE ranging from 100 GHz to 124 GHz at interval of 1 GHz. The measurement range, which is shown in Fig. 5, is as follows: $R=2.8 - 3.4 \text{ m}$ for $B_t=1.9 \text{ T}$, $R=3.2 - 3.8 \text{ m}$ for $B_t=2.1 \text{ T}$ and $R=3.7 - 4.4 \text{ m}$ for $B_t=2.4 \text{ T}$. An example of the measurement is shown in Fig. 7, where toroidal field is 2.1 T. Sawtooth oscillations are clearly observed in this figure. The inversion radius can be also determined using this radiometer. The information is also important for the measurement of safety factor using the MSE (Motional Stark Effect) polarimeter.

4. Upgrade of the grating polychromator system

The grating polychromator system is advantageous in that the measurement points can be freely changed by changing the grating mirror angle. The channel separation can be changed by replacing the grating mirror. Usually the grating mirror whose pitch is 1.6 mm is used. In this case, the channel separation is typically 6 cm in major radius. The channel separation is too wider when we measure the structure of MHD instabilities and time evolution of electron temperature near the internal transport barrier. In FY1999, new detector system was installed aiming at decreasing the channel separation. Twenty new waveguides are inserted between the existing waveguides with which ECE is led to the InSb detectors. In order to cool the detectors, a cryostat was used since a refrigerator makes mechanical noise (See Fig. 8). By using the cryostat, 30 liters of liquid helium can be held. After the installation in the diagnostic room, start-up of the cryostat had been successfully done. The boil-off rate of the helium is about 0.3 liters/min, which corresponds to the hold-time of about 7 weeks. ECE measurement using this system will be started in 2000.

5. Summary

Development ECE diagnostic systems in FY1999 is summarized as follows: a) installation of a highpass filter to reject electron cyclotron wave from the gyrotron, b) upgrade of the heterodyne radiometer system, c) installation of the heterodyne radiometer for JFT-2M to measure electron temperature perturbation in low field region, d) upgrade of the grating polychromator system. All of them worked as planned. Measurement of ECE using the upgraded grating polychromator will be done in 2000.
References

Fig. 1: Layout of ECE diagnostic systems.

Fig. 2: Transmittance of the highpass filter.

Fig. 3: (a) Time evolution of electron temperature measured by GPS without power divider.
(b) Time evolution of electron temperature measured by GPS with power divider.
Fig. 4: Schematic diagram of the new radiometer.

Fig. 5: Measurement range of the heterodyne radiometer system at various magnetic field.

Fig. 6: Schematic diagram of the front-end of the radiometer for JFT-2M.

Fig. 7: Typical waveform of sawtooth crash measured by the radiometer for JFT-2M.

Fig. 8: Schematic diagram of the upgraded grating polychromator system.
8.7 Visible spectrometer with spatial resolution for divertor plasmas
T. Nakano, H. Kubo, S. Higashijima and T. Sugie

1. Introduction
Spectroscopic measurement is one of the most effective methods to understand complex processes such as divertor detachment, particle recycling, and impurity generation due to plasma-wall interactions. For the purpose of investigating atomic and molecular processes in divertor plasmas, it is necessary to observe the spectral lines or bands, which represent the characteristics of the process the most. Therefore, a spectroscopic measurement system, which consists of a spectrometer, a detector and other optics, must be flexible enough to be set up suitably for the objective of the measurement. A visible spectrometer has newly been installed in JT-60U. This system is designed to meet the demands for a variety of measurements. In this report, features of this spectroscopic measurement system are summarized.

2. Setup

![Fig. 1 16ch viewing chords for divertor region](image1)

![Fig. 2 Schematic view of JT-60U at P-8 section and the spectrometer setup.](image2)

Figure 1 and 2 show a schematic view of the spectrometer setup. Using 16 viewing chords, the divertor region is covered with spatial resolution of ~4cm. Emission from divertor plasmas is collimated with lenses mounted at P8-section and transmitted to the spectrometer in Shield Room through optical fibers. It is possible via the TCP/IP net working from Control Room to set up the ND filter, the width of the entrance slit, the grating and the voltage applied to the image intensifier, and also possible to configure the exposure time, the time resolution and the binning pattern of the CCD camera.
3. Spectrometer

The spectrometer is an imaging Czerny-Turner type with aspheric mirrors (SP-558, Action Research Corporation). Three gratings are set on a turret. ND filters with 3 different transmittance, a synthesized silica glass plate and a multi order cut filter (~665nm) are mounted in front of the entrance slit. The grating, the ND filter and the slit width can be controlled via serial communication interfaces by a software which works on windows 98. Specifications of the spectrometer and gratings are summarized in Table 1 and 2 and the wavelength resolution as a function of the slit width and diffraction efficiency of each grating are shown in Fig.3 and 4.

<table>
<thead>
<tr>
<th>Table 1 Spectrometer specifications</th>
</tr>
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<tr>
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<td>2400g/mm</td>
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**Fig.3** Full width at half maximum dependence on slit width
(source: HgI 546.1nm, grating 600g/mm)

**Fig.4** Diffraction efficiency for Grating with 150g/mm and 600g/mm
4. Detector

The detector is a frame transfer CCD with an image intensifier, ICCD-512EFT (Princeton Instrument Inc.). This CCD camera has two 512 x 512 pixel whose size is 15 \( \mu \text{m} \times 15 \mu \text{m} \). One is used as a light sensing section while the other as a shielded storage one. Detected signals are transferred into the shielded CCD within \( \sim 2 \text{ms} \).

RB (better Red) Gen II is equipped as an image intensifier. The quantum efficiency of the image intensifier is shown as a function of wavelength in Fig. 5. Since it is connected to the light sensing CCD with 1.5:1 tapered optical fibers, the effective pixel size of the CCD is 22.5 \( \mu \text{m} \times 22.5 \mu \text{m} \). High voltage supply for the image intensifier can be controlled by the software on the PC. Signal intensity is approximately proportional to an exponential function of the value of supplied voltage (Fig. 6).

The CCD must be cooled with a peltier element down to \(-40^\circ\) C. Dried air circulates on the CCD in order to suppress arc and corona discharges.

5. Controller

The controller is ST-138 (Princeton Instruments Inc.). Stored photoelectrons in the shielded CCD are transferred to this controller and then, digitized according to a preprogrammed binning pattern with 14bit resolution at a speed of 1MHz. Finally, the digital data are transmitted to the control PC via a special interface board (PCI).
Figure 7 shows the linearity of the signal digitized by this controller. It shows an outstanding characteristic, that is within $\pm 1\%$.

Maximum time resolution of the system is determined by a binning pattern and exposure time. Typical time resolution is 25ms with 16 viewing-codes and 10ms exposure time.

*Fig. 7 Linearity of the data digitized by ST-138.*
8.8 Bolometer Tomography

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W H Meyer\(^4\), S Sakurai, H Kubo, N Hosogane and H Tamai

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\(^2\) Plasma Research Center, University of Tsukuba, Ibaraki, Japan
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\(^4\) Lawrence Livermore National Laboratory, Livermore, CA, USA

Hardware improvements of in-vessel divertor bolometer cameras to withstand severe electrical and thermal loads and the development of tomography software have made more detailed visual studies of divertor radiation possible. Line integrated bolometer signals of 48 total sight lines are successfully mapped onto the JT-60U divertor geometry indicating characteristic profiles for tokamak operational regimes. Tomographic analysis was confirmed to be consistent with an independent measurement of the radiating layer width at the target plate. Upstream extension of the radiative zone in reversed shear mode appears to depend on seeded impurity species.

Figure 1 schematically shows the arrangement of the bolometer cameras. The in-vessel bolometer cameras were recently made operational by improving the electrical and thermal insulation of the sensors during the last two years. The modifications include adding a 3mm thick ceramic plate at the windows, replacing the first collimator with small apertures and a grounded point for the vacuum feed-through flange. The small apertures of the in-vessel cameras enable detailed radiation profile measurements. On the other hand, the reduction of incident power has made the detectors more sensitive to cooling by cold gas flow, which sometimes create non-negligible baseline drift during measurements.

Radiating Layer Width

Taking advantage of the good spatial resolution of the divertor bolometer camera, the width of the radiating layer at the target tiles was investigated during strike point sweep discharges. Figure 2 shows an example of the data. Signals ID3,4 and OD4 of the divertor camera views a specific local area of approximately 5 to 7 mm in poloidal length along the tile surface at the inboard and outboard target. The divertor strike points were intentionally swept up and down 10 cm along the target in 4 seconds during a high power ELMing discharge with 23 MW at average density of \(2.5 \times 10^{19}\) m\(^{-3}\). The strike points of the separatrix at the tile surface are expressed as Li for inside and Lo for outside, shown
schematically in the top. The bolometer signals ID3, 4 and OD4 show an asymmetric waveform with a gentle slope as the strike points, Li and Lo, move from a lower position up to a peak and a sharp fall-off above the peak. The gentle slope, peak and subsequent sharp fall-off correspond to radiation in the scrape-off layer, separatrix and private flux region, respectively. It clearly indicates that radiation in the private region is almost negligible. The sharp fall-off into the private flux region is expected due to the short connection length. The total width of the scrape-off layer (SOL) radiation is identified to be in the range of 4–6 cm in this case. A flux line of 8 (or 3) cm from the separatrix at the target corresponds approximately to 2 (or 1) cm at the mid-plane for this particular shot. The full width of the radiating layer equivalently corresponds to the width of about 3 to 4 grids in the tomography analysis.

Two Dimensional Tomographic Reconstruction

A bolometer measures a line-integrated value of the local radiative emissivity along a viewing chord. In order to obtain the distribution of the local radiation emissivity in the poloidal cross section of the plasma, the measured line integrals must be unfolded. The entire poloidal cross section of the radiating plasma including first walls and the W shaped divertor is partitioned into 128 x 128 grid points. The size of each element grid, which determines the minimum spatial resolution of the system, is $\delta x = 2.03$ cm horizontally and $\delta y = 2.81$ cm vertically. Emissivity is assumed to be constant within each cell. The core plasma radiation profile is possible to determine separately with upper 8 chord signals without viewing the divertor by an Abel inversion (precisely a Radon transform) technique assuming emissivity is poloidally uniform along each magnetic flux surface in the core. Results of the Abel inversion are then distributed on the grid space according to their respective flux values. Before inverting the divertor radiation, the contribution of the core radiation is subtracted from the divertor chords which also pass through the main plasma. The cells within a divertor radiation zone are to be determined. Six additional variable boundaries other than divertor tiles and walls define the divertor radiation zone. Each viewing chord of total 48 channels passes through specific grid points. The line integral value $P_i$ can be expressed by $\Sigma_j (E_{ij})(R_j)$ using a predetermined response matrix ($E_{ij}$) for each detector i, where $R_j$ is the emissivity in each grid. The response matrix has mostly zero values with non-zero values only for grid points viewed by the specified chord. Since the number of measurements is small compared to the number of cells, i.e. unknowns, the equation can not be solved uniquely. An additional constraint equation, a smoothing term, is added to avoid singularity as follows;

$$\min. = \Sigma_j \left\{ (E_{ij})(R_j) - P_i \right\} \left( (\sigma_i P_i) \right)^2 + S \int \left( \nabla^2 R_j \right)^2 \, dxdy$$

The first term on the right is the chi-square value of fitting the local emissivity to the
measured line integral value with measurement uncertainty $\sigma_i$. The second term imposes a finite rate of change on the gradient between neighboring cells with a smoothing factor $S$, a diffusion term of local emissivity with a radiating source. Approximately 40 channels which view the divertor area from many different angles are inverted using a conventional 'linearly constrained least squares problem subroutine' for the above matrix equation. The typical number of cells to be determined is the order of 1000. Inverted profiles shown in the following are the results of a parameter study with these boundary conditions, in addition to $S$ and the relative weight between channels through measurement uncertainty.

**Reversed Shear Discharges with Argon and Neon seeding**

Figure 3 shows an example of preliminary analysis for neon and argon seeded reversed shear discharges together with a reference discharge, no impurity gas puff. Emissivity contours are expressed in a linear scale from 0.05 MW/m$^3$ to 5 MW/m$^3$. Discharge conditions are kept almost identical other than the impurity gas puff for the three discharges and comparison is made at a timing of approximately the same line average density of around $3 \times 10^{19}$ m$^{-3}$. These discharges have a higher triangularity shape of 0.3 to 0.35. About half of the input power is dissipated with radiation in all cases. The divertor radiation fraction of the total radiation is 37, 54 and 41 % for the case of no impurity (carbon is a main radiator in this case), neon and argon, respectively. A highly localized radiation peak at the inboard target is observed in the cases of no impurity gas puff and neon seeding. A notable characteristic of neon seeding is that a ring shaped radiating mantle at the separatrix is uniformly formed around the core plasma including the divertor. In the case of argon, the highly radiative inboard zone spreads out reaching the x-point and the core plasma, indicating a clear contrast to the former two cases. Also significant radiation of about 0.2 MW/m$^3$ in the central core, inside the internal barrier, is identified in this case. The difference between species may crudely be understood from a simple coronal equilibrium for impurity radiation rate coefficients, in which argon emissivity extends to a significantly higher temperature than carbon and neon. A more in depth study with a divertor simulation code such as IMPMC will be necessary to fully examine this issue.
Fig. 1 Overview of the JT-60U bolometer arrangement and the in-vessel divertor bolometer cameras.

Fig. 2 An asymmetric shape of the radiating zone across the separatrix with approximate e-folding width of 2 to 3 cm in the scrape-off layer in ELMing H-mode.

Fig. 3 Radiation distribution of a typical reversed shear discharge without impurity gas puff (reference case), neon seeding case and argon seeding case. Average density is about $3 \times 10^{19}$ m$^{-3}$. Maximum emissivity is 4.5 MW/m$^3$. 
8.9 110GHz, 1MW ECRF system on JT-60U [1]


The 110GHz 1MW Electron Cyclotron Range of Frequency (ECRF) system was designed and constructed on JT-60U to locally heat and control the plasmas. The gyrotron has a diamond window to transmit RF power with Gaussian mode, which is easily transformed to HE_{11} mode for the transmission line of the corrugated waveguide. The second diamond window is installed at the inlet of the antenna for a vacuum seal between the transmission line and the JT-60U tokamak. The total length of the transmission line from the gyrotron to the antenna is about 60 m including nine miter bends. The antenna has a focusing mirror and a flat steerable one to focus and to control the RF beam angle mainly in the poloidal direction. The main design parameters of the JT-60U ECRF system are listed in Table 1.

<table>
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In the initial operation, the power of $P_{\text{EC}} \sim 0.75$ MW for 2 s was successfully injected into plasma when the gyrotron generated the power up to 1 MW. The total transmission efficiency from the gyrotron to the plasma was about 75%. A controllability of local electron heating with the deposition width of $\leq 15$ cm was well demonstrated by using the steerable mirror. A large downshift in the deposition position was observed at the high $T_e$ plasma. Strong central electron heating was obtained from 2.2 keV to 6.6 keV for $P_{\text{EC}} \sim 0.75$ MW, 0.3 s at the optimized polarization. An effective electron heating was also obtained up to $\sim$10keV during EC injection for $\sim$1.6 s in the high $\beta_p$ H-mode plasma produced by NBI.

8.10 Polarization control of JT-60U ECRF System

K. Kajiwara, K. Takahashi

Introduction

An electron cyclotron range of frequency (ECRF) system is considered as one of effective tools for plasma heating, electron cyclotron current drive (ECCD), magnetohydrodynamics (MHD) control and plasma start-up. From March in 1999, ECRF experiments in JT-60U were started to demonstrate the technical feasibility of the high power ECRF system and the improvement of plasma performance.

In JT-60U, the RF wave is obliquely injected from low magnetic field side for ECCD. On such a configuration, the polarization study of ECRF wave is important to obtain the appropriate wave coupling to plasma. Since the cutoff density for X-mode launch from the low magnetic field side is relatively lower, the O-mode injection is necessary. The polarization to excite O-mode is the elliptical polarization, in case of oblique injection. So as to control the polarization, a pair of polarizer is installed in transmission line. The optimum polarization depends on the magnetic field strength, the plasma density and the injection angle between the RF ray and magnetic field line. In this section, the measurement of polarization that is changed by polarizers, calculation of O-mode purity and dependence of electron heating on polarization, is described.

Polarization measurement of ECRF in transmission line

Two rotatable miter bend polarizers with different groove depths are sufficient to generate a rather wide range of polarization. The polarization is changed by rotating the stage. Since polarization is quite sensitive to figure of the groove and there are, practically, some errors in grating groove on its manufacture, it is necessary to measure the polarization that is changed by polarizers. The low power measurement of wave polarization was carried out in the JT-60U transmission line.

The experimental configuration is shown in Fig. 1. The low cw oscillator (<1mW) and mode converter TE01->HE11 were used for the measurement, instead of high power gyrotron, to produce a linearly polarized HE11 mode. A rectangular horn antenna and a crystal diode detector set on a rotational stage were installed at the location of the directional coupler, near the torus window in the tokamak hole. The rectangular horn detects only one direction electric field, consequently, the ellipticity $R_m$ and polarization angle $\alpha$ (see Fig. 2) can be identified by rotating the horn. We measured $R_m$ and $\alpha$ each 10° angle of polarizer 1 and polarizer 2 from 0 to 180° (The number of measurement is $19 \times 19 = 361$).
Calculation of the O mode purity

The 3-dimensional coordination of polarization, plasma shape and magnetic field line should be considered in O mode purity calculation. The rotation of wave polarization at two miter bends mirror and the injection antenna that are installed in front of the directional coupler was considered. The tilt of the magnetic field line was roughly obtained using poloidal coil currents, plasma position and plasma current. We examined about the discharge as shown in Fig. 3.

The calculation coordinate is shown in Fig. 2. The O mode purity is calculated by using following equations\(^2\).

\[
\frac{E_{y}^{x,o}}{iE_{z}^{x,o}} = \frac{D(P - N_{r,o}^{2} \sin^{2} \Theta)}{N_{r,o}^{2} \cos \Theta \sin \Theta(S - N_{r,o}^{2})} \quad \frac{E_{z}^{o}}{E_{z}^{o}} = \frac{P - N_{r}^{2} \sin^{2} \Theta}{-N_{r}^{2} \cos \Theta \sin \Theta}
\]

\[
E_{z}^{o} = \frac{E_{l}}{E_{y}^{o}/sE_{z}^{o} - E_{y}^{x}/sE_{z}^{x}} \left( R - \sin \Theta \frac{E_{z}^{o}}{E_{z}^{o}} \right)
\]

\[
P = 1 - \frac{\omega_{p}^{2}}{\omega^{2}}, \quad S = 1 - \frac{\omega_{p}^{2}}{\omega^{2}} \left( \frac{\omega_{c}^{2}}{\omega_{c}^{2}} \right), \quad D = \frac{\omega_{p}^{2}}{\omega^{2}} \left( \frac{\omega_{c}^{2}}{\omega_{c}^{2}} \right)
\]

The symbol N shows the refractive index of plasma waves. The suffix o and x indicate O and X mode, respectively. In this calculation the refraction of the wave at the plasma edge is neglected because the refractive index is almost one. The symbol R is the modified polarization ratio which is derived with the measured ellipticity \(R_{m}\) and measured polarization angle \(\alpha\) as follows.

\[
R = -iR_{m} \exp \left[ \pm i \cos^{-1} \left( \sqrt{\tan^{2} 2\alpha \cos^{2}(2 \tan^{-1} R_{m})} \right) \right]
\]

+; Left-handed wave  -; Right-handed wave

In the polarization measurement, the right handed-wave and left handed-wave could not be distinguished. Therefore, we confirm this problem by injecting the polarized wave to plasma. O-mode component of the pointing flux \(S_{o}\) is indicated as follows

\[
S_{o} = (E_{x}^{x,o}^{2} + E_{y}^{x,o}^{2} + E_{z}^{x,o}^{2})N_{o}
\]

The O-mode purity map of the excited wave in plasma, shown in Fig. 3, is indicated in Fig 4(a) and (b).

Dependence of electron heating on polarization

The ECH experiment was carried out to investigate the dependence of electron heating on polarization. The fundamental O mode wave was selected for the experiment\(^3\). Plasma parameters are shown in Fig. 3. Electron temperature was measured by grating polycromator. To distinguish the right-handed and left-handed wave, we set the polarizer 1 of 50° and polarizer 2 of 20°. The O mode purity is calculated to 0.33 and 0.31 with right-handed and left handed assumption.
respectively. These are almost same O mode purity. However, in the set of the polarizer 1 of 50° and polarizer 2 of 50°, the O mode purity only increases with right-handed assumption (left handed assumption: 0.34, right-handed assumption: 0.96). The temperature increase is 0.71 keV in this setting. Therefore, the right-handed wave injection was confirmed.

The dependence of electron heating on polarization is shown in Fig. 5 with right handed assumption. These plots show the increase of the central electron temperature for the first 20 ms of the RF pulse. The difference of temperature increase is well explained by ECRF polarization. It is confirmed that the accuracy of measurement of polarization and calculation of O-mode purity. It is also suggests that the polarization control is significantly effective for ECRF experiments in JT-60U.

Summary
The polarization measurement on the JT-60U transmission line of ECRF system was carried out. The map of wave polarization and that of the O mode purity of the incidence wave for JT-60U plasma were obtained. The dependence of electron heating on polarization was investigated with the O mode purity map in JT-60U. It is confirmed that the accuracy of measurement of polarization and calculation of O-mode purity. It is also suggests that the polarization control is significantly effective for ECRF experiments in JT-60U.

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Fig. 4 O-mode purity map of the excited wave in the discharge that is shown in Fig. 3. Figure (a) and (b) are assumed the right-handed and left-handed elliptical polarization, respectively.

Fig. 5 The dependence of electron heating on polarization. The horizontal axes is decided by calculational result of O-mode purity with right-handed assumption.

References
Acknowledgments

The authors wish to acknowledge the dedicated efforts of the members of Japan Atomic Energy Research Institute in support of the JT-60U experiments reported here. They also wish to express their gratitude for domestic and international collaborations with the JT-60U program. The contribution of collaborators from many institutions and universities have been very important for the success of the JT-60U experiments.
国際単位系（SI）と換算表

<table>
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（注）

1. 表1-5は「国際単位系」第5版、国際度量衡局1985年制定による。ただし、1 eV および1 uの値はCODATAの1986年推奨値によって。
2. 表4には秒、ノット、アーム、ヘルツも含まれているが日常の単位なのでここでは省略した。
3. barは、JISでは流体の圧力を表す場合には表2のカレーリーに分類されている。
4. EC関連理事会ではbar、barnおよび「面積の単位」mmHgを表2のカレーリーに入れている。

<table>
<thead>
<tr>
<th>算出表</th>
<th>MPa(-10 bar)</th>
<th>kgf/cm²</th>
<th>atm</th>
<th>mmHg(Torr)</th>
<th>lbf/in²(psi)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>10.1972</td>
<td>9.86923</td>
<td>75062.0×10^2</td>
<td>145.038</td>
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</tr>
<tr>
<td>1</td>
<td>0.098665</td>
<td>0.967841</td>
<td>735.559</td>
<td>14.2233</td>
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<tr>
<td>1</td>
<td>0.101325</td>
<td>1.03233</td>
<td>762</td>
<td>14.6959</td>
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<tr>
<td>1</td>
<td>1.33322×10^-4</td>
<td>1.3591×10^-3</td>
<td>1.31579×10^-3</td>
<td>1</td>
<td>1.93398×10^-5</td>
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<tr>
<td>1</td>
<td>6.8946×10^-7</td>
<td>7.0703×10^-7</td>
<td>6.8046×10^-7</td>
<td>51.7149</td>
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</table>

<table>
<thead>
<tr>
<th>エネルギー</th>
<th>J(10^12erg)</th>
<th>kgf・m</th>
<th>kW・h</th>
<th>cal(計算法)</th>
<th>Btu</th>
<th>ft・lbf</th>
<th>eV</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>1.0109172</td>
<td>2.77778×10^-1</td>
<td>0.238889</td>
<td>9.47813×10 ^5</td>
<td>0.737652</td>
<td>6.24150×10^-7</td>
<td>= 1.814 J (熱化学)</td>
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<tr>
<td>1</td>
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<td>2.72407×10^-1</td>
<td>0.234720</td>
<td>9.29487×10^-7</td>
<td>7.23301</td>
<td>6.12082×10^-7</td>
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<tr>
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<td>1</td>
<td>8.59899×10^-4</td>
<td>3412.13</td>
<td>2.65522×10^4</td>
<td>2.24694×10^-4</td>
<td>= 1.8168 J (国際標準気)</td>
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<td>107.586</td>
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<tr>
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<td>1.63377×10^-10</td>
<td>4.4506×10^-10</td>
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<tr>
<th>放射能</th>
<th>Bq</th>
<th>Ci</th>
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</thead>
<tbody>
<tr>
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<td>2.72027×10^-11</td>
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<tr>
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<th>Gy</th>
<th>rad</th>
</tr>
</thead>
<tbody>
<tr>
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</table>

<table>
<thead>
<tr>
<th>吸収線量</th>
<th>C/μ</th>
<th>R</th>
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</thead>
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<td>1</td>
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</tbody>
</table>

<table>
<thead>
<tr>
<th>塩基度</th>
<th>Sv</th>
<th>rem</th>
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<tbody>
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<td>1</td>
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