JAERI-Review
2003-011

JAERI CONTRIBUTION TO THE 19TH IAEA FUSION ENERGY CONFERENCE
(October 14-19, 2002, Lyon, France)

March 2003

(Ed.) Tokamak Program Division

日本原子力研究所
Japan Atomic Energy Research Institute
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編集兼発行 日本原子力研究所
JAERI Contribution to the 19th IAEA Fusion Energy Conference
(October 14–19, 2002, Lyon, France)

(Ed.) Tokamak Program Division
Department of Fusion Plasma Research
Naka Fusion Research Establishment
Japan Atomic Energy Research Institute
Naka-machi, Naka-gun, Ibaraki-ken

(Received January 31, 2003)

This report compiles the contributed papers and presentation materials from JAERI to the 19th IAEA Fusion Energy Conference held at Lyon, France, from October 14th to 19th, 2002. The papers describe the recent progress in the experimental research in JT-60U and JFT-2M tokamaks, theoretical studies, fusion technology and R&D for ITER and fusion reactors. Total 32 papers consist of 1 overview talk, 14 oral and 17 poster presentations. Eight papers written by authors from other institutes and universities under collaboration with JAERI are also included.

Keywords: IAEA, JAERI, Conference Meeting, Fusion Energy, Papers and Oral Presentation, Progress Research,
IAEA 主催第 19 回核融合エネルギー会議原発表論文集
（2002 年 10 月 14 日～10 月 19 日、リヨン、フランス）

日本原子力研究所那珂研究所炉心プラズマ研究部
（編）炉心プラズマ計画室

（2003 年 1 月 31 日受理）

本報告書は、2002 年 10 月 14 日から 10 月 19 日にかけてフランフ、リヨンで開催された
IAEA 主催第 19 回核融合エネルギー会議において発表された原発の論文とその発表資料
をまとめたものである。発表論文総数 32 編のうち、総合講演 1 編、口頭発表 14 編、ポスター
発表が 17 編である。国内外の研究機関や大学との共同論文 8 編も合わせて収録した。
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List of Authors

N. Asakura\textsuperscript{11}, C. Z. Cheng\textsuperscript{2}, M. Enoeda\textsuperscript{5}, T. Fujita\textsuperscript{4}, M. Furukawa\textsuperscript{15}, A. Hatayama\textsuperscript{16}, K. W. Hill\textsuperscript{2}, S. Ide\textsuperscript{1}, Y. Idomura\textsuperscript{15}, K. Ioki\textsuperscript{17}, A. Isayama\textsuperscript{21}, S. Ishida\textsuperscript{18}, Y. Ishii\textsuperscript{15}, Y. Kamada\textsuperscript{18}, K. Kamiya\textsuperscript{14}, H. Kawamura\textsuperscript{19}, Y. Kishimoto\textsuperscript{15}, G. Kurita\textsuperscript{18}, Y. Miura\textsuperscript{11}, Y. Nakamura\textsuperscript{10}, S. V. Neudatchin\textsuperscript{11}, S. Nishio\textsuperscript{10}, S. Ohtake\textsuperscript{12}, N. Oyama\textsuperscript{11}, K. Sakamoto\textsuperscript{11}, Y. Sakamoto\textsuperscript{13}, A. Sakasai\textsuperscript{18}, S. Sato\textsuperscript{31}, T. Shikama\textsuperscript{15}, M. Shimada\textsuperscript{17}, K. Shinohara\textsuperscript{11}, T. Suzuki\textsuperscript{11}, Y. Takase\textsuperscript{16}, M. Takechi\textsuperscript{11}, H. Takenaga\textsuperscript{11}, T. Tanabe\textsuperscript{17}, M. Taniguchi\textsuperscript{13}, K. Tsuzuki\textsuperscript{14}, N. Umeda\textsuperscript{18}

\textsuperscript{1} Large Tokamak Experiment and Diagnostics Division, Department of Fusion Plasma Research
\textsuperscript{2} Princeton Plasma Physics Laboratory, USA
\textsuperscript{3} Blanket Engineering Laboratory, Department of Fusion Engineering Research
\textsuperscript{4} Experimental Plasma Physics Laboratory, Department of Fusion Plasma Research
\textsuperscript{5} Plasma Theory Laboratory, Department of Fusion Plasma Research
\textsuperscript{6} Keio University
\textsuperscript{7} International Coordination Division, Department of ITER Project
\textsuperscript{8} Tokamak Program Division, Department of Fusion Plasma Research
\textsuperscript{9} Blanket Irradiation and Analysis Laboratory, Department of JMTR, Oarai Research Establishment
\textsuperscript{10} Reactor System Laboratory, Department of Fusion Plasma Research
\textsuperscript{11} Kurchatov Institute, Russia
\textsuperscript{12} Ministry of Education, Culture, Sports, Science and Technology
\textsuperscript{13} Plasma Heating Laboratory, Department of Fusion Engineering Research
\textsuperscript{14} Fusion Neutron Laboratory, Department of Fusion Engineering Research
\textsuperscript{15} Tohoku University
\textsuperscript{16} The University of Tokyo
\textsuperscript{17} Nagoya University
\textsuperscript{18} NBI Facilities Division, Department of Fusion Facilities
執筆者一覧

朝倉 信幸^{1}・C. Z. Cheng^{2}・桜枝 幹男^{3}・藤田 隆明^{4}・古川 勝^{5}・細山 明聖^{6}
K. W. Hill^{12}・井手 俊介^{1}・井戸村 泰宏^{15}・伊尾木 公裕^{17}・諏山 明彦^{11}
石田 真一^{18}・石井 康友^{5}・篠田 裕^{8}・神谷 健作^{14}・河村 弘^{9}・岸本 泰明^{15}
栗田 源一^{18}・三浦 幸俊^{1}・中村 幸治^{10}・S. V. Neudatchin^{11}・西尾 敏^{10}
大竹 素^{12}・大山 直幸^{1}・坂本 慶司^{1}・坂本 宜照^{13}・逆井 章^{8}・佐藤 聡^{14}
四竜 樹男^{15}・船田 道也^{17}・篠原 孝司^{1}・鈴木 隆博^{1}・高瀬 雄一^{16}・武智 学^{11}
竹永 秀信^{14}・田辺 哲朗^{17}・谷口 正樹^{13}・都築 和泰^{14}・梅田 尚孝^{18}

^1 炉心プラズマ研究部 炉心プラズマ実験計測開発室
^2 プリンストンプラズマ物理研究所、米国
^3 核融合工学部 プランケット工学研究室
^4 炉心プラズマ研究部 プラズマ物理実験研究室
^5 炉心プラズマ研究部 プラズマ理論研究室
^6 慶応義塾大学
^7 ITER 開発室 ITER 協力調整室
^8 炉心プラズマ研究部 炉心プラズマ計画室
^9 大洗研究所 材料試験炉部 プランケット照射開発室
^10 炉心プラズマ研究部 核融合炉システム研究室
^11 クルチャトフ研究所、ロシア
^12 文部科学省
^13 核融合工学部 加熱工学研究室
^14 核融合工学部 核融合中性子工学研究室
^15 東北大学
^16 東京大学
^17 名古屋大学
^18 核融合装置試験部 NBI 装置試験室
1. JAERI Papers

1.1 Overview of JT-60U Results toward High Integrated Performance in Reactor-Relevant Regime

T. Fujita, JT-60 Team

Japan Atomic Energy Research Institute, Naka Fusion Research Establishment
Naka-machi, Naka-gun, Ibaraki-ken, Japan.

e-mail contact of main author: fujitata@fusion.naka.jaeri.go.jp

Abstract. Recent JT-60U results toward high integrated performance are reported with emphasis on the projection to the reactor-relevant regime. N-NB and EC power increased up to 6.2 MW and 3 MW, respectively. A high $\beta_p$ H-mode plasma with full non-inductive current drive has been obtained at 1.8 MA and the fusion triple product reached $3.1 \times 10^{26}$ m$^{-3}$keVs. High beta with $\beta_p = 2.7$ was maintained for 7.4 s. NTM suppression with EC was accomplished using a real-time feedback control system and improvement in $\beta_p$ was obtained. A stable existence of current hole was observed. High DT-equivalent fusion gain of 0.8 was maintained for 0.55 s in a plasma with a current hole. The current profile control in high bootstrap current reversed shear plasmas was demonstrated using N-NB and LH. A new operation scenario has been established in which a plasma with high bootstrap current fraction and ITBs is produced without the use of OH coil. ECCD study was undertaken in a reactor-relevant high $T_e$ regime. A new type of AE mode has been proposed and found to explain the observed frequency chirp quite well. High confinement reversed shear plasmas with $T_2 > T_1$ were obtained. Air exhaust with EC heating was obtained in a high $\beta_p$ mode plasma. Impurity accumulation related to strong ITBs in a reversed shear plasma and degradation of ITB by ECH in a weak positive shear plasma have been found. Dedicated measurement of ELM dynamics and SOL plasma flow advanced the physics understanding. N-NB heating in an Ar-seed plasma extended the density region to 95% of Greenwald density with $HH_{N\alpha} = 9$. The enhancement of pedestal pressure was obtained with an increase of $\beta_p$ in a high triangularity configuration.

1. Introduction

The main purpose of JT-60U project is to establish scientific basis for ITER and demo tokamak reactor. Our ultimate goal is to achieve and sustain high integrated performance, namely high beta, high confinement, high bootstrap current fraction, full non-inductive current drive and heat/particle control, in a reactor-relevant regime. We have developed weak magnetic shear ("high $\beta_p$ mode") [1,2] and reversed magnetic shear [3] plasmas toward this goal [4]. In both regimes, the internal transport barrier (ITB) [5,6] and the edge pedestal are obtained simultaneously. As a large-sized tokamak equipped with a variety of devices for heating, current drive and profile control, JT-60U has high ability to approach the conditions required in reactors (ITER or demo): low values of normalized Larmor radius and collisionality, high toroidal field, high temperature with $T_2 > T_1$, small central fueling, small ELM activities, etc. This paper reports recent JT-60U results, after the last IAEA conference [7], with emphasis on the projection to the reactor-relevant regime.

2. Improved Machine Status

The JT-60U tokamak has a large variety of heating and current drive systems, which consist of the conventional positive-ion-based neutral beams (P-NBs, co- and counter-tangential and perpendicular injection), high-energy tangential negative-ion-based neutral beams (N-NBs), LHRF, ICRF and ECRF systems. The N-NB provides, in addition to the current drive, the electron heating through high-energy ions with small particle fueling, and can simulate the plasma heating and/or excitation of Alfvén instability by $\alpha$ particles in reactors. The ECRF system enables us to vary the ratio of $T_2/T_1$ in a wide region including the reactor-relevant regime ($T_2 > T_1$) by efficient on-axis electron heating.

In the N-NB system, the beam deflection and uniformity of the source plasma have been improved. The increased acceleration current resulted in the injection power of 6.2 MW with
the injection energy of 381 keV and the pulse length of 1.7 s. The improved beam divergence extended the pulse length and the long pulse injection for 10 seconds has been achieved at 355 keV and 2.6 MW with one ion source [8]. In the 110 GHz ECRF system, a new unit has been installed in addition to previous three units in 2001. The antenna for this unit is independent from the antenna for previous three units and the injection angle can be scanned toroidally as well as poloidally. Each unit has one gyrotron whose output power is \( \sim 1 \) MW. The torus-injected EC power is plotted as a function of pulse length in Fig.

1. The injected power reached 3 MW, twice as large as before 2000, for 2.7 s and the injected energy reached 10 MJ (2.8 MW for 3.6 s).

The core fuelling with pellets and the plasma shape control are also important for improving the plasma performance. In the multiple pellet injector system [9,10], injection from a high-field-side equatorial plane has been made possible, in addition to a high-field-side top injection, in which efficient fuelling is expected in terms of the effect of ExB drift. The capacity of poloidal coils for control of plasma triangularity \( \delta \) has been expanded by raising their maximum current from 40 kA to 48 kA and the pulse length from \( \sim 5 \) s to \( \sim 10 \) s for 40 kA.

These improvements have enhanced our capability of current drive, profile control and shape control.

3. Improved Performance in High \( \beta_p \) H-mode and Sustainment of High \( \beta_N \)

In the high \( \beta_p \) H-mode, the full non-inductive current drive with \( \sim 50\% \) of bootstrap current fraction at the plasma current of 1.5 MA was realized in 2000 with N-NB injection [7]. The current drive efficiency of N-NB reached \( 1.55 \times 10^{11} \) Am\(^2\)W\(^{-1}\) with increased \( T_e(0) \) of 13 keV by the EC heating. High beta \( (\beta_N = 2.5) \) and high confinement (\( H_{H/2} = 1.4 \)) were also obtained.

In 2001, extension to higher current regime of full non-inductive current drive was pursued by utilizing the increased N-NB power and increased current in poloidal coils for higher triangularity. As a result, the full non-inductive current drive has been achieved in a 1.8 MA high \( \beta_p \) H-mode plasma shown in Fig. 2 [11]. In this discharge, N-NB with 5.7 MW and 402 keV was injected and high beta \( (\beta_N = 2.4) \) and high confinement \( (H_{H/2} = 1.2) \) were maintained. At \( t = 6.5 \) s, the bootstrap current fraction \( (f_{BS}) \) was 50% and the NBCD+ECCD fraction was 50%. The fusion triple product and DT-equivalent fusion gain reached 3.1\times10^{30} \ m^3\text{keVs} \ (n_{0}(0) = 4.2x10^{17} \ m^{-3}, \ T_{\text{e}}(0) = 21.5 \text{ keV}, \ \tau_{\text{e}} = 0.344 \text{ s}) \text{ and 0.185, respectively.}

Other parameters are listed in Table I. The value of fusion triple product renewed remarkably
the previous record under the full non-inductive current drive, 2x10^{20} m^{-3} keVs. The toroidal field and the normalized Larmor radius also approached the reactor-relevant regime.

The sustainable beta in high $\beta_p$ H-mode plasmas is limited by neoclassical tearing modes (NTMs) with $m/n = 3/2$ and/or 2/1. In the discharge shown in Fig. 2, NTM was suppressed by tailoring the pressure and current profiles so that the steep gradient is not located at the mode rational surface, namely $q = 1.5$ and 2. Long sustainment of high $\beta_N$ approaching the current diffusion time scale was undertaken in a lower current regime, where higher triangularity can be maintained for longer time. The improvement of capacity of poloidal field coils enables us to maintain $\delta \sim 0.4$ for 10 s at the plasma current of 1 MA. In a shot E39511, $\beta_N=2.7$ was maintained for 6.5 s or 45 $\tau_E$ without appearance of NTM. To avoid the large heat load to the first wall due to the shine-through power of perpendicular NBs, the plasma density was kept relatively high, namely 67% of the Greenwald density, by injecting pellets continuously. The high density may result in the moderate confinement, $H_{92}=0.9$. Figure 3 indicates sustained $\beta_N$ as a function of its duration. The duration of $\beta_N = 2.7$, which is required for the steady-state operation scenario in ITER [12], was extended remarkably. In another shot E39706, longer sustainment (7.4 s or 60 $\tau_E$) of $\beta_N = 2.7$ was also achieved, though the confinement was lower due to the continuous m/n=3/2 NTM. The cause of appearance of NTM in this discharge is attributed to slightly higher $\beta_N \sim 3$ in the initial phase of heating. In these long-pulse discharges, the density was kept constant and no accumulation of impurities was observed. In shorter pulse length, we can obtain higher current for $\delta$ control coil and hence higher $\delta$. The quasi-steady beta values (maintained for longer than $5\tau_E$) has been raised to $\beta_N = 3.05$ by the increase of $\delta$ to 0.6.

To sustain high $\beta_N$ in a low collisionality ($v^*_{e}$) regime, suppression of NTM is required. A system for real-time NTM detection and EC wave injection has been developed and

![FIG. 3. Progress in sustained $\beta_N$.](image)

![FIG. 4. NTM suppression experiment. (a) Schematic view of the system. (b) Profile of electron temperature perturbation. (c) Waveforms of a typical discharge.](image)
worked successfully. In this system, the location of island center is evaluated by using the electron temperature perturbations (ΔT_e) measured with electron cyclotron emission (ECE) diagnostic as shown in Fig. 4 (b). Then the injection angle of EC wave is determined so that the EC wave deposits its power at the island center taking into account the shape of q=3/2 surface as shown in Fig. 4 (a). Waveforms of a discharge where the NTM was suppressed by using the system are shown in Fig. 4 (c). The 3/2 NTM decreased gradually and it was completely stabilized after the start of EC injection. Even after the turn-off of EC injection, the 3/2 NTM did not appear until t = 10.8 s. The β_N increased from 1.5 to 1.67, which indicated that the energy confinement was improved by the NTM suppression.

4. Current Profile Control in Reversed Shear Plasmas

4.1. Current Hole and High Performance Reversed Shear Plasma

We observed that an equilibrium with a nearly zero current density in the core or the "current hole" persisted stably for several seconds in a reversed shear plasma of I_p = 1.35 MA and q_95 = 5.2 [13] by using upgraded MSE system and equilibrium code. The q profile of higher I_p reversed shear plasma with higher fusion performance was measured and reconstructed in 2001. The profiles of MSE polarization angle, T_i, T_e, q and j are shown in Fig. 5. The MSE polarization angle shown in Fig. 5 (c), which is proportional to B_p/B_n, is very close to zero near the axis and the existence of current hole extended to a normalized radius of 0.3-0.35 was confirmed as shown in q and j profiles in Fig. 5 (b). The T_i and T_e profiles are nearly flat inside the current hole indicating poor confinement in the current hole. It is noted that, however, the flat portions in T_i and T_e profiles extended beyond the current hole radii, which cannot be explained in terms of the small poloidal field. The high temperature plasma (T_i ~ 18 keV, T_e ~ 10 keV) in the current hole was confined by off-axis poloidal field and the ITB. In this discharge, T_e ITB and the current hole were established by EC heating in a low I_p phase with a limiter configuration and they were maintained during the I_p ramp. The discharge terminated in a disruptive beta collapse when q_min became less than 2. Just before the collapse we achieved the DD neutron production rate = 4.6x10^{16}/s, β_N = 1.6, τ_E = 0.89 s, H_{99} = 3.0 and Q_{B_s} = 1.2 with I_p = 2.60 MA and q_95 = 3.3 (see Table 1). Hence it was
confirmed that the current hole was compatible with a high $I_p$, low $q$, high performance plasma. In this discharge, the value of $q_{\text{min}}$ at the time of major collapse has been decreased from $\sim 1.85$ to $\sim 1.7$, and the duration of high fusion performance has been extended. As a result, we were able to maintain the DT-equivalent fusion power gain of 0.8 for 0.55 s. The time derivative of the plasma stored energy was small and the ratio of DT-equivalent fusion power to the absorbed NB power reached 0.8, which is the highest value in JT-60U.

No significant negative central current density has been observed so far. Furthermore, no current is generated with ECCD or NBCD in the center of current hole [14]. These observations of clamping central current density at zero level suggest that the current hole is not a result of a transient zero inductive field near the axis, but rather of some kind of self-organized structure. The response of current profile to various kinds of drive, EC, NB and OH, in a plasma with a current hole is under investigation to establish the method for current profile control in such a discharge.

4.2. Plasma Current Start-up without Use of OH Coil

In a steady-state tokamak reactor, the plasma current is driven mainly ($\approx 70\%$) by the bootstrap current and the rest is by NBCD and/or RFCGD without use of OH coils [15]. Full no-inductively driven plasmas with high bootstrap current fraction have been obtained experimentally [16,17]. However, even in these discharges, the OH coil was used to start the tokamak discharge and ramp up the plasma current. If the plasma start-up and $I_p$ ramp-up, in addition to $I_p$ sustainment, are accomplished without the use of OH coil, the OH coil can be removed from tokamak reactors and the substantial improvement in the economic competitiveness is expected as the machine size is reduced at a higher magnetic field [18,19]. The first demonstration of plasma start-up, $I_p$ ramp-up, and subsequent transition to a high-performance advanced tokamak plasma without the use of the OH coil has been successfully achieved in JT-60U. Waveforms of a typical discharge are shown in Fig. 6. In this discharge, a plasma with $I_p = 0.2$ MA was formed at $t = 2.2$ s by a combination of EC preionization and induction by vertical field coils (VR and VT coils). Subsequently, $I_p$ was ramped up by LHCD until $t = 5$ s when $I_p$ reached 0.4 MA. Finally, NB injection was started to raise the plasma beta. The plasma stored energy was feedback controlled by using the P-NB power, which resulted in an irregular shape in P-NB power waveform in Fig. 6. The increased $\beta_p$ is expected to be effective to raise $I_p$ through the flux provided by the increased current in vertical field coils. The highest plasma current achieved so far in this scenario is 0.7 MA. The ITB and the edge transport barrier (H-mode) were obtained with NB heating. The radius of ITB was large and high confinement, $HH_{\rho_e} = 1.6$, was obtained with $\beta_p = 3.6$ and $\beta_p = 1.6$. The q profile was reversed with the normalized radius of $q_{\text{min}}$ of 0.7 and the current hole existed in
the center. An interesting point is that the current hole seems to be already formed before NB heating, namely during the LHCD phase. This may imply that the current hole will also be formed in future reactors if this scenario is employed and that the control of current profile in a plasma with the current hole is important.

4.3. Current Profile Control in Reversed Shear Plasmas

The active modification of current profile in a reversed shear plasma with a large bootstrap current fraction was demonstrated by using LHCD and NBCD [20]. Waveforms of a typical discharge are shown in Fig. 7 (a). The reversed shear q profile and the ITB were established with P-NB heating during \( I_p \) ramp-up. Injection of N-NB and LH started at \( t = 6.1 \) s and \( 6.3 \) s, respectively. The surface loop voltage started to decrease after the injection of LH and reached -0.2 V. The internal loop voltage evaluated with the MSE measurement was also negative in the whole plasma volume. This indicates that the plasma current was entirely driven by the non-inductive current. The bootstrap current was estimated 62% of the plasma current and the rest was provided by LHCD and NBCD. High confinement (\( H_{95} = 1.4 \)) and high beta (\( \beta_N = 2.2 \)) was maintained at a high normalized density (\( n_e \)-bar/\( n_{GW} = 0.8 \)) under the full non-inductive current drive. The high confinement and high beta are supposed to be due to the large ITB radius as shown in Fig. 7 (b) [21]. The normalized radius of \( q_{min} \), \( \rho_{qmin} \) increased from \( t = 6.6 \) s as shown in the bottom panel of Fig. 7 (a). This is due to the off-axis current drive by LH and is in clear contrast to the case without LHCD where \( \rho_{qmin} \) continued to shrink due to the penetration of inductive current. The radius of \( T_i \) ITB foot, \( \rho_{foot} \), increased as the \( \rho_{qmin} \) was expanded. From the q profiles shown in Fig. 7 (b), it is also noted that the q values inside the ITB, \( \rho < 0.5 \), was decreased. This is not due to the penetration of inductive current but due to the central current drive with N-NB because the loop voltage in this region was negative. The reduction in q in the central region is effective to improve the confinement of high energy particles including \( \alpha \) particles in fusion reactors. Therefore, we have demonstrated that the q profile can be modified favorably by means of combined external non-inductive current drive even with a high bootstrap current fraction.
5. Internal Transport Barrier

Major issues for the application of the internal transport barrier to ITER or demo reactors are (i) ITB formation condition and ITB control, (ii) formation of electron ITB to improve the electron confinement, (iii) sustainment of ITBs, especially the ion ITB, under the reactor-relevant conditions, with dominant electron heating and small central particle fueling, and (iv) impurity accumulation in ITB. Recent JT-60U research on ITB has focused on these items.

Formation of ITB in weak positive shear and reversed shear plasmas was studied by varying the heating power systematically. The relation between the thermal diffusivity ($\chi_t$ and $\chi_e$) and the radial electric field shear ($E_r$ shear) was investigated extensively [22]. It is found that in the weak ITB $\chi_t$ decreases gradually with the increase of heat flux density and the $E_r$ shear. In positive shear plasmas, there was a threshold heating power for formation of weak ITB while no threshold power was found in reversed shear plasmas. In both of positive shear and reversed shear plasmas, the strong (box-type) ITB was formed with a bifurcation or a sudden drop in $\chi_e$ as a function of time or $E_r$ shear. The critical value of $E_r$ shear for the transition to the strong ITB was found to increase with the poloidal magnetic field.

The electron ITB is important to improve the energy confinement in ITER and demo reactors in which the electron heating power is dominant and $T_e$ is expected to be lower than $T_i$. The electron ITB also attracts attention recently since it is considered that the formation mechanism of electron ITB is different from that of ion ITB [23]. The improved performance of EC heating in JT-60U enables us to investigate the formation of electron ITB in a wide range of electron heating power. The electron ITB formation in low $T_e$, low beta plasmas was investigated by injecting the EC wave into a low density plasma with zero or small NB heating power. In reversed shear plasmas, the normalized inverse scale length of $T_e$, R/L$_T$ increased with EC heating power and exceeded 20 with 3 MW heating. On the other hand, in weak positive shear plasmas, R/L$_T$ stayed constant (~10) up to 3 MW. These results indicate that the electron ITB is formed with a small heating power in reversed shear plasmas but is not formed in positive shear plasmas with the present available electron heating power. An electron ITB with high $T_e$ (~25 keV) in a wide region (~30% of plasma minor radius) was obtained with EC heating in a reversed shear plasma sustained by the LH current drive [24]. It is noted that the electron ITB in reversed shear plasmas was already observed in previous JT-60U experiments where LHRF was employed for electron heating and formation of reversed shear [25]. When high power NB heating was applied to EC heated plasmas, the situation was

FIG. 8. HH factor as a function of $T_e/T_i$ in reversed shear plasmas. Squares denote P-NB heated discharges while circles denote P-NB + EC heated discharges.

FIG. 9. Ar exhaust with EC heating in a high $\beta_p$ mode plasma. Profiles of (a) $n_e$, (b) Ar density, (c) $T_e$ and (d) $T_i$ before and during EC heating.
different. Both in weak positive shear and reversed shear plasmas, the electron ITB was formed at the similar location to ion ITB. This implies that the large pressure gradient or \( E_r \) shear accompanied by the ion ITB affects the formation of electron ITB.

As for item (iii), sustainment of ITB in the reactor-relevant regime, effects of electron heating on the ion ITB were mainly investigated [20]. The increased \( T_e/T_i \) may cause an increase of growth rate of ion temperature gradient (ITG) mode and enhancement of ion transport. In a reversed shear plasma with box-type ITBs, heat and particle barriers for ions and electrons were sustained in a regime with \( T_e > T_i \). The HH factor, HH\( \gamma_2 \), is plotted as a function of \( T_e/T_i \) in the central region in Fig. 8. High confinement with HH\( \gamma_2 \approx 1.5 \) was obtained even in a regime with \( T_e > T_i \) in reversed shear plasmas. In E41738, HH\( \gamma_2 = 2.05 \), which is among the highest values in JT-60U plasmas, was obtained with \( T_e/T_i = 1.3 \). In this discharge, EC heating power was 2.6 MW, which corresponded to 44\% of total heating power. The ITB foot and \( \eta_{\text{min}} \) were located around \( \rho \sim 0.6 \) and the box-type ITB with a very narrow ITB layer (5-10 cm) was observed. The efficient heating inside the ITB shoulder with EC may be related to the high confinement. The parameters in E41738 are listed in Table I. On the other hand, the decrease in \( T_i \) gradient and the toroidal rotation shear were often observed when the EC wave was injected into a weak positive shear plasma with parabolic type ITBs. In positive shear plasmas, ion ITB with \( T_e > T_i \) was not obtained. It is noted, however, the NB heating power was limited below ~15 MW in this experiment so as to achieve \( T_e > T_i \) with available EC power. Sustainment of ITB in a regime of \( T_e > T_i \) will be attempted using higher electron heating power, for instance injecting N-NB in addition to EC, in future.

Impurity accumulation is one of the largest issues for application of ITB to reactors. Transport of various impurities (He, C, Ar) and bulk particles have been investigated in weak and strong ITB plasmas [26]. It was found that diffusivities of impurities and electrons were strongly correlated with the heat transport or \( \chi_e \) in a wide range of \( \chi_e \). \( 1 < \chi_e \chi_{NC} < 10 \). In a plasma with a strong ITB with \( \chi_e \chi_{NC} \), heavier impurities tended to accumulate but the peaking factor in the profile shape was smaller than the values from neoclassical prediction for C and Ar. When EC was injected into a high \( \beta_p \) mode plasma after Ar puff, the electron density was reduced and Ar was exhausted. The profiles before and during EC injection are shown in Fig. 9. The density ITB was almost lost during EC heating while the ion ITB was maintained. The profile of Ar was evaluated from the profile of soft X-ray emission. The exhaust of Ar can be explained by the reduction of inward velocity of Ar caused by reduction of \( \eta_e \) gradient in neoclassical transport. This indicates that it is important to control the density gradient for the control of impurity accumulation.

ITB control through perturbation in density or temperature or magnetic field was also attempted [27]. Reduction of density fluctuation and improvement of confinement was observed when a pellet was injected into reversed shear plasmas [26]. The role of rational \( q \) values in the non-local transport bifurcations inside and around the ITB, ITB event, has been investigated [28].

6. ECRF Study

The ECRF current drive is considered a powerful tool for the local current profile control and suppression of NTM. In JT-60U, localized current drive, consistent with the linearized Fokker-Planck calculation [29], was confirmed with the MSE measurement [30]. In ITER or demo reactors, ECCD in high \( n_e \) and high \( T_e \) plasmas is required. Furthermore, the location of ECCD should be varied from the axis to the off-axis (\( \rho < 0.6 \)) region. To evaluate the current drive efficiency \( \eta_{CD} \) in a high \( T_e \) regime, a high \( T_e \) plasma was produced with high power EC heating up to 3 MW in a low density regime (\( n_e \sim 5 \times 10^{19} \text{ m}^{-3} \)). To evaluate \( \eta_{CD} \) experimentally, it is required to avoid MHD instability and sustain high \( T_e \) stably. By optimizing the target current profile and EC deposition location, a stable high \( T_e \) plasma was obtained where \( T_e(0) \sim 23 \text{ keV} \) was maintained for 0.8 s. The evaluated \( \eta_{CD} \) increases with \( T_e \) but the value was lower than the linearized Fokker-Planck calculation in the high \( T_e \) regime (\( T_e \sim 20 \text{ keV} \)). This is considered due to the negative toroidal electric field induced by the large
FIG. 10. Normalized EC current drive efficiency $\xi$ as a function of normalized radius for deposition locations at the high-field-side (HFS) and the low-field-side (LFS). Open and closed circles denote measured values and solid and dotted lines denote the results of linearized Fokker-Plank calculation.

EC driven current density, $j_{EC}$ [31]. To evaluate $\eta_{CD}$ precisely, a higher density plasma with high $T_e$ is required to reduce $j_{EC}$. In off-axis CD, the effect of trapped electrons was found by changing the EC deposition position in the poloidal direction as shown in Fig. 10. The dependence on $n_e$ and $T_e$ was also investigated and a strong dependence on $n_e$ was found in the range of $(1-1.8) \times 10^{19} \text{ m}^{-3}$. This is considered due to the enhanced coupling to increased number of fast electrons.

7. Alfvén Eigenmode

In the Alfvén eigenmode (AE) study, a new model has been found consistent with the observed frequency sweep and saturation during N-NB injection in reversed shear plasmas [32]. In the discharge shown in Fig. 11, the hydrogen N-NB of 360 keV and 4 MW was injected into a reversed shear plasma with $B_t = 3.7$ T and $I_p = 1.3$ MA. Magnetic fluctuations in the AE frequency range have been observed during N-NB injection. Only the $n=1$ mode was observed due to relatively small fast ion beta ($<0.2\%$). The frequency chirp both in upward and downward directions was observed. The dotted lines are frequencies of the shear AE near the zero shear region of reversed shear plasma, which is called the reversed-shear-induced AE (RSAE). The frequency of RSAE depends strongly on the value of $q_{min}$ and it can explain the observed frequency chirp very well as shown in Fig. 11. The AE frequency saturates at $t = 6.65-6.85$ s and the amplitude was enhanced. This can be explained by the transition of RSAE to TAE which occurs at $q_{min} \sim 2.5$. To avoid these large amplitude TAEs, it is suggested to operate a reversed shear plasma outside the range of this transition, for instance $2.4 < q_{min} < 2.7$. The RSAE can also explain the previously observed rapid frequency sweeping modes with $n>1$ in JT-60U ICRF heated reversed shear plasmas [33].

8. H-mode and ELM Study

In the H-mode pedestal regime, important points are to maintain high confinement in the high density regime and to reduce the ELM heat load to the target plates.

In Ar-seeded H-mode plasmas, higher confinement is obtained in the high density regime than no impurity-seeded plasmas. A higher pedestal ion temperature is obtained with a fixed pedestal pressure in Ar-seeded plasmas where the ion density is lower, and this results in a
higher core temperature and higher confinement though the profile stiffness [34]. The reduction of instability growth rates with an increase of $Z_{\text{eff}}$ may also be involved [35]. In JT-60U, a configuration with the outer strike point located on the dome-top ("dome-top configuration") was found to enhance the confinement further [36]. With this configuration, giant ELMs disappeared in the high density regime and the heat load due to ELMs decreased to 1/10 compared to the low density case. In 2002, high power N-NB injection extended the density regime to $n_{\text{e}}/n_{\text{GW}} = 0.95$ keeping $HH_{y_2} = 0.9$ as shown in Fig.12. The parameters of this discharge (E41536) are listed in Table I.

It is very important to raise the pedestal pressure in order to enhance the global stability and confinement properties. We have found that the pedestal pressure increased by a factor of 2-2.5 with total (or core) poloidal beta in high triangularity ($\delta$=0.45) plasmas with type I and type II ELMs as shown in Fig. 13 [37]. As a result, high confinement of $HH_{y_2} = 1.1$ was achieved at $n_{\text{e}}/n_{\text{GW}} = 0.7$ in a pellet injected discharge. An increase in the pedestal pressure was not caused by an increase of the pedestal width but by an increase of the pressure gradient (\alpha parameter) at the pedestal. In contrast, the pressure and the \alpha parameter at the pedestal stayed almost constant in low \delta plasmas. The pedestal pressure in high \delta H-mode plasma increases gradually in time even with constant $\beta_p$. This is considered to be due to the change of local magnetic shear by the edge bootstrap current.

The type II ELM with a small heat load onto the divertor plates, appears in a high \delta and high $q_{95}$ region. By increasing \delta up to 0.6, the type II ELMy regime has been expanded to a lower q regime ($q_{95} = 3.8$). In this region, the type II ELM was maintained during pellet injection while type I ELM appears after each pellet with a smaller \delta(-0.45) or $q_{95}$.

The dynamics of density collapse due to type I ELMs was measured using a reflectometer [38]. The time scale (100-350 \mu s), penetration depth (11 cm or twice the pedestal width) and a poloidally asymmetric structure of the collapse were observed. It was revealed that the ELM heat load is mainly carried by the convective transport using SOL Mach probes and IRTV.

Understanding of SOL flow pattern and the driving mechanism is crucial for particle and impurity control in divertor. A new reciprocating Mach probe has been installed above the high-field-side baffle plate and SOL measurement at three locations (high-field-side, low-field-side and X-point) has been made possible. The SOL flow measurements revealed the importance of ExB drift flow [39]. As for the plasma-surface interaction, tritium retention of the first wall materials has been studied by analyzing the carbon tiles [40, 41].
TABLE I. Parameters of typical discharges

<table>
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<th>Shot</th>
<th>39713</th>
<th>40259</th>
<th>37964</th>
<th>41738</th>
<th>41536</th>
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<tbody>
<tr>
<td>mode</td>
<td>High $\beta_p$, Full CD</td>
<td>RS, $Q_{\text{det}}=0.8\times0.55$s</td>
<td>RS, Full CD</td>
<td>RS, High HH with $T_e&gt;T_i$</td>
<td>Ar-seeded H, Dome-top, High $n_e$</td>
</tr>
<tr>
<td>time [s]</td>
<td>6.5</td>
<td>7.22</td>
<td>7.24</td>
<td>6.5</td>
<td>9.0</td>
</tr>
<tr>
<td>$I_p$ [MA]</td>
<td>1.795</td>
<td>2.6</td>
<td>0.905</td>
<td>1.3</td>
<td>1.20</td>
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<td>0.36</td>
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<td>4.3</td>
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<td>0.95</td>
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<td>0.62</td>
<td>0.56</td>
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<td>-</td>
<td>&gt;0.38</td>
<td>0.12</td>
<td>-</td>
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<tr>
<td>$P_{\text{in}}/P_{\text{det}}$</td>
<td>0.54</td>
<td>-</td>
<td>-</td>
<td>0.43</td>
<td>0.8</td>
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9. Summary and Discussions

Enhanced capabilities of N-NB and ECH enabled the research approaching the reactor-relevant regime including high $\pi T$ full non-inductive CD, NTM stabilization, current profile control in reversed shear plasmas, high confinement reversed shear plasma with $T_e>T_i$ and high confinement at the high density. Achieved parameters of typical discharges are listed in Table I. A new operation scenario has been established in which a plasma with high bootstrap current fraction and ITBs is produced without using OH coil. Significant advance in physics understanding on a high temperature plasma has been obtained including discovery of the current hole, measurement of ELM dynamics and a new type of AE mode.

Studies on the ITB and the edge pedestal were intensively carried out with emphasis on reactor-relevant conditions. Impurity accumulation related to large density gradient in strong ITBs in a reversed shear plasma and degradation of ITB by ECH in a weak positive shear plasma have been found. In order to decide what type of ITB and $q$ profile are suitable in reactors, experiments with low particle fueling condition, in addition to the dominant electron heating, will be important.

In future, we will continue to develop discharges with higher integrated performance with improvement of N-NB, pellet and EC. Extensive physics study will also be performed in enhanced collaboration with other institutes.
Acknowledgements

The authors wish to express their thanks for the continuous efforts of the engineering and technical staff at JAERI contributing to the JT-60U project. The scientific contributions under international and domestic collaboration programs are also greatly appreciated.

References

[22] SAKAMOTO, Y., et al., this conference, EX/P2-08.
[28] NEUDATCHIN, S.V., et al., this conference, EX/P2-06.
[38] OYAMA, N., et al., this conference, EX/S1-1.
Overview of JT-60U Results toward High Integrated Performance in Reactor-Relevant Regime

T. Fujita and the JT-60 Team
Japan Atomic Energy Research Institute, Naka Fusion Research Establishment

19th IAEA Fusion Energy Conference
Lyon, France, Oct. 14-19, 2002

Enhanced collaboration with domestic and foreign universities/laboratories

- Many institutions contribute to the JT-60 program.
- As a result, five papers from collaborators in this conference:
  Tansbri: EX/P2-11, Hatayama: TH/P2-15, Hill: EX/P2-03,
  Neudatchin: EX/P2-08, Takase: submitted to PD.

In JAPAN

From abroad
ASIPP (China), Ecole Polytech (Switzerland), EFDA-IET (EU),
GAT (USA), JET Ltd. (UK), KSTAR (Korea), KFA Juelich (Germany),
Kurchatov Inst. (RF), LANL (USA), MPR-Garching (Germany),
MIT (USA), ORNL (USA), PPPL (USA), SWIP (China), TRODUCTION (RF),
U. Stradby (UK)

Objectives/Strategy of JT-60 program

- ITER Physics R&D
- Advanced Tokamak Concepts for ITER & DEMO
- High integrated performance; high values of $p_n$, $HH_{95}$, $n_{BS}$, $f_{CD}$,
  $n/n_{GW}$, fuel purity, $P_{rad}$/P$_{abs}$
- Other conditions are also required in reactors:
  $p^*$, low $v^*$, $T_e$-$T_i$ small central fueling, small ELM, etc.
  Reactor-relevant regime
- Recent JT-60U experiments are mainly devoted to:
  - Extend regime toward high integrated performance
  - Investigate issues for reactor-relevant regime

Contents

1. Improved control/heating/CD systems
2. Sustainment of high performance and high $\beta_N$
in high $\beta_p$ H mode
3. Extended region and control of current profile in reversed shear plasmas
4. Internal transport barrier research towards reactors
5. H-mode pedestal and divertor/SOL research towards reactors
6. Summary
1. Improved control/heating/CD systems

- N-NB
  - 5.2 MW (350 keV, 0.8 s) -> 6.2 MW (381 keV, 1.7s) through improvement of beam deflection and uniformity of source plasma.
  - 10 s injection was achieved (355keV, 2.6MW). [Umeda, CT-6Rd, Wed.]
- 110 GHz ECRF
  - 4th unit (gyrotron) has been installed. [Sakamoto, K., CT-7Ra, Wed.]
  - 3 MW for 2.7 s.
  - ECCD in high $T_e$ [Suzuki, EXW-2, Sat.]
- Triangularity ($\delta$) control coil
  - $\delta=0.45/1\text{Max5s}$
  - $\delta=0.60/1\text{Max5s}$ or $\delta=0.45/1\text{Max10s}$

High fusion performance with full CD

- High $\delta=0.34$ at high $I_p=1.8$ MA and high power N-NB (5.7 MW)
- $\beta_p=2.3-2.5$, $HH_{cp}=1.2$, $\eta_0(0)\tau_1 T_1 (0) = 3.1 \times 10^9 \text{ m}^2 \text{s} \text{keV}$ (1.5 times previous record), full non-inductive CD (BS:50%)
- NTM was suppressed by tailoring $p(r)\omega(r)$ in low collisionality regime ($v_\parallel=0.02$ - ITER).

$\beta_N = 2.7$ sustained for 7s

- In higher $\delta$ (-0.45), long sustainment ($6.5-7.4s$, $45\pi-60\pi_0$) of high $\beta_N(2.7)$ was achieved through improvement of capability of $\delta$-control coils.
- $v_\parallel=0.1$, $q_{95}=3.3-3.6$.
- The attainable $\beta_N$ was limited by 3/2 NTM.

Suppression of NTM is a key
3. Extended region and control of current profile in reversed shear plasmas

In RS, current profile is crucial for stability and confinement.

The properties of AE modes in RS depend strongly on $q_{\text{min}}$. $\rightarrow$ Takechi, EX/W-6, Sat.
Control of current hole radius is required

- A large current hole will be a problem for confinement of α particles in reactors.
- Large radius of $q_{min}$ is favorable to obtain large ITB radius and high confinement.

Independent control of the current hole radius or $q(0)$ and the $q_{min}$ radius is required for optimization of $\langle j \rangle$ in RS.

- CD in the center of current hole is difficult ("current clamp"). Large negative $E_{\phi}$ in the current hole cancels driven current?

$\rightarrow$ CD outside the current hole

4. Internal transport barrier

Issues for application of ITB to reactors:
- Electron ITB
- Ion ITB with electron heating
- Impurity accumulation (control of particle transport)

- ITB formation condition and ITB control
  -> Sakamoto, Y., EX/P2-08, Wed.
- ITB event -> Neudatchin. EX/P2-06, Wed.

Current profile control demonstrated in RS

- Full non-inductive CD with BS(62%), LHCD and N-NBCD.
- The radii of $q_{min}$ and ITB-foot expanded by peripheral LHCD. $HH_{\phi}=1.4$, $\beta_n=2.2$, $n/I_{dne}=0.8$ due to large ITB radius.
- Reduction of the central $q$-value (outside the current hole) and current hole radius by central N-NBCD.

$\rightarrow$ approaching weak RS q-value

Electron ITB is obtained together with ion ITB

- Electron ITB is important to improve confinement in ITER/Demo, in which $T_e=T_i$.
- With no/lowlow NB power, $T_e$ ITB was formed in RS ($P_{EC} \sim 1$ MW) but not formed in PS ($P_{EC}$ up to 3 MW).
- With high NB power, $T_e$ ITB was formed at the similar location to $T_i$ ITB both in PS and RS. $\chi_e$ decreases with the decrease of $\chi_i$.

$\rightarrow$ Containment of ion ITB under electron heating is important

Sakamoto, Y., EX/P2-08, Wed.
Ion ITB with electron heating

- In RS, ion-electron ITBs were maintained, \( HH_{\text{p}} \rightarrow 2 \) with \( T_e \rightarrow T_i \).
- In PS, decrease of grad-\( E_z \) followed by decrease of grad-\( T_e \) was observed with electron heating. ECH seems to affect \( E_z \). Higher heating power may be required to sustain \( E_z \) gradient and ITB.

\[ \text{Reversed Shear} \begin{array}{c}
\text{Positive Shear}
\end{array} \]

\[ \text{Positive shear} \]

5. H-mode pedestal and divertor/SOL

Issues for application of H-mode to reactors:
- Pedestal/Confinement degradation with density
- ELM dynamics for physical understanding of ELM
- SOL flow and impurity shielding

Particle transport in ITB plasmas

- Particle transport was strongly related to thermal transport.
- He: flat, high ITB plasmas.
- On weak ITB in PS, ECH caused decrease of \( n_e \) and Ar exhaust. Ar exhaust can be explained by reduction of \( V_{th} \) due to reduction of \( n_e \)-gradient. On strong ITB in RS, no effects of ECH on \( n_e/Ar \).
- Low central fueling will be favorable.

\[ \text{Positive shear} \]

High confinement at high \( n_e \) by high \( \delta / \text{Ar} \)

- Pedestal \( \beta_p (P_{\text{PED}}) \) was increased with total \( \beta_p \) in high \( \delta \) while \( P_{\text{PED}} \) was almost constant in low \( \delta \).
- Ar-seeded H-mode plasma with “dome-top” configuration extends to higher \( n_e \) regime (\( -n_{GW} \)).
- Giant ELMs disappear in both regimes.

\[ \text{Positive shear} \]

\[ \text{High confinement at high } n_e \text{ by high } \delta / \text{Ar} \]

\[ \text{High confinement at high } n_e \text{ by high } \delta / \text{Ar} \]

\[ \text{Kamada, EX/P2-04, Wed.} \]

\[ \text{Hill, EX/P2-03, Wed.} \]
ELM dynamics and SOL flow revealed by improved diagnostics

- Density collapse at type I ELMs was measured with high time resolution reflectometer. Poloidally localized at LFS. [Oyama, EX/S1-1, Tue.]

- SOL flow was measured at three locations using Mach probes. Increased parallel flow by "puff &pump" was observed. The change in SOL flux is large at HFS. [Asakura, EXD1-3, Thu.]

Summary and future directions

- Enhanced N-NB and ECH capability enables the research approaching the reactor-relevant regime.
  - high mT full CD, NTM stabilization, ECCD, current profile control, ITB/pedestal studies

- Advance in physics study
  - current hole, new type AE (RSAE), ELM dynamics, SOL flow

- Newly developed operation mode
  - high bootstrap current plasma without use of OH coils

Future:
- Higher integrated performance with improvement of N-NB, pellet and ECH.
- Extensive physics study in enhanced collaboration with other institutions.
1.2 High Performance Tokamak Experiments with Ferritic Steel Wall on JFT-2M


1)Naka Fusion Research Establishment, Japan Atomic Energy Research Institute
2)Tokyo Institute of Technology, 3)National Institute for Fusion Science,
4)Himeji Institute of Technology, 5)Hokkaido University
E-mail: tsuzukik@fusion.naka.jaeri.go.jp

Abstract. In the JFT-2M tokamak compatibility between plasma and the low activation ferritic steel, which is candidate material of a fusion reactor, has been investigated step by step. We have entered the 3rd stage of Advanced Material Tokamak Experiment (AMTEX), where the inside of vacuum vessel wall is fully covered with ferritic steel plates (Ferritic Inside Wall; FIW). The effects of FIW have been investigated on the plasma production, the impurity release, the operation region, and H-mode characteristics. No deteriorative effect has been observed up to now. High normalized beta plasma of \( \beta_n \approx 3 \) having both internal transport barrier and the steady H-mode edge was obtained. Remarkable reduction of ripple trapped loss from 0.26 MW/m² (w/o ferritic steel) to less than 0.01 MW/m² was demonstrated due to optimization of the thickness profile of FIW. The effect of the localized ripple was also investigated with additional ferritic plates outside the vacuum vessel. In parallel with AMTEX, advanced and basic research for the development of high performance plasma such as evaluation of fluctuation induced particle flux and development of advanced fuelling (compact toroid injection) are also performed on JFT-2M.

1. Introduction

A low activation ferritic steel (such as F82H [1]) is a leading candidate material for a fusion demonstration reactor [2] because of its better properties for heat and neutron load, compared to austenite stainless steel. In addition, application of ferromagnetic material is planned in ITER aiming at reducing toroidal field ripple [3-5]. However, the effect of ferromagnetism on plasma control, stability and confinement is not well understood. Vacuum properties could also be problems because it easily rusts in the air [6]. Thus compatibility with plasma, in view of ferromagnetic effects and vacuum properties should be investigated. In small tokamak, HT-2, the ferritic steel, F82H, was installed inside the vacuum vessel. The compatibility of the ferritic steel was well demonstrated for ohmic heating plasma [7]. To investigate the compatibility with higher performance plasma, the Advanced Material Tokamak Experiment (AMTEX) program is being performed, in the medium size tokamak JFT-2M (R=1.31 m, a≤0.35 m, k≤1.7, B₉≤2.2T) [5,8-11]. In the last IAEA Fusion Energy Conference, results of the 1st stage of AMTEX were presented, where ferritic plates (FPs) were installed between the vacuum vessel and toroidal field coils, aiming at reducing the toroidal field ripple (similar configuration as ITER) [8,9]. In the second stage, the FPs of thickness 7 mm were installed along the whole toroidal circumference in the low field side above and below the horizontal ports. They covered 20% of the inside area of the vacuum vessel. No adverse effects of ferritic steel on the plasma operation and stability was observed at least for the normalized beta up to 2.7 [9-11]. With these encouraging results, we have entered the 3rd stage of AMTEX. The main purpose of this stage is 1) an investigation of the compatibility between plasma and a full covering ferritic inside wall (FIW) as a simulation of blanket wall of the demo-reactor and 2) a demonstration of the significant reduction of the toroidal field ripple by optimizing the thickness profile of FIW. The setup and results are presented from Section 2 to Section 6.
EX/C1-1

Fig. 1. Picture of inside vacuum vessel after installation of FIW, magnetic sensors and graphite tiles.

In parallel with the AMTEX program, advanced and basic research for development of the high performance tokamak plasma are also performed with an MSE polarimetry, a heavy ion beam probe and a compact toroid injector etc. These results are shown from Section 7 to Section 9.

2. Design and Installation of Ferritic Steel

The ferritic inside wall (FIW) was installed along the vacuum vessel, keeping a distance of 30 mm from the inside surface of the vacuum vessel, so that the similar plasma configuration and shape as before can be obtained. The thickness of the ferritic steel is determined to meet both magnetic effect and ripple reduction. When the frequency of magnetic fluctuation increases, the effect of the eddy current increases, and thus, the effect of ferromagnetism are weakened [10]. The average thickness is determined to be 8 mm so that the effect of eddy current exceeds the ferromagnetic effect for typical frequency of tearing mode on JFT-2M (>6kHz). Furthermore, reduction of toroidal field ripple for almost 1/4 as large as that without FIW was conducted by optimization of thickness profile (6mm, 8mm, and 10.5mm) of FIW. Most of the magnetic sensors were installed on plasma side of the ferritic steel. Part of them were installed behind FIW to investigate shielding effect of the ferritic steel. Graphite tiles were installed on FIW keeping a distance of 50 mm. The configuration is similar to the previous one, namely, the high-field side was fully covered with the graphite tiles and the divertor region and low field side were covered discretely with them. Thus, main plasma-facing component is the graphite tiles and FIW is exposed to peripheral plasma.

Precise 3-D magnetic field measurements inside the vacuum vessel were carried out with rails, a vehicle, and hole elements installed inside the vacuum vessel. The magnetic field was measured for whole toroidal angle (192 points in toroidal direction and 5 points in poloidal direction, 3-D component (X, Y, Z) for each points). The results are shown in Fig. 2. The experimental results are close to the calculated one. The designed value of the toroidal field ripple has been realized. It should be noted that the toroidal periodicity of the magnetic field is broken due to the limitation of installation area of the FIW by interferences with the existing component such as neutral beam ports. The measurement also indicated that a shift of the central axis of the vacuum vessel from that of the toroidal field coils is ~3 mm. Since the FIW was installed along the vacuum vessel, the shift of the vacuum vessel induces low-n error field, which might induce locked mode [12]. However, in the case of m/n=2/1 component (B_{21}), which is regarded as the most dangerous mode for locked mode, induced error field due to the shift of ~3 mm is less than (B_{21}/B_c=4x10^{-5}). This is much lower than the allowable limit for a joule heating plasma (B_{21}/B_c=2x10^{-4}) [12,13].
3. Wall Conditioning

The same procedure as the second stage was applied on the FPs, namely (1) removal of oxide layer by machining, (2) degreasing, and (3) 350°C baking for 20 hours [14]. Impurity release was not problem in the second stage [10]. However, it took 6 months for the installation (10days for the 2nd stage), and thus, the oxidation during air exposure might degrade the vacuum properties. The vacuum properties of the ferritic steel, which had been exposed to air for 2 months and slightly oxidized, were investigated in a test-stand, prior to the tokamak experiment. It has been confirmed that the out gas rate from the ferritic steel is sufficiently low for the FIW (10⁻⁸ Pa·m³/sm²) [15].

The usual procedure was applied in the initial pumping just after the installation of the FIW, namely, baking at 120 °C for 3 weeks and Taylor discharge cleaning (TDC) for 30 hours. The obtained base pressure was 6x10⁻⁶ Pa, which is same as that before the installation of FIW, as is predicted from the test-stand experiment. As for a discharge cleaning, the Taylor discharge cleaning was performed without changing discharge conditions, namely, working gas: H₂, pulse length: ~10 ms, interval: 0.7 sec, pressure: 2x10⁻⁴ Pa. The partial pressure of H₂O, CO and CO₂ clearly increased when the TDC ignited and the out gas rate is 1.5 times as large as the case without ferritic wall. It means that the TDC is effective to remove oxygen impurity though the ferritic wall affects the magnetic structure. The glow discharge was also ignited and utilized to reduce hydrogen recycling.

4. Plasma Production and Impurity Release

To evaluate the effects of FIW on the plasma control, the magnetic field caused by the FIW was calculated with the equilibrium code including the effect of ferritic steel. Figure 3 shows schematic cross-sectional view of the magnetic field. The FPs are magnetized in the direction of poloidal field (typically, specific magnetization is 2 ~ 3). In a case of the elongated plasma, direction of the magnetization of ferritic steel is opposite in diverter region. Thus stray filed forms vertical field as shown in the figure. It weakens the vertical field by ~10 %. In addition, this field affects magnetic sensors located on FIW. The JFT-2M uses magnetic probes and 8 flux loops for plasma control and equilibrium calculation. The magnetic field caused by FIW is in the order of several percentage of the averaged poloidal field at the probe position. If the separatrix (or last closed flux surface) is estimated without considering the effect on the magnetic sensors, major radius of the separatrix is ~1 cm smaller than the real one. The vertical field from FIW can be compensated by increasing vertical field by ~10%. The shift of plasma position can also be compensated by changing the setting value of the plasma control system.
EX/C1-1

As is predicted from the calculation, tokamak discharges were obtained without a marked change in the plasma control system. Increase in vertical field by \( \sim 10\% \) compared to the value wo FIW was observed, which is consistent with calculated results.

As a measure of impurity release during the discharge, total radiation loss is shown in Fig. 4 for limiter discharges at \( B_t=1.3 \) T and \( I_p=200 \) kA. The total radiation loss for initial 1 month was slightly higher than the cases without FPs and partial coverage. After experiments of 1 month, it was decreased by \( \sim 20\% \) of the previous level. The oxygen line intensity measured by the visible spectroscopy became almost half of the previous level. The metal impurity measured by a vacuum ultraviolet spectroscopy was under the detection level. Thus, it is concluded that impurity release from the ferritic steel is not large in this experimental condition (main plasma facing component is graphite tiles), and improvement of the oxygen level could be attributed by the replacement of the graphite tiles.

After a series of experiment, boron coating was carried to obtain higher performance plasma by reducing impurities in the plasma. The experiments related to plasma stability and confinement were mainly carried out after the boron coating.

5. Investigation of Ripple Loss [16]

The ripple loss was evaluated with an infrared TV system (IRT). Fast hydrogen ions are supplied by neutral beam injection (NBI) with its primary energy of 36 keV (CO-direction on \( I_p \), tangentially on \( B_t \)). The injection power is about 500 kW. The maximum heat flux due to the ripple trapped loss decreased from 0.26 MW/m\(^2\) (w/o ferritic steel) to less than 0.01 MW/m\(^2\) (with FIW) in optimized case (Bital=1.3T).

As indicated in Fig.2, the toroidal field ripple is not reduced uniformly because of the limited installation of FIW. Such situation might also occur in a demo-reactor. In addition, large local ripple could be used for low energy beam injection and He exhaust [17-19]. Thus, understanding of the behavior of fast ions with complex ripple, including the large local ripple, is important. To investigate these effects experimentally, the ferritic steel plates outside the vacuum vessel were used to induce strongly localized ripple. The results show that the ripple loss depends on ripple well structure, e.g. the thickness of the ripple well. In other words, whole structure of the magnetic filed has to be considered to analyze the ripple loss behavior. The experimental results were almost consistent with the newly developed Fully three Dimensional magnetic field Orbit-Following Monte-Carlo (F3D OFMC) code including the three dimensional complex structure of the toroidal field ripple and the non-axisymmetric first wall geometry. From the F3D OFMC calculation, the total loss of fast ions with FIW is about 1/3 as large as that with only TFC.

6. Effect of the Ferritic Steel on Plasma Performance

Figure 5 shows Hugill diagram after the FIW installation and boron coating. The density can be increased near Greenwald density and the safety factor, \( q \), can be decreased around 2. It was reported that the region, where collapse due to a tearing mode occurs, exists around \( q=3 \), \( n_e \sim 1 \times 10^{19} \) m\(^{-3}\) [11,20] as indicated in the figure. This region didn’t change by the ferritic steel installation, which means that the effect of FIW on tearing mode is negligible. It should be noted that the ratio of minor radius of the resonance surface \( r_e \) and the FIW \( d \) is \( d/r_e \sim 1.6 \). The calculation code to clarify the wall effect is under development.

Growth rate of vertical instability was measured by switching off the feed-back control during the discharge [21]. The ferritic steel makes vertical instability unstable in qualitative manner, because the vertical shift of plasma causes unbalance of magnetization, which enhances the shift. However, this effect is limited within a few percentages. On the other hand, FIW acts as an additional conductive wall, which makes the instability stable.
Experimental results shown in Fig. 6. The growth rate didn’t change by the FIW installation presumably due to the compensation of above effects.

In the single-null divertor configuration, H-mode was obtained with similar conditions as before. The threshold power for the L-H transition is 420kW at B_T=1.3 T, I_p=200 kA, and n_e=3x10^{19} m^{-3}. It is almost the same as the value before the FIW installation (440 kW) and comparable to the value from scaling low ~ 500kW [22]. Averaged H factor (H_{95}) is ~1.8 for ELMy H-mode, which is also comparable to the previous results. Thus, no adverse effect of FIW on plasma confinement and stability has been observed up to now.

7. High Beta Experiment

In parallel with the AMTEX program, the advanced and basic studies for development of high performance plasma are performed on JFT-2M. As for the H-mode study, an attractive new operating regime, which is named “High Recycling Steady” (HRS) H-mode, have been discovered. It was first observed with deuterium saturated boronized wall. Important features of this mode are, (1) the steady-state H-mode edge condition at high density with good energy confinement, (2) the complete disappearance of giant ELMs, and (3) the
compatibility with the internal transport barrier (ITB) [23]. The HRS was observed in high density region \( n_e/n_{GW} > 0.4 \), including the low safety factor region \( q_{95} \sim 2 \). The formation of ITB was typically observed in low \( B_T \) (<1.1T) conditions during balance neutral beam injection of \( \sim 1.4 \) MW (full power). The ion temperature measured by charge exchange recombination spectroscopy (CXRS) clearly peaked, which indicate the formation of ITB around \( r/a = 0.1 - 0.2 \) as shown in Fig. 7a. The q-profile also changes from monotonic one to zero/weak shear in the plasma core region. To show the effect of the ITB formation on normalized beta, toroidal beta values are plotted against normalized current \( (I_p/aB_T) \) in Fig. 7b. The slope of the figure corresponds to normalized beta \( (\beta_n) \). In the cases without ITB, the normalized beta is limited less than 2.5. This value is same as previous results, without FIW and boronization. The \( \beta_n \) increased to 3.0 with the ITB as shown in box-plus in the Fig. 7b. This increase can be attributed to the improvement of the core confinement with keeping the steady H mode edge. The experiment was performed in the presence of FIW described above. These results are also import as the demonstration of compatibility of such FIW with high normalized beta plasma up to 3.0.

8. Estimation of Particle Flux in H-mode

The estimation of fluctuation induced particle flux at L-H transition is important for understanding of H-mode behavior. To investigate the flux, the fluctuations of the potential and the density were measured by Heavy Ion Beam Probe (HIBP) [24]. Poloidal wave number of the fluctuation was evaluated by taking several sample volumes located on almost the same magnetic flux surface at the same time. Using the measured wave number \( 1-2 \) cm\(^{-1}\) in L mode plasma, the radial particle flux induced by the fluctuation is evaluated. In the frequency range up to 100 kHz, it was found that the particle flux in L mode plasma is dominated in the frequency range of 20-60kHz. After the L-H transition, the flux of this frequency range almost disappeared. In H-mode phase, the Doppler-shifted fluctuation possibly exists in the frequency range of 100 kHz and more, but it couldn’t measured mainly due to cross talk of the amplifier.

9. Advanced Fuelling (Compact Toroid Injection)

Compact toroid (CT) injection is an advanced method of the particle fueling into the plasma, and being investigated on JFT-2M. Prior to the injection experiment in 2002, the CT injector was modified in 2001, aiming at improving injection efficiency, i.e., a focus corn was replaced to straight type at the nozzle and the shape of the pre-compression region was
modified to reduce focusing effect. In CT injection experiments into JFT-2M plasma, rapid increase of electron density within 60 μs was clearly observed as shown in Fig. 8a. This is the first demonstration of the density increase in such a time scale. Figure 8b shows profile of soft X-ray. The intensity of weak field side initially increased and is kept for ~100 μs. Then it propagated to the plasma center. The fuelling efficiency within the rapid phase (60 μs) was estimated to be 25% by \((Δn_e×V_{\text{plasma}})/(n_{CT}×V_{CT})\). In the case of other discharges, the time constant of the density increase is ~500μs. This time scale is much shorter than that with only gas puff. The mechanism of density increase with different time scale has been unclear so far.

In addition, we have a plan to inject CTs vertically into the JFT-2M tokamak by using a curved drift tube for the improvement of CT injection efficiency. The proof-of-principle experiments on CT transport with a curved drift tube have been successfully carried out at the Himeji Institute of Technology [25].

Summary

The compatibility of low activation ferritic steel with plasma has been investigated in the JFT-2M with respect to ferromagnetic effects and impurity release. The vacuum vessel was fully covered with the ferritic steel as a simulation of blanket wall (FIW: Ferritic Inside Wall). The base pressure with FIW was almost same as the cases without ferritic steel. Tokamak discharge was obtained without marked change in plasma control system as is consistent with the expectation. The radiation loss and oxygen line intensity decreased for 20% probably due to replacement of graphite tiles. The metal line intensity measured by a vacuum ultraviolet spectroscopy was under the detection level. As a measure of plasma stability and confinement, the operation region, threshold power for L-H transition and H factor were investigated, showing similar results as the case before the FIW installation. During the study, attractive operating regime named High Recycling Steady (HRS) H-mode, have been discovered. This mode has characteristics as follows; the steady-state H-mode edge condition at high density with good energy confinement, the complete disappearance of Giant ELMs, and the compatibility with internal transport barrier (ITB) [23]. The normalize beta \((\beta_n)\) increased up to ~3.0, which is the highest value of JFT-2M by the combination of steady H-mode edge and improvement of core confinement. This result is also important as a demonstration of compatibility of ferritic steel wall with high-normalized beta plasma.

As for the ripple loss, remarkable reduction from 0.26 MW/m² (w/o ferritic steel) to less than 0.01 MW/m² (with FIW) was demonstrated in optimized case (B_r=1.3T). To investigate the effect of complex field produced by the ferritic steel, the ferritic steel outside the vacuum vessel was additionally installed for inducing strongly localized ripple. The results show that the ripple loss depend on ripple well structure, e.g. the thickness of the ripple well. The experimental results were almost consistent with the newly developed Fully three Dimensional magnetic field Orbit-Following Monte-Carlo (F3D OFMC) code including the three dimensional complex structure of the toroidal field ripple and the non-axisymmetric first wall geometry.

As for the H-mode study, the fluctuation induced particle flux was evaluated with Heavy Ion Beam Probe (HIBP). A peak of the flux in L mode plasma is observed in the frequency range of 20-60kHz. After the L-H transition, the flux of this frequency range almost disappeared. For the development of advanced fuelling method, compact toroid injection experiments were carried out. Rapid increase in line averaged density with time scale of 60 μs was clearly demonstrated.

Acknowledgement

The authors appreciate Prof. K. Kawahata of National Institute for Fusion Science (NIFS) and
EX/C1-1

Prof. S. Okajima of Chubu university for adjustment of FIR. They thank to Prof. K. Ida of NIFS for measurement of CXRS. They appreciate Dr. M. Azumi of JAERI for development of equilibrium code with ferritic steel. The authors are indebted to Dr. H. Kishimoto, Dr. S. Matsuda, Dr. A. Kitsunezaki, Dr. M. Shimizu, Dr. H. Ninomiya and Dr. M. Kikuchi of JAERI for their continuous encouragement and support.

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High Performance Tokamak Experiments with Ferritic Steel Wall on JFT-2M


1)Naka Fusion Research Establishment, Japan Atomic Energy Research Institute, 2)Tokyo Institute of Technology, 3)National Institute for Fusion Science, 4)Himeji Institute of Technology, 5)Hokkaido University, 6)University of Tokyo, 7)The Institute of Physical and Chemical Research, 8)Jose University of Education, 9)Yokohama National University

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Research objectives on JFT-2M (medium size tokamak: R = 1.3 m, a = 0.35m, B_T < 2.2T)

- Investigation of compatibility of low activation ferritic steel with plasma
  - It is candidate material for fusion reactor because of better properties for thermal and neutron load.
  - However, it is ferromagnetic material, and thus, the compatibility with plasma should be investigated.
  - It can be used to reduce toroidal field ripple.
    --> It will be applied in ITER.

- Advanced studies
  - High normalized beta plasma
  - Investigation of behavior of L-H transition
  - Development of advanced fueling
  - MHD suppression using ECH

The compatibility has been investigated step by step

- Outside VV
  - Vacuum vessel (VV) of JFT-2M
  - Toroidal field coil (TFC)
  - Plasma
  - FP
  - 1999
    - Reduction of ripple loss was well demonstrated

- Partial covering
  - VV
  - TFC
  - FP
  - 2000
    - Compatibility of high performance plasma up to B_T=2.7 was demonstrated

- Full covering
  - VV
  - TFC
  - FP wall
  - 2002
    - Simulation of blanket wall
    - ※ FP: Ferritic Plate (F82H)
Design Concept and Installation of Ferritic Inside Wall (FIW)

- Thickness of FPs
  - For wall stabilization effect
    - FPs should have certain thickness
  - For ripple reduction
    - Thickness profile should be optimized
  - 3 components (10.5 mm, 8 mm, and 6 mm) were employed
  - It is installed along the vacuum vessel not to change the plasma volume
  - Graphite limiter was installed with similar configuration as before for impurity analysis

Precise 3D measurement of the magnetic field was carried out

Vacuum vessel
- B0 probe
- Flux loops

Magnetic field from FIWs
- Ferritic plates (FPs)
- Direction of vertical field

Tokamak discharge was obtained without changing control system
- Calculation of the magnetic field with ferritic steel was performed
  - FPs are saturated, μF ≈ 2-3 (Bt ≈ 1-2 T)
  - Effect on poloidal field
    - Reduction of ~10%
    - It could be cancelled by increasing PF current by ~10%
  - Effect on magnetic sensors
    - Effect is limited in several %
    - Calculation w/o this effect will result in inner shift ~ 1 cm
    - It could be cancelled by changing the setting value of plasma position

Impurity release during the discharge is not large
- Vacuum properties
  - Base pressure was the same as before (6x10^4 Pa) reported in 15th PSI (2002)
  - Impurity release might occur during plasma discharge
    - Due to chemical and physical sputtering
    - Total radiation loss decreased by ~20%
    - Oxygen line intensity became ~ 1/2
    - Meral impurity is less than detection level
    - Reduction of oxygen is probably due to replacement of graphite limiter
Reduction of ripple trapped losses was clearly demonstrated.

Fast ion loss was monitored with IRTV system.

W/O FPs

After FIW

Max. temperature: 75 °C -> >10 °C
Heat flux: 0.26 MW/m² -> >0.01 MW/m²

No adverse effect on plasma has been observed up to now.

- Wide operation region was obtained with FIW
- Collapse region due to tearing mode didn't change
- Threshold power for H mode
  440 kW -> 420 kW
  ~500 kW (scaling)
- H_{θ} factor of ELM/H mode ~1.8
- Growth rate of vertical instability didn't change

Ripple trapped loss increase with thickness of ripple well.

- To investigate the effect of local ripple, outside FP was used.
- OFMC (Orbit Following Monte-Carlo) code was modified, to include full 3D structure of the complex field.

Attractive new "High Recycling Steady" (HRS) operating regime was discovered.

- Steady-state H-mode edge at high density with good confinement
- Complete disappearance of giant ELMs
- Compatibility with ITB
  - K.Kamiya et al., EX/P2-05

Shinohara et al. EXP2-14
CT Injection Experiments on JFT-2M
collaboration with Himeji institute of technology FT/P2-7

- Modification of CT Injector (2001)
  - Drift Tube (to avoid deceleration due to rapid compression)
  - W-Coating of Electrode (to reduce impurity)

- Improvement of Diagnostics
  - Fast Sampling (1MHz) of μ-Wave Interferometer Signal

Before Modification:
All shots disrupted within 5 ms.
After Modification:
About 50% of shots didn't disrupt.

Summary

- Compatibility of low activation ferritic steel with plasma
- Tokamak discharge was obtained without changing control system
- Impurity release is not large
- No adverse effect on plasma stability and confinement
  - Threshold power ~420kW
  - Averaged H factor of ELMy H mode ~1.8
  - Wide operation region (q=2, n_e/n_i)
- Behavior of ripple loss
  - Significant reduction of ripple trapped (RT) loss was demonstrated
- RT loss is roughly proportional to width of ripple well

- Advanced studies for higher performance
  - New operation region called high recycling steady H mode was discovered
  - Compatibility with H mode edge and ITB, \( \beta_n H_{1000} = 6 \)
  - Development of advanced fuelling is in progress
1.3 Achievement of a High Fusion Triple Product and Steady State Sustainment in High $\beta_p$ ELMy H-mode Discharges in JT-60U

A. Isayama$^1$, Y. Kamada$^1$, N. Hayashi$^1$, T. Suzuki$^1$, T. Oikawa$^1$, T. Fujita$^1$, T. Fukuda$^1$, S. Ide$^1$, H. Takenaga$^1$, K. Ushigusa$^1$, T. Ozeki$^1$, Y. Ikeda$^1$, N. Umeda$^1$, H. Yamada$^2$, M. Isobe$^2$, Y. Narushima$^2$, K. Ikeda$^2$, S. Sakakibara$^2$, K. Yamazaki$^2$, K. Nagasaki$^3$ and the JT-60 Team$^1$

1) Japan Atomic Energy Research Institute, Naka, Ibaraki 311-0193, Japan
2) National Institute for Fusion Science, Toki, Gifu 509-5292, Japan
3) Institute of Advanced Energy, Kyoto University, Gokasho, Uji, Kyoto 611-0011, Japan

e-mail contact of main author: isayama@naka.jaeri.go.jp

Abstract. This paper reports results on the progress in steady-state high-$\beta_p$ ELMy H-mode discharges in JT-60U. A fusion triple product $n_{e0}T_{e0}T_\delta(0)$ of $3.1\times10^{20}$ m$^{-3}$ s keV under full non-inductive current drive has been achieved at $I_p=1.8$ MA, which renews the record value of the fusion triple product under full non-inductive current drive by 50%. A high-beta plasma with $\beta_N=2.7$ has been sustained for $7.4\,\text{s} \sim 60\,\tau_D$, where the duration is determined only by the facility limit such as the capability of the poloidal field coils and the upper limit of the injection duration of neutral beams. Destabilization of neoclassical tearing modes (NTMs) has been avoided with good reproducibility by tailoring the current and pressure profiles. On the other hand, the real-time NTM stabilization system has been developed, where the detection of the center of magnetic island and the optimization of injection angle of electron cyclotron (EC) wave are done in real-time. By the application of this system, a 3/2 NTM has been completely stabilized in high beta region ($\beta_N=1.2$, $\beta_C=1.5$), and beta value and confinement enhancement factor have been improved by the stabilization.

1. Introduction

High $\beta_p$ ELMy H-mode plasma is characterized by the weak positive shear with central safety factor above unity, which is compatible with the plasma in the standard operational scenario of the international thermonuclear experimental reactor (ITER). We have optimized the discharge scenario pursuing the steady-state plasmas with high-beta, high-confinement and high non-inductively driven current fraction [1-6]. Since the last IAEA conference, significant improvements have been made in facility: (a) increase in the capacity of poloidal field coils by 20%, (b) increase in beam energy $E_{NNB}$ and injection power $P_{NNB}$ of the negative-ion-based neutral beam (NNB) injection system [7], (c) increase in the number of gyrotrons (4 units in total) and generation power (~1 MW/unit) in the electron cyclotron (EC) wave injection system [8] and (d) routine operation of the pellet injection system. From the operational point of view, one of the key issues for obtaining the high performance plasmas is to suppress neoclassical tearing modes (NTMs). We have adopted two approaches for the NTM suppression: (a) avoidance of destabilization through profile optimization and (b) stabilization by electron cyclotron current drive (ECCD) / electron cyclotron heating (ECH). For the NTM stabilization, real-time NTM stabilization system has been developed, which incorporates with the real-time plasma shape calculation.
2. High Fusion Triple Product under Full Non-inductive Current Drive

We have performed the optimization of high \( \beta_p \) ELMMy H-mode discharges, in which important issues are (a) simultaneous achievement of high-beta and high-confinement with high fraction of non-inductive driven current and (b) steady-state sustainment of the high performance plasma [1-6].

Typical waveforms of a high \( \beta_p \) ELMMy H-mode discharge are shown in Fig. 1, where plasma parameters are as follows: plasma current \( I_p = 1.8 \text{ MA} \), toroidal field \( B_t = 4.1 \text{ T} \), major radius \( R = 3.23 \text{ m} \), minor radius \( a = 0.78 \text{ m} \), safety factor at the 95% flux surface \( q_{95} = 4.1 \) and triangularity at the separatrix \( \delta_b = 0.34 \). In this discharge, NNB with \( E_{\text{NNB}} = 402 \text{ keV} \) and \( P_{\text{NNB}} = 5.7 \text{ MW} \) was injected from 5.7 s. At \( t = 6.5 \text{ s} \), a high-performance plasma with the following parameters was obtained: stored energy \( W_{\text{dia}} = 7.5 \text{ MJ} \), H-factor \( H_{\text{HHF}} = 2.5 \), H-factor \( H_{\text{HJ}} = 1.2 \), poloidal beta \( \beta_p = 1.7 \), normalized beta \( \beta_n = 2.4 \), fusion triple product \( n_D(0)\tau_i T_i(0) = 3.1 \times 10^{20} \text{ m}^{-3} \text{ s} \cdot \text{keV} \) (Here, \( n_D(0) \), \( \tau_i \), \( T_i(0) \) are the central deuterium density, energy confinement time, the central ion temperature, respectively.) and equivalent fusion gain \( Q_{\text{DT}} = 0.185 \). Figures 1(c) and (d) suggest that plasma current is fully non-inductively driven. Note that in evaluating the internal inductance \( \ell_i \), plasma equilibria were reconstructed by using the motional Stark effect (MSE) diagnostic. Profiles of ion temperature \( T_i \), electron temperature \( T_e \), electron density \( n_e \) and safety factor \( q \) at 6.5 s are shown in Fig. 2. Ion temperature was measured with charge exchange recombination spectroscopy, and electron temperature was measured with Thomson scattering. In evaluating the electron density, far-infrared interferometer with tangency radius of \( r = 0.4 \), tangential \( \text{CO}_2 \) laser interferometer and Thomson scattering are used. The central safety factor \( q_0 \) is kept above unity throughout the NB phase to avoid sawtooth oscillations.
In order to validate the full non-inductive current drive, time evolution of the bootstrap current, beam-driven current was simulated using the time-dependent transport code TOPICS. The EC-driven current was evaluated using the Fokker-Planck code combined with the ray-tracing code. As shown in Fig. 3(a), plasma current is fully maintained by the non-inductively driven current. The result of simulation is validated by comparing the stored energy in experiment with simulation: as shown in Fig. 3(b), both agree well each other until the appearance of a mini-collapse at t=6.5 s. Slowing-down of the NNB fast ions was also simulated with the Orbit Following Monte-Carlo (OFMC) code, which indicates that about 95% of energy and parallel momentum are transferred to the plasma in 0.8 s after the injection, suggesting that the plasma is nearly in steady state at t=6.5 s. Fractions of the NB-driven current (f_{NB}) and bootstrap current (f_{BS}) are nearly equal, which is similar to the situation in the steady-state operation in ITER.

In order to visualize the integrated plasma performance, we have used the septangular plot (Fig. 4), which contains (a) normalized beta, (b) HH-factor, (c) line average electron density n_e normalized by the Greenwald density n_G, (d) ratio of radiation power P_{rad} to absorption power P_{abs}, (e) fuel purity defined as the ratio of the number of deuterons to that

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**FIG. 3.** (a) Time evolution of driven current in E39713 simulated with the TOPICS code, (b) Comparison of stored energy between measurement and simulation.

**FIG. 4.** Septangular plot of E39713 data. Values in this figure correspond to those in the steady-state weak positive shear scenario in ITER.

**FIG. 5.** Progress in fusion triple products under full non-inductive current drive.

**FIG. 6.** Operational region plotted on \( \rho_\text{eff} - v_e \) space.

**FIG. 7.** Profiles of \( b(y_{\text{eff}}, l_p) \) near the \( q=2 \) surface. ' + ' symbol denotes the location of the \( q=2 \) surface.
of electrons, (f) fraction of bootstrap current, and (g) fraction of non-inductively driven current \( f_{CD} \). In this plot, each value is normalized by that at ITER [9]. In E39713, \( n_e/n_{cl}=0.42 \), \( P_{rad}/P_{abs}=0.54 \), fuel purity=0.6, \( f_{CD}=1 \) and \( f_{BS}=0.49 \). The values of \( f_{CD} \) and \( f_{BS} \) meet the requirement at ITER, and \( \beta_N \) and \( HH_{1/2} \) are close to the requirement (\( \geq 80\% \)). Longer pulse duration with high density and high radiation fraction is a remaining issue.

Progress of \( nT \) under full non-inductive current drive in high \( \beta_p \) H-mode and reversed shear discharges in JT-60U is shown in Fig. 5. The fusion triple product obtained in E39713 exceeds the previous record (2.0\times10^{20} \text{m}^{-3} \text{s}^{-1} \text{keV} \text{under full non-inductive current drive}) by as much as 50%. In figure 6, ion poloidal Larmor radius \( r_{p,i} \) is plotted against electron collisionality \( v_e* \). In evaluating \( r_{p,i}/v_e \) and \( v_e* \), we used volume average \( n_e \) density weighted volume average \( T_i \) and \( T_e \), poloidal field at the edge, \( q=2 \), effective ion charge \( Z_{eff}=1 \), inverse aspect ratio \( \varepsilon=0.5a/R \). In E39713, \( v_e*=0.017 \) and \( p_{p,i}=0.067 \), and the values of \( v_e*/v_{e,\text{ITER}} \) and \( r_{p,i}/r_{p,i,\text{ITER}} \) are 1.2 and 3.8, respectively (\( v_{e,\text{ITER}} \) and \( r_{p,i,\text{ITER}} \) are \( v_e* \) and \( r_{p,i} \) at ITER, respectively). Accordingly, it indicates that the above result was obtained near the operational region of ITER.

In this series of discharges, NTMs with \( m/n=3/2 \) and 2/1, which appear even at \( q_i>1 \), limit the sustainment of high beta values. Here, \( m \) and \( n \) are poloidal and toroidal mode numbers, respectively. Destabilization of NTMs is avoided by optimizing the current and pressure profiles with good reproducibility so that steep pressure gradient is not located at the mode rational surfaces. As shown in Fig. 2, the \( q=1.5 \) surface is located near the center of the plasma where the pressure gradient is small, which is effective in avoiding the 3/2 mode.

As for the 2/1 mode, we have attempted to decrease the value of \( \beta_p(L_q/L_p) \), which is one of the measures for the onset of neoclassical tearing modes (\( L_q=q'(dq/dr) \), \( L_p=p'(dp/dr) \); \( \beta_p \) is evaluated using local parameters). In Fig. 7, profiles of \( \beta_p(L_q/L_p) \) in several discharges are displayed (\( q_{95}=4.1-4.75 \)). Note that high \( \beta_p \) (\( \sim 2.5 \)) and high \( nT \) were sustained without NTM in shot E36715, and that the 2/1 mode was destabilized in shot E36706. In shot E39713, the value of \( \beta_p(L_q/L_p) \) at the \( q=2 \) surface ('+' symbol in Fig. 7) is kept as low as that in E36715 even at higher stored energy by making the q-profile broader. Accordingly, the profile optimization is thought to be effective in avoiding the destabilization of NTMs.

3. Steady-state Sustainment of High-beta Plasmas

Demonstration of the steady-state sustainment of high-beta plasmas is important to investigate the phenomena which might manifest corresponding to the time scale of current diffusion. The current diffusion time in a high \( \beta_p \) H-mode discharge is an order of 10 s in JT-60U due to its large size and high temperature. Such a long-pulse and high-\( \beta \) discharge has been realized by the improvement in the poloidal coil power supply: a high triangularity plasma with \( \delta_e=0.45 \) can be sustained for 10 s, which duration also corresponds to the maximum pulse width of the neutral beams in JT-60U.

Typical waveforms of a long-pulse high-\( \beta_p \) ELMy H-mode discharge are shown in Fig. 8. Plasma parameters are as follows: \( I_p=1.0 \text{ MA} \), \( B_t=1.8 \text{ T} \), \( q_{95}=3.3 \), \( \delta_e=0.45 \). NB power was gradually increased by the stored energy feedback in order to avoid the 2/1 mode and
large-amplitude 3/2 mode, which cause serious confinement degradation. Deuterium pellets were injected from the high-field side at 120 m/s, 10 Hz from t=3 s, which contribute to obtaining a high density plasma without significant confinement degradation. The pellet injection also contributes to keeping core density for reducing the shine-through loss power, which is small (≤ 10%) but becomes important in the long-pulse operation. Although a 3/2 mode started to grow at t=4.3 s (βn~2.9), normalized beta was kept almost stationary and no continuous degradation was observed, and an ELMy H-mode plasma with βN~2.7 and βp~1.5 was sustained for 7.4 s, which corresponds to ~60τE. Here, the duration is determined by the facility constraint. In E39706, total NB injection power reaches 180 MJ, but no significant increase in the Dα signal and the impurity content was observed. As shown in Fig. 9, safety factor profile is almost identical, and the central safety factor is kept above unity during the NB phase. This fact was also confirmed by electron temperature perturbations measured with electron cyclotron emission (ECE) diagnostics: no sawtooth is observed, and the location of the magnetic island corresponding to the 3/2 mode is almost unchanged.

Figure 10 shows the value of βN against duration τduration normalized by τE in typical high βp ELMy H-mode discharges in JT-60U. In shot E39511 (Ip=1 MA, B=2 T, q95=3.6), a high-performance plasma with βN=2.7, βp=1.6, H95%=1.8, HHy=0.89, te/e=0.67 was sustained for 6.5 s. It is obvious that the operational region has been significantly extended by these discharges: High value of βN, which is comparable to that in ITER, was obtained in larger τduration/τE region.
4. Real-time NTM Stabilization

Stabilization of NTMs by EC is important to sustain high beta plasmas. In JT-60U, it was demonstrated that the 3/2 NTM can be completely stabilized by injecting the fundamental O-mode EC wave to the center of the magnetic island [10]. At the same time, it has been found that the precise adjustment of EC injection angle is indispensable in order to achieve the complete stabilization. In case the deposition location is misaligned by only about half of the island width (~several centimeters), the stabilization effect by EC was significantly decreased.

When EC is used as a tool for the NTM stabilization in a future device such as ITER, NTMs are needed to be detected and stabilized in real-time, since the optimum injection angle changes in time. In JT-60U, we developed a real-time NTM stabilization system and demonstrated that the center of the magnetic island can be detected from the electron temperature perturbation profile. In 2002, extensive development has been made in the real-time control system, where the plasma shape is first reconstructed in real-time (every 10 ms) with the Cauchy condition surface (CCS) method [11], and the mode location is coarsely estimated. Subsequently, fine tuning is performed by evaluating the electron temperature perturbation profile, utilizing the fact that an M-shaped perturbation profile (see Fig. 13) is obtained at the magnetic island and that the center of the island corresponds to the local minimum point of the profile. In obtaining the perturbation profile, standard deviation of the ECE heterodyne radiometer signal is evaluated. This evaluation method has the advantage that the mode amplitude can be obtained without evaluating the mode frequency.

Typical waveforms of the NTM stabilization experiment are shown in Fig. 11. Plasma parameters of this discharge are as follows: \( I_p = 1.5 \text{ MA} \), \( B_t = 3.7 \text{ T} \), \( R = 3.3 \text{ m} \), \( a = 0.78 \text{ m} \), \( q_{95} = 3.8 \). A 3/2 NTM was destabilized at \( \beta_N \sim 1.5 \) by the NB injection power of \( \sim 20 \text{ MW} \). The mode amplitude gradually decreased after 3 MW EC injection at \( t = 7.56 \text{ s} \) (\( H_{\text{95FL}} = 1.8 \), \( HH_{2} = 1.0 \)), and the 3/2 mode was completely stabilized at 8.8 s. Even after the turn-off of the EC injection at 9.5 s, the 3/2 mode did not appear and \( \beta_N \) continued to increase to 1.67. Since NB injection power was fixed, this shows confinement improvement. In fact, \( H_{95FL} \) and \( HH_{2} \) increase to 1.9 and 1.1, respectively. At \( t = 10.8 \text{ s} \), the 3/2 mode reappeared, and \( \beta_N \) and \( \bar{n}_e \) decreased. Profiles of electron temperature at \( t = 7.5 \text{ s} \) (just before EC), 9.4 s (after the stabilization), 10.7 s (just before the mode reappearance) and 11.5 s are shown in Fig. 12. It is shown that a flat region is formed at \( R = 3.65-3.7 \text{ m} \) suggesting the formation of magnetic island, and \( T_e \) inside the mode rational surface is affected by the NTM. An increase of \( \beta_N \) by the stabilization is 11% and degradation due to the NTM reappearance is 13%. These values are comparable to the prediction of the island model (~13%) by Chang et al.[12].

Figure 11(e) shows time traces of channel number of the heterodyne radiometer. The value for \( X_{\text{CCS}} \) corresponds to the channel number at the mode rational surface evaluated with the CCS method, and \( X_{\text{min}} \) corresponds to the location at which the perturbation level reaches the local minimum. The value of \( X_{\text{CCS}} \) stays at channel 8 since the plasma position is fixed. The real-time system functions in a way that EC angle is changed and EC is injected at
FIG. 11. Typical waveforms of a real-time NTM stabilization experiment: (a) injection power of NBs and EC, (b) amplitude of magnetic perturbations with n=2, (c) normalized beta, (d) line average electron density, (e) channel number of the heterodyne radiometer.

the flux surface corresponding to $X_{\text{min}}$. The validity of this function is confirmed by the amplitude profile depicted in Fig. 13: the local minimum is actually located at channel 6 or 7 at $t=7.5\,\text{s}$. The island center is located at $R=3.68\,\text{m}$ at $t=11\,\text{s}$, which indicates that the mode rational surface has changed its location. The real-time system recognizes the change and indicates that the island center is located at channel 8.

The ratio of the EC-driven current to the bootstrap current at the mode rational surface is an important parameter as a measure of the efficiency of the stabilization. According to the Fokker-Planck code and the ACCOME code, the maximum EC-driven current density at the mode rational surface is comparable to the bootstrap current density ($\sim0.25\,\text{MA/m}^2$), which shows that the stabilization was effectively accomplished.

Time evolution of the amplitude of electron temperature perturbations $\tilde{T}_e$ is shown in Fig. 14, where $\tilde{T}_e$ is measured at 2 cm intervals. The brighter region corresponds to the region with large mode amplitude. The 'valley' of the contour plot corresponds to the center of the island. One can see that the center of the island is slightly meandering in time. After the EC injection, the mode amplitude at the high field side rapidly decreases, while it is not the case at the low field side. This shows asymmetry in the perturbation profile. It is also noteworthy that the amplitude at the low field side increases after the EC injection for $\sim100\,\text{ms}$. As time goes on, the mode amplitude at the low-field side decreases and at the same time island width ($\sim$distance between the two peaks) also decreases. At $t=8.4\,\text{s}$, the island width rapidly decreases. This behavior is consistent with the description of the modified Rutherford equation, as shown in Fig. 15: $dw/dt<0$ for $w<0.05$ (w is island width.). It should be also noted that the increase in beta and density starts at 8.3–8.5 s (Fig. 12), which is close to the time of the quick shrink.
5. Conclusions

In the high $\beta_p$ ELMy H-mode discharges, we have obtained the following results:

- Highest fusion triple product of $3.1 \times 10^{20}$ m$^{-3}$ s$^{-1}$ keV under full non-inductive current drive has been obtained at $I_p=1.8$ MA, $B_t=4.1$ T, $q_{95}=4.1$. Values of collisionality and poloidal Larmor radius are close to those at ITER ($v_e/v_e^{\text{ITER}}=1$, $\rho_{p\parallel}/\rho_{p\parallel}^{\text{ITER}}=4$). In the same series of discharges, destabilization of NTMs has been avoided with good reproducibility through the optimization of current and pressure profiles.

- A high beta plasma with $\beta_N=2.7$, $\beta_p=1.5$ has been sustained for 7.4 s at $q_{95}=3.3$. The duration time of high beta extends to $\sim 60\tau_E$, which is limited by the facility capability.

- Real-time NTM stabilization system, where the identification of the mode location and the feedback control of EC injection angle are performed in real-time, has been developed. By using this system, a 3/2 NTM in high beta region ($\beta_N=1.5$, $\beta_p=1.1$; $B_t=3.7$ T) has been completely stabilized, and beta value and H-factor have increased spontaneously after the stabilization.

References

Achievement of a High Fusion Triple Product and Steady State Sustainment in High $\beta_p$ ELM My H-mode Discharges in JT-60U

A. Isayama$^1$, Y. Kamada$^1$, N. Hayashi$^1$, T. Suzuki$^1$, T. Okawa$^1$, T. Fujita$^1$, T. Fukuda$^1$, S. Ide$^1$, H. Takenaga$^1$, K. Ushigusa$^1$, T. Ozeki$^1$, Y. Ikeda$^1$, N. Umeda$^1$, H. Yamada$^2$, M. Isobe$^2$, Y. Narushima$^2$, K. Ikeda$^2$, S. Sakakibara$^2$, K. Yamazaki$^2$, K. Nagasaka$^3$ and the JT-60 Team$^1$

1) Japan Atomic Energy Research Institute
2) National Institute for Fusion Science
3) Institute of Advanced Energy, Kyoto University

The 19th IAEA Fusion Energy Conference, October 14-19, 2002, Lyon, France

Introduction

- ELM My H-mode: Standard operation scenario in ITER
- High $\beta_p$ ELM My H-mode in JT-60U
  - Compatible with the ITER scenario
  - Weak positive shear with $q_0>1$
- One of the key issues:
  Neoclassical tearing modes (NTMs)
  - Two approaches for NTM suppression in low $v$ & $\rho$ region
    - Avoidance of destabilization by $p(r)$ & $q(r)$ control
    - Stabilization by ECCD/ECH

Topics in this talk

1. High nT under full non-inductive current drive $p$ & $q$ control
2. Long-pulse high-$\beta$ discharge
3. Real-time NTM stabilization

High in JT-60U

Destabilization of NTMs has been avoided through profile optimization.

- NTMs appear at $p=0.3-0.7$: large $V_p$
- Avoidance of destabilization
  - 3/2 mode: $q=1.5$ near the center
  - 2/1 mode: decrease $\beta_p/Lp$ $\rightarrow$ NTM suppression with good reproducibility

High Fusion Triple Product of $3.1\times10^{20}\text{m}^{-3}\text{keV}^2\text{s}$ has been obtained under full non-inductive current drive.

- No NTM
- NNB: 402keV, 5.7MW
- At $t=6.5s$:
  - $W_{\text{pol}}=7.5MJ$, $\beta_p=2.4$, $v_p=1.7$, $H_{\text{H}}=1.2$
  - $H_{\text{pol}}=2.5$, $t_{\text{pol}}=0.34s$
  - $n_0(0)=4.2\times10^{19}\text{m}^{-3}$
  - $Z_{\text{eff}}=3.0$, $T_e(0)=21.5\text{keV}$
  - $n_T=3.1\times10^{20}\text{m}^{-3}\text{keV}^2\text{s}$
  - $V_{\text{loop}}\approx0$, $dE/dt\approx0$
  - Full-CD
High nT under Full Non-Inductive Current Drive

**Time-dependent transport code supports the experimental result.**

- Non-inductively driven current exceeds 1.8MA at ~6.4s → full-CD
- $f_{\text{w-nocd}}$ similar in ITER
- Simulation result agrees well with experimental one until 6.5s (mini-collapse)

**Significant progress in nT has been made in low $\nu$ and $\rho$ region with NTM suppression.**

- Increase by 50% $\rho$ and $\nu$: closer to ITER
- Remaining issue: longer pulse width at high density

High beta plasma with $\beta_N=2.7$, $\beta_p=1.5$ has been sustained for 7.4s ($\sim 60\tau_e$) in low-q ($q_{95}=3.3$) region.

**Long-Pulse High $\beta$ Discharge**

**Real-Time NTM Stabilization**

Real-time NTM stabilization system has been developed.

1. Coarse estimation of mode location
2. Fine tuning using $T_e$ profile
3. EC mirror mirror steering

- Calculation: 10ms
- Mirror scan: $\Delta R_{\text{dep}}/\Delta t \sim 10 \text{cm/s}$
**Real-time NTM stabilization**

A 3/2 NTM at high beta ($\beta_n=1.5$, $\beta_p=1.1$) has been completely stabilized with the real-time system.

- $\beta_n$ increased by the stabilization, and even after the EC turn-off
- Confinement improvement
  (H$\alpha$: 1.8 $\rightarrow$ 1.9; H$\beta_p$: 1.0 $\rightarrow$ 1.1)

**Summary**

**High $\alpha T$ under full non-inductive current drive**
- $\alpha T=3.1x10^{10}$m$^{-3}$s·keV (increase by 50%)
- Achievement at low $v$ & $p$ ($v_{i*}$/$v_{i*}$-ITER=1, $p_i/p_{i*}$-ITER=4)
- NTM suppression by profile optimization

**Long-pulse high-$\beta$ discharge**
- $\beta_n$=2.7, $\beta_p$=1.5 for $\sim$60s at $q_{95}$=3.3
- $\beta_n$ comparable to that in ITER
- Duration limited by facility capability (NB, power supply)

**Real-time NTM stabilization**
- Demonstration of real-time NTM detection & mirror steering
- Complete stabilization of NTM at $\beta_n=1.5$, $\beta_p=1.1$
- Increase in $\beta_n$ and H-factor

**Perturbation decreases asymmetrically during the stabilization.**

- Just after the EC injection,
  - $T_e$ at HFS: decrease
  - $T_e$ at LFS: increase
- Asymmetry ... future work
- Rapid decrease in island width at 8.3s:
  - Consistent with the modified Rutherford eq.
1.4 Stable Existence of Central Current Hole in the JT-60U Tokamak

Y. Miura 1), T. Fujita 1), T. Oikawa 1), T. Suzuki 1), S. Ide 1), Y. Sakamoto 1), Y. Koide 1), T. Hatae 1), O. Naito 1), A. Isayama 1), H. Shirai 1), R. Nazikian 2), K. Shinozaka 1), A.V.Chankin 1) and the JT-60 Team

1) Japan Atomic Energy Research Institute, Ibaraki, 311-0193, Japan
2) Princeton Plasma Physics Laboratory, Princeton New Jersey, 08543-0451, U.S.A.

E-mail: miura@naka.jaeri.go.jp

Abstract. In an extreme state of a reversed magnetic shear configuration, it was found in JT-60U that there is almost no plasma current in the central region (called Current Hole). The Current Hole region extends to 40% of the plasma minor radius and it exists stably for several seconds. The Current Hole is formed by the growth of the bootstrap current and it is impossible to drive current in either positive or negative direction by ECH or N-NB inside the Current Hole. In that region, there is almost no gradient of density, temperature and toroidal rotation velocity. It means that there is almost no confinement in the Current Hole and the large energy in that region is sustained only by an internal transport barrier (ITB). The effects of the Current Hole on particle orbits and the effects on an error field on the Current Hole are also discussed.

1. Introduction

An advanced tokamak scenario by a reversed magnetic shear configuration has a hollow current profile and shows very high confinement with an internal transport barrier (ITB) inside the position of the minimum safety factor, q_{min}. In an extreme situation of the hollow current profile, it was found for the first time in JT-60U [1] that there is almost no plasma current in the central region. Though the current was believed to be necessary for tokamak plasma equilibrium. Moreover, it was observed for the first time that the Current Hole region extends to 40% of the plasma minor radius and it exists stably for several seconds [2]. The Current Hole is formed by the growth of the bootstrap current and it is impossible to drive current in either positive or negative direction by ECH or N-NB inside the Current Hole. In that region, there is almost no gradient of density, temperature and toroidal rotation velocity, and ECH heat wave travels very rapidly. It means that there is almost no confinement in the Current Hole and the large energy in that region is sustained only by an ITB. The property of the Current Hole plasma is studied.

2. Formation of the Current Hole and Stable Existence of the Current Hole

Figure 1 shows an example of the formation of the Current Hole. High power neutral beams are injected during plasma current ramp-up as shown in Fig.1 (a). The q-profiles and current density profiles are shown in Fig.1 (b) and (c), respectively. The profiles of loop voltage are shown in Fig.1 (d). After increasing the neutral beam power at 5.0 sec, off-axis current increases. Since no counter current expected due to balanced NB injection in this case, the off-axis current comes mainly from the bootstrap current. The growth of the bootstrap current makes the central loop voltage decrease, then the central current density decreases and finally goes to nearly zero (q (0) goes up very high value). After 5.4 sec, even the central loop voltage goes below negative, a negative central current is not observed. The central current is likely to be clamped zero. This special feature will be discussed in section 4. The Current Hole is also observed in JET with off-axis lower hybrid current drive [3]. The big difference in the observation between JET and JT-60U is the existence of MHD modes. The sawtooth like MHD modes present in the JET observation, but not present in JT-60U. Figure
2 shows the sustainment of the Current Hole. In this discharge, ECH of 0.75 MW was injected into a central region during the current ramp. Though the Current Hole was also observed in other discharges without ECH, the high central electron temperature by ECH may be related to the formation of Current Hole with a large radius. In Fig. 2 (c), $B_\theta/B_\parallel$ at five Motional Stark Effect polarimetry (MSE) points near the axis are shown. The values of inner three points, $\rho < 0.27$, stay nearly zero from $t = 4$ sec to 8 sec, which indicates that no substantial current exists within $\rho = 0.27$ during this period. In Fig. 2 (d) the contour plot of current density is shown. The hatched region indicating the Current Hole remains for 5 sec though its radius shrinks slowly according to the shrinkage of the peak in current density and of ITB radius. Since the power is reduced to about 8 MW at 6.8 sec to avoid a collapse and when the heating power dropped to 1 MW leaving only one NB unit for MSE diagnostics at 9.2 sec, the ITB disappeared and the Current Hole disappeared. In this discharge, the
Current Hole is sustained for 4 to 5 seconds without any global MHD instabilities. This implies that the equilibrium with the Current Hole has a good stability.

3. Confinement and Particle orbit in the Current Hole and Effect of Error Field

Figure 3 shows the temperature profiles at 5.4 sec of the discharge shown in Fig.2. These profiles inside the Current Hole show nearly flat, but steep gradients are formed outside the Current Hole where \( j(r) \) is peaked. In the case of injecting ECH, the heat wave travels very rapidly inside the Current Hole. These results mean that there is almost no confinement in the Current Hole and the high temperature plasma can be sustained only by the ITB. In this situation, it is very important to know how large the radial excursion of particle orbit is. The orbits of thermal ions with 8 keV, which is equal to the central ion temperature shown in Fig. 3 are calculated and shown in Fig. 4 (a). Here, the ion orbits are traced from the magnetic axis varying the pitch angle of the velocity. In the Current Hole region, ions move almost vertically due to grad-B and curvature drifts and start to move in the poloidal direction when they go out of the Current Hole and enter the region with a significant poloidal field. The largest radius reached by each orbit is plotted as a function of pitch angle at the axis in Fig. 4 (b). The largest banana width is as large as 65 % of the plasma minor radius and any particle starting from the axis reaches \( \rho = 0.47 \). This implies that the poor ion confinement is expected in the region inside \( \rho = 0.47 \). This position is almost equal to the ITB shoulder in this plasma as shown in Fig. 3. On the other hand, the radial drift of orbit of thermal electrons is much smaller than that of thermal ions or a few cm for the case with \( q(0) = 100 \). Hence the flat portion in \( T_e \) profiles cannot be understood by the orbit size if \( q(0) = 100 \). These imply that large anomalous transport exists or \( q(0) \) is much larger than 100.

Fig 4. (a) Deuterium ion orbits with temperature 8keV and different pitch angle at the magnetic axis. (b) Maximum radial excursion of ions with different pitch angle started from the magnetic axis.
We also have the case that the shoulder in $T_i$ is located at $\rho \sim 0.45$ while the width of largest banana of thermal ions is $\rho \sim 0.25$. This suggests that the location of shoulder is not directly related to the weak poloidal field or existence of the Current Hole.

To investigate the relation between weakness of poloidal field and radial transport, a non-axisymmetric error field was applied to plasma with the Current Hole. The radial component of the error field had a maximum value of $\sim 40$ Gauss in the plasma region, which is comparable to the poloidal field just outside the Current Hole. We expect that the radial transport, if it is determined by the weak poloidal field, is affected by the applied error field. Figure 5 shows the time evolution of ion temperatures at several locations. The error field was applied during quasi-steady phase with constant heating power. We find that the ion temperatures were hardly affected by the error field. No large responses were either observed in $T_e, n_e, V_A$ (toroidal rotation of carbon ions) and MSE polarization angle (poloidal field). It shows that the radius of the Current Hole is not determined by the error field.

The density fluctuation in the flat density region (different shot), which is evaluated by the reflectometry measurement, is about one order of magnitude smaller than that at the ITB region [4]. Small density fluctuation in the flat density region suggests that nearly flat electron temperature profile may not be the result of anomalous transport, but may be the result of larger $q'(0)$ value ($q(0)>100$). Since it is not found a clear effect of the error field on confinement, the confinement region with small $B_P$ (outside of the Current Hole) may be determined by the anomalous transport. Further studies are necessary to conclude these explanations.

Fig.5. Application of an error field (DCW) to the Current Hole $I_p = 1.4$ MA, $B_0 = 3.7$T $q_{95} = 4.4$ $\beta_p = 0.8$. (a) Ion temperature and electron density profile. (b) Safety factor profile. (c) Time evolution of NB power. (d) Time evolution of ion temperature at several points and DCW coil current.
4. Response of the Current Drive inside the Current Hole

One of the interesting points is that no substantial negative current in the central region was observed so far. This suggests the decrease of central current caused by the growth of off-axis bootstrap current stops and does not go below negative when the central current becomes zero. On the other hand, in the discharge shown in Fig. 2, ECH was injected tangentially and a non-inductive current by ECH was expected. The driven current by the injection of 0.75 MW ECH is estimated 44 kA and localized inside $\rho = 0.2$ according to the analysis using a ray-tracing Fokker-Plank code. This current would have generated poloidal fields of $B_\theta/B_r \sim 0.010-0.014$ at three MSE channels at $\rho = 0.17, 0.27$ and 0.37. However, as shown in Fig. 2 (c), these poloidal fields are not detected during ECH. In Fig. 6, compared with the change of MSE measured pitch angle for the case with finite current, no change of the pitch angle by ECH is observed in the Current Hole. Even the central electron temperatures are high in both cases and around 6 keV, the constant pitch angle in the Current Hole is not explained by the delay of current penetration. These observations show the existence of the mechanism to keep nearly zero current in the central region. Recently, Huysmans et al. proposed the influence of a resistive kink MHD instability as the mechanism of the zero central current density and the absence of negative central current [5]. However, since no global MHD activity is observed in JT-60U, it may not explain the observation in JT-60U result, or the expected MHD instability is too small to detect.

5. Summary

In JT-60U experiment, the Current Hole state, which is characterized with nearly zero central toroidal current density, is found. The Current Hole region extends to 40% of the plasma minor radius and it exists stably for several seconds. The Current Hole is formed by the growth of the bootstrap current. In that region, there is almost no gradient of density, temperature and toroidal rotation velocity. The radial excursion of ion orbit is very large and the ion orbit is not kept inside the Current Hole. It is different from electron orbit with $q(0)=100$ which is well confined within the Current Hole. Those results suggest that large anomalous transport exists or $q(0)$ is much larger than 100. The EC current drive experiment shows that it is impossible to drive current in either positive or negative direction inside the Current Hole. And it is not found the clear effect of the error field ($Br=40G$). It means that there is mechanisms to clamp the central current density nearly zero and the radius of the Current Hole is not determined by the error field.

References

Radial excursion of ion and electron orbit

- Orbit is calculated with $q(0)=100$ for ions and electrons.
- Ion banana orbits are wide and can reach $\rho = 0.65$.

- Electron orbits are maintained even in the current hole if $q(0)=100$.
- $q(0) >> 100$ (?)
- Anomalous transport (?)

Shoulder radius can be larger than banana width

- No current hole or small current hole with $\rho < 0.1$ if any.
- The width of largest banana of thermal ions is $\rho = 0.25$ while $\rho_{\text{shoulder}} = 0.45$.
- The radius of flat region of $V_t$ is $\rho = 0.25$ and equal to the banana width.
Central Current is clamped to Zero in JT-60U

- The peaked $j_{EC}$ is not generated in a current hole. As suggested by a flat Te profile, it implies low radial confinement of electron momentum.

- Even if a uniform $j_{EC}$ is generated in the current hole, it should be detected by MSE (green curve).

- Absence of this current suggests that $j$ is clamped to zero in the current hole.

- Absence of the ECCD current in the current hole is also true for the case of counter ECCD.

Correlation Reflectometer & Full-wave Simulation Reveals Strong Radial Variation of Fluctuation Level

Very low fluctuation levels were observed in flat pressure region in the core of the ITB plasma. What is the role of fluctuations in the transport inside the flat region?
Effect of error field in a plasma with a current hole

- Non-axisymmetric error field ($B_r$ max = 40 gauss) was applied to a plasma with a current hole.

- No clear effects were observed in $T_I$, $T_e$, $n_e$, toroidal rotation and MSE angle.

- This supports that the position of ITB shoulder is not determined by large orbits of ions or electrons in a weak poloidal field but by other mechanisms (turbulence?).

- The radius of current hole is not either determined by error fields.

$I_p = 1.4$ MA, $B_t = 3.7T$, $q_{95} = 4.4$, $\beta_N = 0.8$. 
Summary

- In extremely reversed shear plasmas on JT-60U, $j(0)$ is very small or zero (Current Hole). The current hole was sustained stably for several seconds. The radius of current hole extended up to 40% of plasma minor radius.
- The current hole was generated by a negatively induced toroidal electric field through the increase of off-axis bootstrap current.
- The observed $T(r)$ and $n(r)$ are nearly flat inside the Current Hole. All ion orbits show that ions travel from inside to outside of the current hole.
- The radius of current hole is not either determined by error fields.
- Clear negative or positive driven current is not observed inside the current hole.
1.5 Studies of Current Profile Optimization and Influence of Electron Heating towards Advanced Tokamak Operation on JT-60U


Naka Fusion Research Establishment Japan Atomic Energy Research Institute
Naka-machi, Naka-gun, Ibaraki, 311-0193 Japan

e-mail contact of main author: ide@naka.jaeri.go.jp

Abstract. Experimental results of studies towards steady state operation of an advanced tokamak on JT-60U are presented. Especially, issues that are related to a current profile and internal transport barriers (ITBs) with emphasis on fusion relevant conditions are discussed. The major results are; 1) High confinement improvement at high normalized density regime which is expected in the ITER steady state operational scenario was obtained in a reversed magnetic shear (RS) plasma under full non inductive current drive (CD). In the discharge, capability of active current profile control was demonstrated. 2) Influence of dominant electron heating on the ITBs was investigated, and it was found that in RS plasmas the confinement improvement by the ITBs could be maintained in the dominant electron heating regime. On the other hand, it was found in positive shear plasmas that the T1 ITB could be degraded by dominant electron heating.

1. Introduction

Toward realization of a steady state (SS) tokamak fusion reactor, it is essential to increase the fraction \( f_{|B|} \) of the bootstrap (BS) current relative to the total plasma current \( (I_p) \); a value \( f_{|B|} \sim 70\% \) is typically required [1]. Formation of an internal transport barrier (ITB) is one effective method of achieving a high \( f_{|B|} \). In such a plasma, the current density profile becomes hollow due to off-axis BS current, and the magnetic shear becomes negative in the plasma core region. The JT-60U experiments on reversed (negative) magnetic shear (RS) plasmas have revealed that ITBs can reduce anomalous heat transport dramatically to enable excellent confinement and \( f_{|B|} \) values, comparable to, or even better than, those expected in future machines [2,3]. On the other hand, too strong ITB may result in a very hollow current profile, or a deeply reversed safety factor \( (q) \) profile. Recently even a current hole has been found in an RS plasma [4,5] Such a hollow current profile would not be desirable in a reactor, since the orbits of alpha particles might become too large to allow confinement of them in the core region. Moreover collective modes might be destabilized and degrade the alpha particle confinement. Therefore lowering central \( q \) with such a strongly localized BS current is one of the critical issues.

On the other hand, study of characteristics of ITBs under fusion plasma relevant conditions, especially on heating method, is also important. In a fusion plasma, main heating source is \( \alpha \) particles therefore the heating power is firstly fed to electrons predominantly. However, in many experiments high performance plasmas are obtained by using positive-ion based neutral beam (P-NB) heating. Since the beam acceleration energy \( (E_B) \) of a P-NB is usually ranging from several tens of keV to about a hundred keV and the electron temperature \( (T_e) \) in a target plasma is in excess of a few keV, heating power is initially absorbed by ions predominantly. Thus the ion temperature \( (T_i) \) tends to be higher than
FIG. 1: Waveforms, (a) – (d), and profiles, (e) – (h) of typical parameters in an RS discharge LHCD and N-NBHD are combined to achieve full-CD with good performance (E37964).

$T_e$ in these experiments. It is theoretically expected that $T_e/T_i > 1$ is a destabilizing factor to the ion temperature gradient (ITG) micro-instabilities, which is expected to be responsible to the confinement degradation. If the mode is dominant, it would affect formation and sustainment of ITBs under dominant electron heating. Therefore, it is important to investigate confinement improvement in such a condition.

On the JT-60U tokamak, various kinds of heating (H) and current drive (CD) systems are installed and therefore is potentially suitable to investigate improved confinement in electron heating dominant regime; the lower hybrid range of frequencies (LHRF) system is for CD (LHCD) and heating, the electron cyclotron range of frequency (ECRF) system is for CD and direct electron heating without particle fueling and the negative-ion based NB (N-NB) system is for CD and mostly electron heating because of higher $E_B$ (<500 keV) with low fueling. Utilizing these various H/CD systems, issues discussed above have been investigated. In this paper, results of current profile modification and dominant electron heating issues on ITB plasmas are shown.

2. Current Drive and Profile Control in a High Performance RS Plasma

Since the current profile is very important in an RS plasma, control and sustainment of a wide but not deeply hollow current profile with good alignment to the pressure profile are key issues. There would be several aspect in view of current profile optimization in an RS plasma; 1) confinement improvement, 2) reduction of central $q$, 3) improvement of MHD limits and so on. Here the experimental results on the first and the second issues are discussed. Concerning the first issue, expansion of location ($\rho_{q_{min}}$) of the minimum in the $q$ profile ($q_{min}$) by means of off-axis CD would be effective. The location of the ITB foot location ($\rho_{foot}$) has been reported to be strongly related to the $\rho_{q_{min}}$ location, and the confinement of an RS plasma can be scaled with $\rho$ [2]. Considering excellent achievement in many tokamaks [7–10], LHCD should be the most appropriate to apply on this issue. Toward the second issue, approach would be straightforward, that is utilization of central CD. To this purpose, N-NB is a good choice. Although, CD capability in a standard plasma has been proved [11], it should be investigate in such a RS plasma, since higher $q$ in the central region may prevent full capability.

The plasma was operated with $I_p = 0.9$ MA, $B_{tor} = 2.5$ T, $q_{95} = 6.9$, and the working gas was deuterium [12]. Waveforms of typical parameters are plotted in Fig. 1.
The plasma was initiated at 3 s with $I_p \sim 0.4$ MA. Then $I_p$ was ramped up to 0.9 MA in 1 s with P-NB heating to form an RS configuration and ITBs as per usual JT-60U RS plasma operation [2]. Up to 6.3 s the stored energy evaluated from the diamagnetics measurement ($W_{diss}$) was feedback controlled by the P-NB power ($P_{P-NB}$) so as to obtain and keep a prescribed value of $\beta_N$. It should be noted that $E_N$ was $\sim 370$ keV for the N-NB. After 6.3 s, only pre-programmed P-NB units were used. Injection of N-NB and LH-RF started at 6.1 and 6.3 s as shown in Fig.s 1 (a) and (b). The frequency of LH-RF was 2 GHz, and two multi-junction launchers were used for the LH-RF injection and the spectra were chosen to enhance off-axis LHCD [7,8,13,14]. The surface loop voltage ($V_{ls}$) keeps decreasing and reaches $\sim 0.2$ V as shown in Fig. 1 (b). Temporal evolutions of $\rho_{foot}$ and $\rho_{q_{min}}$ are plotted in Fig. 1 (c). As shown in the figure, $\rho_{q_{min}}$ starts increasing after the LH-RF injection and $\rho_{foot}$ expands following the movement of $\rho_{foot}$. It should be noted that in this discharge, the ITB foot rather locates outside the q minimum. In order to assess non-inductive current drive, the loop voltage ($V_{l}$) profile inside the plasma was evaluated at around 7 s from the Motional Stark Effect (MSE) measurement [15,16], and is plotted in Fig. 1 (e). The profile is negative. This indicates that the plasma current is fully or even over drive by non-inductive currents. The bootstrap current fraction is evaluated as 62% by the ACCOME code [17]. The profiles of $T_e$, $T_i$, $n_e$ and $q$ are shown in Fig. 1 (f) - (h). Owing to the expansion of $\rho_{foot}$, the temperature and density ITBs build up at very large position.

In Fig. 1 (d) plotted is $q$ at $\rho = 0.4$ and 0.6. The safety factor keeps decreasing at $\rho = 0.4$ and the decrease should be attributed to the central N-NBCD. On the other hand, increase in $q$ at $\rho = 0.6$, near the $q_{min}$ location before external current drive, should be attributed to expansion of $q_{min}$. As the result, the $q$ profile (Fig. 1 (h)) at 7.24 s looks quite differently from that observed in usual P-NB heated RS plasmas. Actually the $q$ profile at 6.15 s is one of quite a common ones. In the $q$ profile at 7.24 s, lowering $q$ by N-NB-CD is quite successful at $\rho > 0.3$. Although the ACCOME code predicts more centrally peaked driven current profile, $q$ near the center looks to stay high. The reason is not clear yet. It might be attributed to a current hole [4]. It remains to be an important issue to investigate intensively. However, lowering $q$ even outside $\rho > 0.3$ would help together with the expansion of the ITB. Even if the high energy ion orbit expands due to high $q$ they might stay well inside very flat profile inside the ITB. It also should be noted that decrease in $q$ or rather flattening of the $q$ profile is clearly highlighted because of off-axis CD by LH-RF. Central intensive CD would simply make $\rho_{q_{min}}$ shrunking to lose RS area.

It should be noted that this plasma is very interesting from a view point of performance. The confinement improvement factors relative to ELMy H-mode scaling ($HH_{ELM}$) and to L-mode scaling ($H_{LPP}$) [18] were $\sim 2.2$ and $\sim 3.5$. This good confinement was sustained for 2.7 s (six times the energy confinement time ($\tau_E$)). Furthermore, the line averaged
density normalized to the Greenwald density [19] reached 0.8 in this experiment. As shown in Fig. 2 (a), this is a good advance from the past JT-60U steady state experiments. The typical parameters achieved in the plasma is plotted against those required in the ITER steady state operational scenario. As is shown in the figure, the achievement is quite successful in core confinement and CD, while more efforts should be required for particle and heat handling.

3. ITB under Dominant Electron Heating

Characteristics of ITBs under dominant electron heating have been investigated by changing a ratio of fed power to electrons (P_e) to that to ions (P_i), or T_e/T_i, utilizing the LHRF, the ECRF and both the P- and N-NB systems on JT-60U [20, 21]. Very recent improvement of the JT-60U ECRF system [16] enables us to extend the region of electron heating experiment with higher P-NB power therefore with stronger ITBs. In the series of the experiments shown here electron heating is done by ECRF mainly. In order to assess high confinement at dominant electron heating regime, ECRF power is injected in addition to the positive NB power or substituting some NB power to raise the electron heating fractional power further. However, P-NB power can not be lowered too much since ITB becomes weakened and shrink in such cases. Although the ECRF power has been increased, it is not enough in some cases.

3.1. ITB in a Reversed Magnetic Shear Plasma

In this subsection, results obtained in RS plasmas are shown. Waveforms of a discharge with high electron heating fraction are shown in Fig. 3. During the I_p ramp-up, P-NB power was injected to form the RS configuration and the T_e, T_i and n_e ITBs. The injection of ECRF, about 3 MW in the discharge, started at the I_p flat-top. Up to the ECRF injection, discharge scenario is the same as usual P-NB heated RS discharge. The P-NB power was feedback controlled in order to achieve certain performance. After 5.8 s, the NB-power was fixed. As shown in the figure, the stored energy keeps increasing. As shown in the figure, due to the electron heating by ECRF the core T_e quickly increased and exceeded the core T_i. The core T_i rather decreased slightly due to step down of the P-NB power. Both electron and ion temperatures look saturated before disruption. The profiles of T_e, T_i, n_e and q at 6.5 s, just before the disruption, are shown in Fig. 3 (d) - (f). As shown in the figure, strong ITBs are confirmed in the T_e, T_i and n_e profiles. In the q profile,
the $q$ minimum is close to two. The disruption should be attributed to that the $q$ minimum is approaching or crossing two as usual P-NB RS discharges, but not to the electron heating. At 6.5 s, the $HH_{98(y,2)}$ factor is evaluated as 2.0. The $HH_{98(y,2)}$ factor is evaluated in the RS plasmas in this electron heating experiments. The $HH_{98(y,2)}$ factor is compared to that obtained in usual P-NB heated RS plasmas and plotted against a ratio $T_e/T_i$ in Fig. 4. Here the temperatures are evaluated at the center, since the profile inside the ITB is usually very flat in high confinement RS plasmas. As shown in the figure, the $HH_{98(y,2)}$ factor obtained in the electron heating experiments (shown by open circles) is comparable to that obtained in P-NB heated RS plasmas (shown by open squares). It should be noted that in the electron heating series data are taken with $I_p \sim 1.3$ MA, while in the P-NB cases $I_p$ 1.3 MA data is included.

In the results shown above, fraction of the electron heating power is changed after the ITBs are developed. Another issue in the electron heating is to investigate influence of the electron heating onto the ITB formation phase. Formation of $T_e$ ITB with electron heating dominant have been widely reported. Formation of $T_i$ ITB would be of interest and important. In order to assess $T_i$ ITB formation under dominant electron heating, ECRF was injected from very early phase in an RS discharge. The experiment was carried out in a hydrogen plasma with hydrogen NB. The waveforms of typical parameters are plotted in Fig. 5. The ECRF injection starts at 3.4 s almost the same timing as the NB injection starts. As shown in Fig. 5 (c), $T_e$ stays higher than $T_i$ in the core region through out the discharge. The $T_e$ profile at 3.6 s is shown in Fig. 5 (d) and the $T_e$ ITB is found to be formed already. The $T_e$ profiles (Fig. 5 (c)) at 3.9 and 4.1 s suggest that the $T_i$ ITB is not formed before 3.9 s. The $q$ profiles at each timing are shown in Fig. 5 (f). It should be noted that at 5.1 s, although 8.5 MW of P-NB is injected the absorbed power is about 5.5 MW with $P_{EC} \sim 3$ MW, the ration $P_e/P_i$ is about 0.9.

**FIG. 5**: Waveforms, (a) – (c), and profiles, (d) – (f) of an RS discharge with dominant electron heating from very early phase (E40795).
FIG. 6: Waveforms, (a) – (d) of typical parameters in an PS discharge with dominant electron heating (E41760). Also profiles of $T_e$, $T_i$, and $V_e$ are shown at 7.3, 7.7 and 7.8 s in (e) – (h).

The power fed to electrons by NB is not small due to faster thermal speed of hydrogen compared to that of deuterium of the same energy. It should be noted that at 5.1 s the $HH_{(n,2)}$ factor is evaluated as 1.2. These results suggest that in an RS plasma formation and sustainment is not seriously affected under dominant electron heating. Quantitative study is left for future issue.

3.2. ITB in a Positive Magnetic Shear Plasma

Electron heating experiments have also been carried out on ITBs in positive shear plasmas. Here positive shear ITB is referred to ITBs similar to those observed in high $\beta_p$ discharges in JT-6U. They are formed at positive magnetic shear region in a plasma. However, in order to make $T_e/T_i$ is closer to unity $F_{NB}$ is limited to modest level compared to that used in high performance high $\beta_p$ discharges. Since in high $\beta_p$ like discharges, $T_i$ tends to developed more than $T_e$. Also it should be noted that the target plasma are of L-mode edge plasmas due to lower input power and higher $B_i$ for ECRF resonance in this experiments. Waveforms of a typical discharge is shown in Fig. 6. Injection of NB starts at 6.5 s to form ITBs. In the plasma $q$ is monotonously decreasing towards the plasma center and near unity at the center. The formation of the ITB is rather slow in the discharge compared to that in higher performance high $\beta_p$ plasmas, due to limited $F_{NB}$. The ITB seems to be formed at around 7.2 s. Profiles of $T_e$ and $T_i$ at 7.3, 7.7 and 7.8 s are shown in Fig. 6 (e) – (g). As shown in the figure, a clear ITB structure is observed in both the $T_i$ and $T_e$ profiles at 7.3 s. At 7.7 s, the $T_i$ ITB looks eve clearer. Due to the electron heating by ECRF, $T_e$ is approaching to $T_i$ as shown in the waveforms and the profiles. However at 7.8 s, the $T_i$ profile becomes smooth at the location ($\rho \sim 0.4 - 0.5$)

FIG. 7: (a) Profile of $E_r$ evaluated at 7.4 and 7.7 s in E41760. (b) Temporal change of $(dE_r/d\tau)_{\rho \pi}$ and the minimum scale length of the $T_i$ profile.
where clear ITB was observed at earlier timing. The scale length of the $T_i$ profile ($L_{T_1} = T_i/(dT_i/dr)$) changes from about 15 cm at 7.7 s to 29 cm at 7.8 s (Fig. 7 (b)). These results indicate that the $T_i$ ITB is lost between 7.7 and 7.8 s. It should be noted that the $T_e$ profile looks smoother at 7.8 s and the $T_e$ scale length also becomes doubled, that means the $T_e$ ITB is lost as well. Waveforms of the toroidal rotation of carbon species ($V_i$) at $\rho \sim 0.25$ and 0.55 are plotted in Fig. 6 (d). The profile of $V_i$ is shown for 7.3, 7.7 and 7.8 s in Fig. 6 (h). As shown in the profile, the difference of $V_i$ between these two location can be representative of gradient of the $V_i$ profile around the ITB location inside the bottom of the notched $V_i$ structure. The profile of the radial electric field ($E_r$) is determined as, $E_r = -\frac{1}{eZ_i n_i} \nabla p_i + V_i R P_e - V_{p,i} B_t$, where $e$ is an electron charge, $Z_i$, $n_i$, $p_i$, $V_{i,i}$, $V_{p,i}$ are charge number, density and pressure, toroidal and poloidal rotational velocities of an ion species and $B_p$ and $B_t$ are the poloidal and toroidal magnetic fields. The $E_r$ profiles at 7.4 and 7.7 s are shown in Fig. 7 (a) they have a notched structure which is more steeply notched when the ITB is clearer. In the JT-60U ITB experiments, it is found that the shear of the $E_r$ profile is a key for formation and sustainment of the $T_i$ ITB [22]. Level, or strength, of the shear can be represented by a parameter $((dE_r/dr)_{max} + |(dE_r/dr)_{min}|)/2$ which is denoted as $(dE_r/dr)_{eff}$ in this paper. That roughly indicates how sharply or deeply the $E_r$ profile is digged. Change of $(dE_r/dr)_{eff}$ is plotted in Fig. 7 (b). As shown in the figure, $(dE_r/dr)_{eff}$ continues to decrease during ECRF injection and reached to the same level as that without the ITB, while $L_{T_1}$ stays small until 7.7 s. The results indicate that the ECRF injection influences the structure of $E_r$.

Influence of electron heating on the ITB formation was also investigated. In the experiment, $P_{EC}$ was injected prior to the NB injection. Furthermore $P_{NB}$ was scanned with the same $P_{NB}$. The waveforms of typical quantities are shown in Fig. 8. Again $I_p$ is 1 MA. Difference of $V_i$ between near the bottom of the notch and $\rho \sim 0.3$ ($\Delta V_i$) is shown in the figure as a measure of the gradient of the $V_i$ profile. As shown in the figure, $\Delta V_i$ starts increasing at about 6.8 s for lower $P_{EC}$ case and 7.3 s for higher $P_{EC}$ cases respectively. In the higher $P_{EC}$ case, slow growth of the ITB might have started earlier. Rapid increase in $\Delta V_i$ can be related to the formation of the $T_i$ ITB. It seems that increasing $P_{EC}$ delays the formation of ITB. As shown in the figure, temperature at the $\Delta V_i$ increase is different in both electrons and ions, higher for larger $P_{EC}$. However, a ratio of $\langle T_e \rangle_{core}$ to $\langle T_i \rangle_{core}$, where $\langle \cdot \rangle_{core}$ means volume average from the plasma center to the ITB foot, is almost the same value at the $\Delta V_i$ increase in two discharges. The ratio evaluated in E41674 at 6.9 s is 0.60 and that taken from E41696 at 7.3 s is 0.62. This may suggest that influence of electron heating can be related not only to the $E_r$ structure, as described above, but also
to the ratio between electron and ion temperatures. However, due to lack of both central heating NB and ECRF powers, scanned range is not wide enough. Further detailed investigation would be required for deeper understanding. Concerning the study of positive shear ITBs, more ECRF and on-axis P-NB power would be necessary. On axis P-NB helps to develop $E_r$ structure more, and more ECRF power enable high $T_e$ at such intense ion heating regime.

4. Summary

Results of studies towards steady state operation of an advanced tokamak accompanied by ITBs concerning fusion relevant conditions on JT-60U are shown. High confinement improvement at high normalized density regime which is expected in the ITER steady state operational scenario was obtained in a reversed magnetic shear (RS) plasma under full non inductive CD. It was demonstrated that off-axis LHCD could extend $\rho_{q_{\text{min}}}$ and $\rho_{\text{hot}}$ as the result. It was also indicated that n axis-N-NBCD could contribute to lower $q$ in the RS region. Influence of dominant electron heating on the ITB was investigated. It was found that in RS plasmas the confinement improvement by the ITBs could be maintained indifferent to the heating method. High enough $HH_{98(7,2)}$ factor of 2.0, which is comparable to that obtained in the dominant ion heating region, was obtained in the dominant electron heating region. On the other hand, it was found that the $T_i$ ITB could be degraded by dominant electron heating in positive shear plasmas. Electron heating would seem to influence the $E_r$ structure at the PS ITB.

Acknowledgement

The authors would like to acknowledge all the members of Japan Atomic Energy Research Institute who have contributed to the JT-60U project.

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Studies of current profile optimization and influence of electron heating towards advanced tokamak operation on JT-60U

Japan Atomic Energy Research Institute

- Contents -
1. Current profile optimization in an RS plasma
2. ITBs under dominant electron heating
3. Summary

Current profile optimization

An RS operation is one of candidates in ITER SS scenario
- desirable performance;
  - high confinement
  - larger bootstrap current fraction (f_{BS})
- require careful tailoring of current profile;
  - keep not very high q in central region
    - confinement of $\alpha$ particles
  - keep/expand $q$ minimum location
  - to keep ITB as wide as possible for better confinement

$\Rightarrow$ Explore possibility of desirable tailoring of RS current profile utilizing JT-60U external current drivers.

Combined full-CD by LHRF and N-NB in an RS plasma with large ITB radius

LHCD (2 GHz) + N-NBCD (E_b ~ 360keV)
$\Rightarrow V_{li} \sim 0V$, $\beta_n \sim 2.2$, $\beta_p \sim 2.1$, $T_e \sim T_i$
$q_{BS} = 6.9$

$\Rightarrow$ full non-inductive CD
$T_e$, $T_i$ and $n_i$, ITBs expand

j/q profile modification by external current drivers

Inner region
- N-NBCD (+ P-NB)
  - increases $j$ $\Rightarrow$ lower $q$
  - however in $p<0.3$, $q$ is still high
- LHCD
  - $L_H$ at just outside $q_{min}$
  - expands $\rho_{q_{min}}$

$\Rightarrow$ ITB expansion.

Outer region:

The results are demonstrated with high $f_{BS}$ (62%).
Extended full CD to high confinement and high density regime

$HH_{\text{HL},\rho} = 1.4$ at $n_e/n_{\text{low}} = 0.82$ is obtained.

Unique region among JT-60U full-CD experiments, high confinement AND high normalized density

- Well satisfy ITER core/CD requirements
- Issues left in particle heat handling
- $q_{95} = 6.9$ is to be lowered

ITBs under dominant electron heating

- In a burning plasma, main heating source is $\alpha$ particles. Therefore electrons are predominantly heated.
- On JT-60U, continuous development of electron heating system, ECRF (110 GHz, O-mode, $n_{\text{low}}$, and N-NB, enable us to explore dominant electron heating regime.
- We have studied:
  - influence on ITBs under dominant electron heating, or $T_e/T_i > 1$, regime.
  - ITBs in RS plasmas and Positive Shear (PS) plasmas

RS plasmas

Strong RS ITBs ($T_e$, $T_i$, and $n_e$) exist under dominant electron heating

Electron heating powers, COFF (and N-NB), are added or substituted for P-NB power to raise fractional electron heating power.
High confinement is assessed in dominant electron heating regime for RS plasmas

By changing combination of $P_{\text{EC}}$ and $P_{\text{NB}}$, $P_{\text{NB}}/P_{\text{EC}}$ is varied and $T_e/T_i$ follows it. $I_e \sim 1.3$ MA for ECRF $I_e \sim 0.8 - 1$ MA for NB

High thermal confinement of $HH_{\text{avg}} \sim 2$ is obtained independently on heating regime.

Positive Shear Plasmas

ITB formation under dominant electron heating

In a PS plasma, $T_i$ ITB is affected by injection of ECRF

Operational regime is limited
$\sim$ due to limited on-axis NB power.
Target ITBs are not as strong as the RS cases.
ECRF injection seems to affect $E_r(\rho)$ and $E_r(\rho)$ seems to affect the ITB.

Summary (continued)

- Effect of dominant electron heating onto ITBs is investigated.
  - RS plasmas:
    - No significant effect is observed for sustainment / formation.
    - An RS plasma can attain $HH_{98(\nu,2)} > 2$ both in $T_e/T_i < 1$, >1.
  - Positive shear plasmas:
    - ITB is found to be affected; degraded or lost.
    - $V_i => E$, structure seems to be affected and then ITB affected.
    - Further investigation should be necessary towards future plasmas.
      - is the effect intrinsic to ECRF?
      - is the effect large against stronger ITB?
        (more on-axis NB power would be required)

Summary

- Active modification of current profile in high confinement RS plasma with higher $f_{bg}$ is investigated.
  - full non-inductive current drive LHCD+NBCD with $f_{bg} = 63\%$
  - Peripheral CD (LHCD) works effectively to extend $\rho_{\text{min}}$ thus $\rho_{\text{bot}}$.
  - Inner CD (NBCD) works to lower $q$, except central region.
  - $HH_{98(\nu,2)} = 1.4$ at $n_e/n_{GW} = 0.82$, with $\beta_n = 2.2$, $q95 = 6.9$ is obtained.
1.6 Relationship between particle and heat transport in JT-60U plasmas with internal transport barrier


1) Japan Atomic Energy Research Institute, Naka Fusion Research Establishment, Naka-machi, Naka-gun, Ibaraki-ken, 311-0193, Japan
2) Princeton Plasma Physics Laboratory, Princeton, New Jersey, 08543-0451, U.S.A.

e-mail contact of main author: takenaga@naka.jaeri.go.jp

Abstract. The relationship between particle and heat transport in an internal transport barrier (ITB) has been systematically investigated in reversed shear (RS) and high $\beta_p$ ELMy H-mode plasmas in JT-60U. No helium and carbon accumulation inside the ITB is observed even with ion heat transport reduced to a neoclassical level. On the other hand, the heavy impurity argon is accumulated inside the ITB. The argon density profile estimated from the soft x-ray profile is more peaked, by a factor of 2-4 in the RS plasma and of 1.6 in the high $\beta_p$ mode plasma, than the electron density profile. The helium diffusivity ($D_{He}$) and the ion thermal diffusivity ($\chi_i$) are at an anomalous level in the high $\beta_p$ mode plasma, where $D_{He}$ and $\chi_i$ are higher by a factor of 5-10 than the neoclassical value. In the RS plasma, $D_{He}$ is reduced from the anomalous to the neoclassical level, together with $\chi_i$. The carbon and argon density profiles calculated using the transport coefficients reduced to the neoclassical level only in the ITB are more peaked than the measured profiles, even when $\chi_i$ is reduced to the neoclassical level. Argon exhaust from the inside of the ITB is demonstrated by applying ECH in the high $\beta_p$ mode plasma, where both electron and argon density profiles become flatter. The reduction of the neoclassical inward velocity for argon due to the reduction of density gradient is consistent with the experimental observation. In the RS plasma, the density gradient is not decreased by ECH and argon is not exhausted. These results suggest the importance of density gradient control to suppress heavy impurity accumulation.

1. Introduction

A reversed or weak positive magnetic shear plasma with internal transport barriers (ITBs) is a most promising operation mode for advanced steady-state operation due to its high bootstrap current fraction and high confinement. In JT-60U, the reversed shear (RS) plasma and the high $\beta_p$ mode plasma with weak positive shear have been optimized to provide a physics basis for ITER and SST-1 [1]. In these plasmas, further optimization for high density, high radiation loss fraction and high fuel purity are necessary, as well as high $\beta$ while keeping high confinement. Since these issues are closely related to the particle transport, understanding of the relationship between particle and heat transport is indispensable for the optimization.

In this paper, the relationship between particle (i.e., electron, helium, carbon and argon) and heat transport is systematically investigated in the high $\beta_p$ mode and RS plasmas. In the high $\beta_p$ mode plasma, anomalous transport is dominant. On the other hand, in the RS plasma, ion heat transport is reduced from the anomalous to the neoclassical level. The characteristics of particle and heat transport at the ITB are discussed in Sec. 2. In Sec. 3, the ITB controllability is studied, followed by a summary in Sec. 4.

2. Characteristics of particle and heat transport at internal transport barrier

Typical profiles of temperatures ($T_e$, $T_i$), safety factor ($q$), and densities ($n_e$, $n_{He}$, $n_C$ and $n_{Ar}$) are shown in Fig. 1 for (a) the RS plasma ($L_p$=1.3 MA, $B_t$=3.7 T, R=3.3 m, a=0.8 m, $\kappa$=1.5, $\delta$=0.18-0.26 and HH$^2$=1.6) and (b) the high $\beta_p$ ELMy H-mode plasma ($L_p$=1.0 MA, $B_t$=2.3-8 T, R=3.4 m, a=0.8 m, $\kappa$=1.4, $\delta$=0.38-0.46 and HH$^2$=1.0). The puffed He and intrinsic C densities are measured with CXRS. The profile of the total Ar density summed over all ionization states is estimated using an impurity transport code, where the transport coefficient
is determined by fitting the calculated soft x-ray profile to the measurement [2]. The Ar radiation coefficient is taken from the ADAS database [3] considering the JT-60U diagnostic setup. In the RS plasma, a box-type profile with a strong ITB is observed in \( n_a(r) \), \( T_a(r) \) and \( T_s(r) \). An ITB is also observed in \( n_{\mbox{He}}(r) \). However, \( n_{\mbox{He}}(r) \) is flatter than \( n_a(r) \), which is favorable for helium ash exhaust. The \( n_a(r) \) shape is similar to that for \( n_s(r) \), suggesting no carbon accumulation inside the ITB. In the RS plasma with a small amount of Ar puffing, the soft x-ray profile becomes a peaked one. In order to fit the calculated soft x-ray profile to the measurement, a more peaked (by a factor of 2.6) \( n_{\mbox{He}}(r) \) inside the ITB than \( n_a(r) \) is necessary, as shown in Fig. 1 (a). This result indicates the Ar accumulation inside the ITB.

In the high \( \beta_p \) mode plasma, \( n_a(r) \), \( T_a(r) \) and \( T_s(r) \) have a parabolic-type profile. Both \( n_{\mbox{He}}(r) \) and \( n_c(r) \) are flat, also suggesting no helium and carbon accumulation. On the other hand, \( n_{\mbox{He}}(r) \) is more peaked by a factor of 1.6 than \( n_a(r) \), which is, however, a smaller factor than that in the RS plasma.

Figure 2 shows \( -\nabla n/n \) as a function of \( -\nabla T/T \) for \( n_a \), \( n_{\mbox{He}} \), \( n_c \), and \( n_{\mbox{Ar}} \) at the ITB. In the high \( \beta_p \) mode plasma, \( -\nabla n/n \) and \( -\nabla T/T \) are limited to small values compared with those in the RS plasma. Although the error range is large due to scattering of the measured \( n_{\mbox{He}} \), \( -\nabla n_{\mbox{He}}/n_{\mbox{He}} \) has a negative value. The value of \( -\nabla n_a/n_a \) is around zero, and is smaller than \( -\nabla n_a/n_a \). Due to the Ar accumulation, \( -\nabla n_{\mbox{Ar}}/n_{\mbox{Ar}} \) is larger than \( -\nabla n_a/n_a \). On the other hand, in the RS plasma, \( -\nabla n_a/n_a \) increases with \( -\nabla T/T \) from the same region as that for the high \( \beta_p \) mode plasma, and it is similar to \( -\nabla T/T \). The value of \( -\nabla n_a/n_a \) is saturated in the range of large \( -\nabla T/T \), and it is smaller than \( -\nabla n_a/n_a \). When He is fuelled inside the ITB using a He beam, \( -\nabla n_{\mbox{He}}/n_{\mbox{He}} \) is larger than that with He gas-puffing. However, even with the He beam, \( -\nabla n_{\mbox{He}}/n_{\mbox{He}} \) is still smaller than \( -\nabla n_a/n_a \). The value of \( -\nabla n_c/n_c \) is almost the same as \( -\nabla n_a/n_a \), and \( -\nabla n_{\mbox{Ar}}/n_{\mbox{Ar}} \) is larger than \( -\nabla n_a/n_a \) due to the Ar accumulation. The higher Z impurity has larger \( -\nabla n/n \) in the RS plasma.

Next, the relationship between particle and thermal diffusivities is discussed. The electron effective diffusivity \( (D_e^{\mbox{eff}}) \), defined considering only the diffusion term, is well correlated with the ion thermal diffusivity \( (\chi_i) \) in both high \( \beta_p \) mode and RS plasmas. The ratio \( (D_e^{\mbox{eff}}/\chi_i) \) is estimated to be 0.2-0.3 in the high \( \beta_p \) mode plasma and 0.04-0.2 in the RS plasma. In order to understand the relationship, the heat and particle fluxes are estimated based on the linear stability analysis using the FULL code [4] in the box-type RS plasma. The FULL code analysis indicates that a positive linear growth rate still remains at the ITB region even with sheared EXB rotation effects. The 2D full wave code analysis [5] also shows the existence of 0.5-2% density fluctuations around the ITB based on O-mode reflectometer measurement [6]. The FULL code estimates the ratio of particle flux to the electron heat flux \( (T_e^0/T_e^0/q_0) \) to be...
around unity in the ITB region, and the ratio drops to a value just slightly negative outside the ITB. Although the experimental profile of the ratio decreases much smoothly from around unity in the ITB region to about 0.1 outside the ITB, it shows a similar tendency. The ratio of ion anomalous heat flux to electron heat flux \( \frac{q_i}{q_e} \) has a similar value \( \frac{q_i}{q_e} = 5-7 \) for both experiment and calculation outside the ITB. However, in the ITB region, \( \frac{q_i}{q_e} \) for the calculation is much smaller than that for the experiment. A possible reason for the disagreement is that the FULL code analysis represents the linear limit with a single value of \( k_i \rho_i \) based on a local theory. The non-linear effects, the effects summed over all values of \( k_i \rho_i \) and non-local effects should be investigated to understand physical mechanisms responsible for the relationship between particle and heat transport.

The relationship between the impurity diffusivity normalized by the neoclassical value \( D/D_{NC} \) and \( \chi/\chi_{NC} \) in the ITB is shown in Fig. 3. The values of \( D \) and the convection velocity \( (v) \) of He are estimated separately based on the He gas-puffing modulation experiment [7]. Since it is difficult to separate \( D \) and \( v \) for C and Ar experimentally, \( D_C \) and \( D_A \) are estimated by assuming the neoclassical \( v (v_{NC}) \) and a steady-state condition. In the RS plasma, a similar \( v \) to the neoclassical value has been obtained for C and Ne [8, 9]. The impurity neoclassical transport coefficient is calculated using NCLASS [10]. The value of \( D_{He} \) is estimated to be 0.5-1.0 m/s in the high \( \beta_p \) mode plasma and 0.1-0.5 m/s in the RS plasma. The ratio \( D_{He}/\chi \) is in the range of 0.2-1.0 for both RS and high \( \beta_p \) mode plasmas. The value of \( D_{He} \) is reduced to the neoclassical level in the box-type RS plasma, where \( \chi \) is also reduced to the neoclassical value. In the parabolic-type RS plasma and high \( \beta_p \) mode plasma, \( D_{He} \) and \( \chi \) are higher by a factor of 5-10 than \( D_{He}^{NC} \) and \( \chi_{NC} \). The value of \( v \) is consistent with the neoclassical theory (from -0 to -1 m/s) in not only box-type but also parabolic-type RS plasmas. On the other hand, an outward \( v \) is observed in the high \( \beta_p \) mode plasma, while the neoclassical theory predicts a small inward \( v \). The outward \( v \) is consistent with the negative value of \( -\nabla n_{He}/n_{He} \) shown in Fig. 2.

The values of \( D_C^{NC} \) and \( D_A^{NC} \) are estimated to be in the range 0.01-0.04 m/s at the ITB. The value of \( v_{NC} \) at the ITB is larger for Ar (-0.2 to -0.8 m/s in the high \( \beta_p \) mode plasma and -2 to -5 m/s in the RS plasma) than for C (-0 to -0.2 m/s in the high \( \beta_p \) mode plasma and -0 to -1.3 m/s in the RS plasma). The steady-state \( n_C(r) \) and \( n_A(r) \) calculated using \( D=D_{NC} \) at ITB and \( D=D_{NC} \) in the other region with \( v_{NC} \) are more peaked than the measurements, even in the box-type RS plasma. Although the experimental \( n_C(r) \) and \( n_A(r) \) is not in steady state in the box-type RS plasma, the experimental \( n_C(r) \) and \( n_A(r) \) is not as peaked as the neoclassical prediction even when time evolution is considered. In order to adjust the calculated steady-state profile to the measurement with \( v_{NC} \), larger \( D_C \) and \( D_A \) are necessary in the ITB region. In the box-type RS plasma, where \( \chi \) is reduced to the neoclassical level, \( D_C/D_{NC}^{NC} \) and \( D_A/D_{NC}^{NC} \) are estimated to be -4 and -9, respectively. The values of \( D_C \) and \( D_A \) are estimated to be 0.1 and 0.2 m/s, respectively. These values are similar to the range of \( D_{He} \) (0.1-0.3 m/s). In the high \( \beta_p \) mode plasma, \( D_C \) and \( D_A \) are estimated to be about 0.1 m/s. The values of \( D_C/D_{NC}^{NC} \) and \( D_A/D_{NC}^{NC} \) are about 4, which is similar to \( \chi/\chi_{NC}^{NC} = 5 \). In some high \( \chi/\chi_{NC}^{NC} \) cases in the parabolic-type RS plasma, \( D_i/D_{NC} \) is estimated to be more than 10. However, in other high \( \chi/\chi_{NC}^{NC} \) cases, \( n_C(r) \) can not be reproduced with \( v_{NC} \). The anomalous \( v \) might be dominant in this region.

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**Fig. 3 Relationship between \( D_{He}/D_{He}^{NC} \) and \( \chi/\chi_{NC}^{NC} \). Open circles show the data in high \( \beta_p \) mode plasma. Closed circles and squares show the data in parabolic- and box-type RS plasmas, respectively. \( D_C/D_{NC}^{NC} \) and \( D_A/D_{NC}^{NC} \) are also plotted, which are estimated by assuming \( v_{NC} \).**
3. Control of internal transport barrier

Although the accumulation is smaller than the neoclassical prediction, the Ar accumulation inside the ITB should be suppressed. In order to control the Ar accumulation inside the ITB, ECH is applied. The density clamp by ECH is a common phenomenon not only in tokamaks but also in helical devices [11, 12]. In addition, a decrease of $T_i$ in the ITB is observed in JT-60U high $\beta$ mode plasma [13]. Figure 4 (a) shows the time evolution of the high $\beta$ mode plasma, where a small amount of Ar was puffed at $t=6.5$ s and ECH was applied inside the ITB from $t=8$ s. The value of $T_i(0)$ is increased to the same value as $T_i(0)$ during ECH. The density is substantially decreased by applying ECH while maintaining $T_i(0)$. In a similar discharge without Ar puffing, a decrease of $T_i$ is observed during ECH with small change of $n_i$. Ar could affect the large reduction of $n_i$ during ECH. Although the thermal confinement is decreased from $HH_i=1.9$ at $t=7.95$ s to 0.9 at $t=8.45$ s, the central soft x-ray signal is drastically reduced by a factor of more than 2. This observation indicates the Ar exhaust from the inside of the ITB. In the RS plasma with Ar puffing, ECH was also applied from $t=5.8$ s as shown in Fig. 4 (b), where NB power was decreased from $t=6.0$ s. The confinement is improved by applying ECH until the NB power is decreased. After the NB power is stepped down, the particle fuelling becomes small, but the density does not substantially decrease. The stored energy gradually decreases and $HH_i$ is estimated to be 1.6 at $t=6.5$ s. The strong ITBs in $n_i(t)$ and $T_i(t)$ remain and the soft x-ray signal does not decrease even after the NB power is stepped down, suggesting no Ar exhaust from the inside of the ITB.

Figure 5 (a) and (b) show $n_i(t)$ and $T_i(t)$, respectively, before and during ECH in the high $\beta$ mode plasma. The $n_i$ ITB is almost lost, while the $T_i$ ITB is kept, although the ITB position moves inward. The decrease of the density gradient leads to the reduction of $v^{NC}$ as shown in Fig. 5 (c). The value of $D_{\alpha}^{NC}$ during ECH is almost the same as that before ECH. Figure 5 (d) shows $n_i(t)$ reproduced using a different $v^{NC}$ and the diffusivity of $D=4\times D_{\alpha}^{NC}$ in the ITB and $D=0.5-1$ m$^2$/s in the other region. A value of $n_i(0)$ is substantially reduced compared with $n_i(0)$. These profiles are consistent with the
soft x-ray measurements both before and during ECH as shown in Fig. 5 (e). A possible mechanism for the Ar exhaust is as follows. First, the density is clamped by ECH and the density gradient becomes small. Then, Ar is exhausted due to the reduction of $v_{\text{sc}}$. Also, the Ar exhaust reduces the density. These processes could interact as a positive feedback loop for Ar exhaust. These results indicate the importance of the density gradient control to suppress the argon accumulation. In the RS plasma, $n_s(r)$ is not changed and $n_s(r)$ is still more peaked by a factor of 3-4 than $n_s(r)$ during ECH. This result suggests that there is no Ar exhaust by ECH in the RS plasma. The development of the density gradient control method in the RS plasma is crucial for suppression of Ar accumulation.

The possibility of density gradient control in the RS plasma is shown with pellet injection. When a high field side pellet is injected into the box-type RS plasma, a clear reduction of density fluctuations is observed. The 2D full wave code shows that the fluctuation level is reduced from 1.3% to 0.8%, assuming $k_y=k_z=3 \, \text{cm}^{-1}$. The dependence of the density fluctuation level on $k_y$ and $k_z$ is checked using an analytical solution of the time-dependent 2D full-wave equation [14]. The density fluctuation level before the pellet injection varies in the range 0.8-1.8% by changing $k_y$ and $k_z=1-5 \, \text{cm}^{-1}$. The ion temperature profiles are not substantially different before and after the reduction of the density fluctuations. While, the density increases inside the ITB and the density gradient becomes large in the ITB region. The value of $n_s(0)$ is also increased. In this discharge, a cold pulse induced by pellet ablation outside the ITB is propagated into the ITB, and the density fluctuations are reduced. Conversely, local heating outside the ITB might work for the reduction of the density gradient.

4. Summary

The relationship between particle and heat transport is systematically investigated at the ITB in reversed shear and high $\beta_p$ mode plasmas. The value of $\langle -\nabla n/n \rangle$ is smaller than $\langle -\nabla T/T \rangle$ for helium, similar for electrons and carbon, and larger for argon, in the RS plasma with a box-type profile. The value of $D_{\text{ep}}$ is reduced to a neoclassical level together with $\chi_c$. On the other hand, 4-9 times larger carbon and argon diffusivities than the neoclassical value are evaluated, even when $\chi_c$ is reduced to the neoclassical level. Ar is exhausted from the inside of the ITB by applying ECH in the high $\beta_p$ mode plasma, while thermal confinement is reduced from $HH_{\alpha}=1.0$ to 0.9. In the RS plasma, the electron density profile is not changed and argon is not exhausted by ECH. These results indicate that density gradient control is important in suppressing impurity accumulation inside the ITB.

Acknowledgement

The authors wish to thank Dr. R. Dux for calculating the soft x-ray emission rate of argon and Dr. W. A. Houlberg for use of the NCLASS.

References

Relationship between particle and heat transport in JT-60U plasmas with internal transport barrier


Japan Atomic Energy Research Institute

K. W. Hill, G. Rewoldt, G. J. Kramer, R. Nazikian

Princeton Plasma Physics Laboratory

19th IAEA Fusion Energy Conference

JAERI

OUTLINE

- Characteristics of particle and heat transport at ITB
- Ion heat transport vs. electron, He, C and Ar transport
- ITB control
  - Effects of ECH on impurity transport
    - Reversed shear plasma
    - Anomalous dominant to neoclassical level
    - High $\beta_p$ mode plasma (weak positive shear)
    - Anomalous dominant

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Introduction

Reversed shear (RS) plasma with
Weak positive shear plasma
(High $\beta_p$ mode plasma)
High bootstrap fraction
High confinement

- Extension of the achieved parameters
  - Fueling
  - Impurity seeding
  - Sufficient He ash exhaust
  - Reduction in impurity

- Control of ITB
  - Development of the ITB control method in a burning plasma where fuel control is important

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Wave-forms in RS and high $\beta_p$ mode plasmas

- Higher confinement is achieved in RS plasma, but higher $\beta$ is obtained in high $\beta_p$ mode plasma.
- Fuel purity is lower in RS plasma than in high $\beta_p$ mode plasma.
- Modulated Helium gas was put to estimate the particle diffusivity ($D$) and the convection velocity ($v$) separately.

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Density & Temperature & Diffusivity profiles

- No He and C accumulation inside the ITB even with Box-type profiles in RS plasma

Reversed shear (Box-type)

Reversed shear (Box-type)

He flattens than $n_B(r)$

C same as $n_B(r)$

High $\beta_p$ mode

(Parabolic-type)

$D_{He}$ and $D_{e eff}$ ($D_{He} = D_{e eff} \times n_{He}$)

reduced at ITB region as well as $\tau$ ...

RS < High $\beta_p$ mode

Calculated soft X-ray profile with Ar puff

- The Ar density profile estimated from the soft-x-ray profile is more peaked by a factor of 2.6 in the RS plasma and 1.5 in the high $\beta_p$ mode plasma than the electron density profile.

Ar puff in RS and high $\beta_p$ mode plasmas

- Soft x-ray profile is peaked in RS plasma, where central chord signal increases, while edge chord signal is kept at a constant value.

- Central chord signal gradually increases in the ELMy phase for high $\beta_p$ mode plasma.

Relationship between $\nabla T_i / T_i$ and $\nabla n / n$

- $(-\nabla n / n)$ is smaller than $(-\nabla T_i / T_i)$ for helium, similar for electron and carbon, and larger for argon.

- When He is fuelled inside the ITB by He beam, $(-\nabla n/ n)$ is larger than that with He gas-puffing. However, $(-\nabla n/ n)$ is still smaller than $(-\nabla T_i / T_i)$.

- Is higher $(-\nabla n / n)$ for higher Z impurity consistent with neoclassical theory, which predicts accumulation of high Z impurity?
Relationship between $D_{\text{He}}$ and $\chi_i$

- $D_{\text{He}}$ seems to be linked with $\chi_i$, however, $D_{\text{He}}/\chi_i$ varies from 0.2 to 1.0.
- $D_{\text{He}}$ is reduced to the neoclassical level in the box-type RS plasma together with $\chi_i$.
- The value of $v$ is consistent with the neoclassical theory (from -0.1 to -1.3 m/s) in not only box-type but also parabolic-type RS plasmas.
- The outward $v$ is observed in the high $\beta_p$ mode plasma, while neoclassical theory predicts a small inward $v$.

Relationship between $D_{\text{He}}$ and $\chi_i$

- $D_{\text{He}}$ is well correlated with $\chi_i$.
- $D_{\text{He}}/\chi_i$ = 0.2-0.3 for high $\beta_p$ mode plasmas
- $D_{\text{He}}/\chi_i$ = 0.1-0.2 for RS plasma with box-type profiles
- $D_{\text{He}}/\chi_i$ = 0.04-0.1 for RS plasma with box-type profiles
- $D_{\text{He}}$ is reduced to the neoclassical level when $\chi_i$ reduce to the neoclassical level, suggesting importance of the inward convection velocity.

**FULL code analysis in RS plasma**

- The ratio $\Gamma_T q_0$ estimated by FULL code shows similar tendency to the experiment. However, the ratio of anomalous $q_0$ ($q_0^{\text{q0}}$) to $q_0$ is smaller than the experiment at the ITB.
- FULL code indicates linear growth rate still remains at the ITB even with sheared E×B rotation effects in the box-type RS plasma.
- The 2D full wave code analysis also shows existence of 0.5-2% density fluctuation around the ITB based on reflectometer measurement.

Relationship between $D_{\text{C,Ar}}/D_{\text{C,Ar}}^{\text{NC}}$ and $\chi_i/\chi_i^{\text{NC}}$

- The theory gives more peaked $n_0(t)$ and $n_A(t)$.
- $D_{\text{C}}$ and $D_{\text{Ar}}$ estimated by assuming $\chi_i^{\text{NC}}$ is higher by a factor of ~4 for $D_{\text{C}}$ and ~9 for $D_{\text{Ar}}$, respectively, at the ITB in the box-type RS plasma.
- In the high $\beta_p$ mode plasma, $D_{\text{C}}/D_{\text{C}}^{\text{NC}}$ and $D_{\text{Ar}}/D_{\text{Ar}}^{\text{NC}}$ are estimated to be ~4, which is similar to $\chi_i/\chi_i^{\text{NC}}$=5.
ITB control by ECH in high $\beta_p$ mode plasma

- The density is decreased by applying ECH inside the ITB and the $n_e$ ITB is almost lost with keeping the $T_i$, ITB.
- The central soft x-ray signal is drastically reduced by ECH, which suggests the Ar exhaust inside the ITB.
- The stored energy and neutron yield are kept at constant values, because the beam component is increased due to a long slowing down time.
- Thermal confinement is reduced.

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ITB response to ECH in RS plasma

- The density profile is not changed, and the soft x-ray signal is not decreased, suggesting no Ar exhaust by ECH in the RS plasma.
- The comparison of the soft x-ray profile between measurement and calculation indicates that Ar profile is still more peaked by a factor of ~4 than the electron density profile during ECH.
- $\nu$n control in the RS plasma is crucial.

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Ar exhaust in the high $\beta_p$ mode plasma

- The reduction of $v_{Ar}^{NC}$ due to the reduction of $\nu$n is consistent with the measured soft x-ray profile.
- Possible mechanism for the Ar exhaust density clump by ECH: reduction of $\nu$n $\rightarrow$ reduction of $v_{NC}$

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ITB control by pellet injection in RS plasma

- The possibility of $\nu$n control in the RS plasma is shown with a high-field-side pellet injection.
- The high frequency component of reflectometer signal is drastically reduced.
- The density fluctuation substantially reduced from 0.5-1.0% to 0.3-0.8%.
- Density gradient increases without large change of $T_i$.

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- Cold pulse induced by a pellet ablation outside the ITB is propagated into the ITB region, and clear reduction of density fluctuation is observed. Conversely, local heating at ITB foot or outside the ITB might work for the reduction of $\nu$n.
Summary

- Characteristics of particle and heat transport at ITB
  - No helium and carbon accumulation is observed inside the ITB.
  - $\Phi$ is reduced to the neoclassical level inside the ITB.
  - $Z_\alpha$ is reduced to the neoclassical level together with $Z_e$.
  - $\alpha$ and $\beta$ density profile from measurements even when $Z_\alpha$ is reduced to the neoclassical level.
  - $\Phi$ is exhausted from the inside of the ITB by ECH in the high $\beta$ mode plasma.
  - The reduction of neoclassical inward velocity due to the reduction of density gradient is consistent with the measurements.
  - The density profile is not changed and $\Phi$ is not exhausted by ECH in the RS plasma.
  - These results indicates that the density gradient control is important to suppress the impurity accumulation inside the ITB.
1.7 Fast Dynamics of Type I ELM and Transport of ELM Pulse in JT-60U


Japan Atomic Energy Research Institute, Naka-machi, Naka-gun, Ibaraki 311-0193, Japan

c-mail contact of main author: oyaman@fusion.naka.jaeri.go.jp

Abstract. Simultaneous fast ELM measurements using reflectometer, interferometers, $D_a$ intensity and magnetic probe reveal the detailed characteristics of type I ELMs. From the phase signal of the reflectometer indicating the radial movement of the cut-off layer, four different phases in an ELM event and radial extent of the collapse of density pedestal up to twice the pedestal width were observed. Detailed edge density measurements at high- and low-field side consistently demonstrate the asymmetry of collapse of density pedestal due to type I ELM, which is localized at the low-field side (LFS) midplane. Expelled particles from LFS midplane were measured by using LFS midplane and X-point Mach probes together with IRTV. Time delay of the enhanced ion saturation current between the two probes, and time constant of the peak heat load and the enhancement of the plasma flow toward divertor target suggest that the convective plasma flow provides large contribution to the ELM heat load.

1. Introduction

The study of the collapse of the pedestal structure due to type I ELMs [1-3] is important to clarify its mechanism and to reduce its severe heat loads on the divertor target, which can erode the divertor target plate in the next step devices such as ITER [4]. Therefore, the mitigation of the large ELM heat load on the divertor target is one of the most important issues to be overcome on ITER [5]. Since the ELM heat load strikes the divertor target not as a time-averaged load but as an instantaneous heat pulse, not only the evaluation of ELM energy but also the time scale of the collapse and transport from the viewpoint of individual ELM pulse is very important. In JT-60U, the study of detailed dynamics of the collapse of the density pedestal [6] has been started using O-mode reflectometer [7]. Fast time series analysis established the characteristic time scale of MHD oscillation and $D_a$ spike in different density regimes [8]. Expelled particles from the plasma have been measured using scrape-off layer (SOL) Mach probe and fast infrared TV camera (IRTV) [9]. These kinds of detailed ELM measurement can provide important information to develop theoretical models of ELMs. In this paper, we report the fast dynamics of type I ELM and transport of ELM pulse in the SOL region from the viewpoint of individual ELM pulse using effective diagnostics with high temporal resolution in JT-60U to provide better understanding of the type I ELM physics.
2. Fast Time Series Analysis During an ELM

The measurement of the collapse of the density pedestal due to type I ELMs was performed on high-$\beta$, H-mode discharges [10] with the following plasma parameters: plasma current $I_p = 1.8$ MA, toroidal magnetic field $B_T = 4.0$ T, NBI heating power $P_{\text{NBI}} = 22$ MW, triangularity $\delta \sim 0.32$, elongation $k \sim 1.6$ and safety factor at 95% flux surface $q_{95} \sim 3.9$. Figure 1 shows the plasma configuration of the discharge together with the line of sight of some diagnostics relevant to this study.

The phase angle change of reflectometer signal, which implies the radial movement of the cut-off layer, suggests the existence of four different phases in an ELM event: a precursor phase, collapse phase, recovery phase and a relaxation phase, as shown in FIG. 2(a) and (f). At first, coherent density precursor, whose amplitude increased with time, was clearly observed before the onset of an ELM. The time scale of the precursor phase was 200–500 $\mu$s in the frequency range of 5-25 kHz. On the other hand, no magnetic precursor was observed so far as shown in FIG. 2(b). In the collapse phase, the cut-off layer moved inward (positive phase change) as a result of the collapse of the density pedestal within 100–350 $\mu$s. It is noted that two vertical interferometers exhibited only small

![Diagram of plasma configuration and reflectometer signal](image)

**FIG. 1.** Plasma configuration together with the line of sight of the FIR interferometer, Thomson scattering, $D_\alpha$ intensity at both divertor, the poloidal position of the magnetic probe (MP) and the reflectometer.

**FIG. 2.** Waveforms of (a) phase change of the reflectometer signal, (b) $D_\alpha$ intensity at inner (thin) and outer (thick) divertor, (c) magnetic fluctuation at outer midplane, (d) and (e) line-integrated density at high and low-field side, respectively. (f)-(j) show expanded waveforms between two dotted lines in (a)-(e) during 1.5s. Sampling time of the reflectometer, $D_\alpha$ intensity and magnetic probe is 1 $\mu$s. That of the interferometer is 5 $\mu$s. All diagnostics are synchronized within 5 $\mu$s.
oscillations in the collapse phase as shown in FIG. 2(i) and (j). Then, in the recovery phase, the cut-off layer moved outward as pedestal density first recovered and then went beyond the initial position over 200–500 µs. This overshoot resulted from the increase in the edge plasma density due to enhancement of divertor recycling and ionization as shown in FIG. 2(d). Finally, it gradually returned to a similar position as the initial one, taking 6–10 ms in the relaxation phase. The time scale of this relaxation phase is comparable to the time it takes the FIR1 signal to return to the initial level as shown in FIG. 2(a) and (d).

The total phase change of the reflected signal, \( \phi \), under the WKB approximation is

\[
\phi \equiv \frac{2 \omega}{c} \int_0^{x_c} \left( 1 - \frac{n(x)}{n_c} \right)^{1/2} dx - \frac{\pi}{2},
\]

where \( n_c \) is the cut-off density, \( \omega \) is the frequency of the incident wave and \( x_c \) is the distance from the antenna to the cut-off position. We can evaluate the radial movement of the cut-off layer using equation (1) and assuming possible density profiles satisfying the observed phase change. In the case of pedestal measurements in H-mode plasmas, the observed phase change is mainly attributed to the radial movement of the cut-off layer and to a smaller extent to changes in the density profile along the line of sight of the reflectometer. Figure 3 shows an example of estimation of radial movement of the cut-off layer for an ELM shown in FIG. 2. The cut-off position moved 7.5 cm inside the separatrix. This suggests that the radial extent of the collapse at the top of density pedestal reaches as large as twice the original pedestal width. The effect of the variation in the plasma equilibrium during ELMs was found to be less than 4 mm, which is 10 times smaller than estimated radial movement of the cut-off layer. For poloidally uniform density collapse, the number of lost particles by an ELM shown by the hatched region in FIG. 3 is estimated to be \( 1.0 \times 10^{20} \). Since the particle source during an ELM (~400 μs) was less than \( 2.0 \times 10^{18} \), this large particle loss should provide large reduction of FIR1 signal by \( \delta n_l \approx 0.48 \times 10^{19} \text{ m}^{-2} \). As mentioned above, however, no considerable reduction was observed during the collapse phase. This result suggests the asymmetry of the collapse of density pedestal by type I ELM.

3. Poloidal Asymmetry of Collapse of Density Pedestal by Type I ELM

In order to confirm the asymmetric collapse of pedestal structure due to type I ELM, three kinds of edge density measurements were performed. At first, simultaneous measurements of
edge density on the high-field side (HFS) and low-field side (LFS) plasma were performed using FIR1 and reflectometer, respectively, with the following plasma parameters: \( I_p = 1.0 \) MA, \( B_T = 2.0 \) T, \( P_{NBI} = 7.4 \) MW, \( \delta = 0.35 \), \( \kappa = 1.4 \) and \( q_{95} \sim 2.9 \). In this plasma configuration shown in FIG. 4(a), the line-integrated density on FIR1 is quite sensitive to the change of pedestal density, since half of the line of sight of FIR1 passed through the top of the density pedestal. The collapse of the density pedestal was manifested as fast inward movement of the cut-off layer at LFS midplane and the increase of \( D_\alpha \) intensity at the timing shown by dotted line in FIG. 4(c).

According to the estimation of radial extent from the phase change of the reflectometer shown in FIG. 4(b), the line-integrated density on FIR1 signal would be reduced by \( \delta n_l \sim 0.57 \times 10^{19} \) m\(^{-2} \), if the collapse of density pedestal was a poloidally symmetric event. There was, however, no immediate response on FIR1 before the increase of \( D_\alpha \) signal. Instead, the FIR1 signal usually showed a small reduction after about 90 \( \mu \)s average time delay. This time delay was comparable to the \( L_i/C_s \sim 95 \) \( \mu \)s, where \( L_i \) is the connection length of \( \sim 25 \) m between LFS and HFS midplane, and \( C_s \) is the ion sound velocity of \( \sim 2.8 \times 10^4 \) m/s. Therefore, the delayed reduction can be explained by the parallel flow from the inside to outside in order to fill the density gap as a result of localized particle expulsion at LFS midplane.

In order to confirm the absence of the direct particle loss corresponding to the collapse of the density pedestal at the HFS midplane, separatrix/SOL density measurement using

\[ \text{FIG. 4. (a) Plasma configuration. (b) Initial density profile (thick) and assumed density profile during collapse phase (thin). (c) Time evolution of phase change of reflectometer signal, } D_\alpha \text{ intensity and line-integrated electron density.} \]

\[ \text{FIG. 5. (a) Plasma configuration together with the line of sight of FIR1. (b) Time evolution of FIR1 signal and } D_\alpha \text{ intensity at both divertor targets.} \]
FIR1 was performed on the outward shifted plasma with the following plasma parameters: \( I_p = 1.2 \) MA, \( B_T = 2.5 \) T, \( P_{\text{NBi}} = 12.3 \) MW, \( \delta \sim 0.40 \), \( \kappa \sim 1.3 \) and \( q_{95} \sim 3.0 \). Since the integration length of separatrix/SOL region of FIR1 is about 0.7 m in this plasma configuration as shown in FIG. 5(a), such a small density increase of \( \delta n_e = 0.5 \times 10^{18} \text{ m}^{-3} \) should provide a detectable increase of \( \delta n_l \sim 3.7 \times 10^{17} \text{ m}^{-2} \) prior to the increase of \( D_\alpha \) signal. However, there was no density increase before the increase of \( D_\alpha \) signal as shown in FIG. 5(b).

Further evidence for the asymmetric collapse of density pedestal was obtained after a modification of the reflectometer system. When we change the polarization of the incident wave of the X-mode reflectometer [11], the system acts as a millimeter-wave interferometer (MWI) with the cut-off density of \( n_e = 1.7 \times 10^{20} \text{ m}^{-3} \). In order to avoid the refraction by the plasma, the horizontal interferometer (MWI) was applied to relatively small plasma as shown in FIG. 6 with the following plasma parameters: \( I_p = 0.9 \) MA, \( B_T = 2.5 \) T, \( P_{\text{NBi}} = 7.1 \) MW, \( \delta \sim 0.13 \), \( \kappa \sim 2.5 \).

![FIG. 6. Plasma configuration together with the line of sight of two vertical interferometers, millimeter-wave horizontal interferometer and \( D_\alpha \) intensity at both divertor targets.](image)

![FIG. 8. Density profile before ELM (thick line). Estimated density profile to satisfy observed line-integrated density reduction on horizontal interferometer assuming linear density profile at pedestal region in two cases: poloidal uniform case (thin line) and only LFS case (dashed line).](image)

![FIG. 7. Time evolution of (a)-(c) three interferometers and (d) \( D_\alpha \) intensity at both divertor targets.](image)
1.6 and \( q_{95} \approx 3.8 \). Figure 7 shows the time evolution of the line-integrated density of three interferometers together with \( D_\alpha \) intensity at both divertor targets. Vertical solid line shows the onset of the ELM defined as the time of the increase in \( D_\alpha \) intensity. At the time indicated by the dashed line, horizontal interferometer exhibited fast density loss of \( \delta n_l \approx 0.47 \times 10^{18} \text{ m}^{-2} \). Assuming poloidally uniform density loss with the profile shown by the thin solid line in FIG. 8 to satisfy the observed reduction of MWI signal, FIR1 signal should be reduced by \( \delta n_l \approx 2.8 \times 10^{18} \text{ m}^{-2} \) (almost full range of FIG. 7(c)) at the timing shown by dashed line. However, no such fast drop was observed on both vertical interferometers as shown in FIG. 7 (b) and (c). Instead, the FIR1 signal showed a slow reduction 78 µs after the reduction of MWI signal, as in the previous experiment. This time delay was comparable to the transit time from LFS to HFS along the field line at the top of the density pedestal, \( L_{\parallel}/C_s \approx 80 \mu s \). Therefore, this reduction was also attributed to the parallel plasma flow toward the lost density region at LFS.

All these observations demonstrate consistently the poloidal asymmetry of the collapse of the density pedestal, which is localized at the low-field side midplane. The estimation of poloidal extent of the collapse of density pedestal as well as its radial extent is quite important to understand the mechanism of the collapse of pedestal structure. Although the direct measurement of the poloidal extent is difficult using the current diagnostics in JT-60U, we can evaluate the upper limit of the poloidal extent of the collapse. The average reduction of \( \delta n_l \approx 9.1 \times 10^{17} \text{ m}^{-2} \) in FIR1 represents about 16% of the expected reduction of \( \delta n_l \approx 5.7 \times 10^{18} \text{ m}^{-2} \) obtained from the phase change of the reflectometer assuming poloidally uniform loss. Taking into account poloidal distribution of edge plasma volume enclosed between the flux surfaces, 16% of edge plasma volume at LFS midplane corresponds to the poloidal angle of the collapse of the density pedestal of about \( \pm 40 \) degree.

4. Heat and particle transport of ELM pulse

Expelled particles from LFS midplane flow toward the divertor target along the magnetic field line. Such ELM pulse passing through the SOL region was studied by using two Mach probes at midplane and x-point, and divertor heat flux was measured with IRTV camera, as shown in FIG. 9. This kind of probe experiment was performed with the following plasma parameters: \( I_p = 1.0-1.1 \text{ MA}, B_T = 2.0 \text{ T}, P_{NBI} = 4.3-4.5 \text{ MW}, \delta \approx 0.34, \kappa \approx 1.3 \) and \( q_{95} \approx 3.8 \). Figure 10 shows the time evolution of ion...
saturation currents at the upstream ($j_s^{\text{up}}$) and downstream ($j_s^{\text{down}}$) sides. At the onset of an ELM, abrupt increase of $j_s$ was observed on the midplane probe. On the other hand, $j_s$ in the x-point probe showed only gradual increase, with the peak value achieved 150 μs after the peak in the $j_s$ of the midplane probe as shown in FIG. 10. Since this value is of the order of $L_{\parallel}/C_\parallel \sim 100$ μs, where $L_{\parallel} \sim 26$ m and $C_\parallel \sim 2.5 \times 10^5$ m/s evaluating from the pedestal plasma temperature, this time delay can be explained by the parallel convective flow of the expelled particles along the field line.

The time evolution of the heat flux at both divertor targets, $j_s^{\text{up}}$ and stored energy is shown in FIG. 11. Duration of the peak heat flux corresponds to that of the enhancement of the plasma flow toward divertor. The fast drop of the stored energy by 19.3 kJ was also observed during the period of large heat flux deposition as shown by two dashed lines. The slow reduction of the stored energy in latter phase ($t > 5361$ ms) was attributed to the tail of the heat flux at the inner target as shown in FIG. 11(a). Summarizing these observations, the bulk of the heat load to the divertor targets may be carried by the parallel convective flow under these plasma conditions.

5. Summary

Simultaneous fast ELM measurements using reflectometer, interferometers, $D_\alpha$ intensity and magnetic probe reveal the detailed characteristics of type I ELMs. From the phase signal of the reflectometer, an ELM event can be divided into a precursor phase, collapse phase, recovery phase and a relaxation phase. The typical time scale of each phase is 200-500 μs, 100-350 μs, 200-500 μs and 6-10 ms, respectively. The radial movement of the cut-off layer was estimated to be about 7.5 cm inside the separatrix during the collapse phase. The
following three kinds of edge density measurements consistently indicate that the collapse of the density pedestal during type I ELMs in JT-60U is localized at the low-field side midplane:

1. The collapse of the pedestal density was manifested as fast inward movement of the cut-off layer at the low-field side and the increase of $D_{\alpha}$ intensity. At this time, however, no considerable reduction was observed on high-field side interferometer.

2. No density increase corresponding to the direct particle loss from the high-field side midplane to the scrape-off layer was observed in SOL/separatrix density measurements.

3. Horizontal interferometer clearly exhibited fast density loss simultaneously with the increase of $D_{\alpha}$ intensity. However, no immediate response was observed on both high- and low-field side vertical interferometers.

Expelled particles from the LFS midplane were measured using LFS midplane and X-point Mach probes together with IRTV. Observed time delay between midplane and X-point probes was comparable to the parallel convective time between two probes, and the duration of the peak heat flux corresponds to that of the enhancement of the plasma flow toward divertor target. Therefore, the bulk of the heat load to the divertor targets may be carried by the parallel convective flow.

Acknowledgement

The authors acknowledge the members of the Japan Atomic Energy Research Institute who have contributed to the JT-60U projects.

Reference

Fast Dynamics of Type I ELM and Transport of ELM Pulse in JT-60U


Naka Fusion Research Establishment, JAERI

Introduction

The heat and particle expulsion caused by the Edge Localized Mode (ELM) are one of the most important issues in ITER and DEMO.

The structure of the collapse should determine the ELM size and the transport process of the expelled particles should determine its impact to the divertor targets.

Therefore, the measurement of fast ELM dynamics such as:

- The evolution of the collapse of density pedestal
- Asymmetry and symmetry of the collapse
- ELM transport

are quite important.

Outline

- Introduction
- Fast dynamics of the collapse of density pedestal
- Asymmetry of the collapse of the density pedestal
- ELM transport study using SOL probe and IRTV
- Summary
Four different phases of the collapse of density pedestal

- Pre-collaps phase: 0-20 ms
- Coherent density fluctuation
- Inward movement indicating collapse
- Collapse phase: 20-90 ms
- Moved outward beyond the initial position (overshoot)
- Gradually returned to the initial position

Asymmetry of the collapse of density pedestal

- Half of the FIR1 chord passed through the top of density pedestal (r/a=0.8)
- Cutoff layer moved 11 cm inside the separatrix (twice the pedestal width)
- No immediate reduction indicating the collapse at HFS plasma
No density increase corresponding to direct particle loss at HFS

Such a small density increase of $0.8 \times 10^{19}$ m$^{-3}$ can be detected by FIR1 in this plasma configuration. If the type I ELM was poloidally uniform event, FIR1 signal would increase before the increase of $\Delta n$ signal.

Parallel convective flow may carry main heat load to the divertor

Observed delay (~160μs) can be explained by the transit time of convective ion flow. ($l_i/C_i \sim 100μs$) Heat flux deposition time is comparable to time scale of enhanced plasma flow.

Horizontal interferometer showed fast density loss at the onset of an ELM

Fast simultaneous measurements of type I ELM in JT-60U reveal the detailed dynamics of the collapse of density pedestal.

- We found four different phases of the collapse of density pedestal, and
- Cutoff layer moved inward up to 30% of the minor radius.
- Three kinds of edge density measurements consistently indicated
- From the observation of ELM pulse using SOL probes and IRTV camera, main heat load seems to be carried by the parallel convective flow.
1.8 Heating and Current Drive by Electron Cyclotron Waves in JT-60U

T. Suzuki 1), S. Ide 1), C. C. Petty 2), Y. Ikeda 1), K. Kajiwara 1), A. Isayama 1), K. Hamamatsu 1), O. Naito 1), M. Seki 1), S. Moriyama 1) and the JT-60 Team 1)

1) Japan Atomic Energy Research Institute, Naka-machi, Naka-gun, Ibaraki 311-0193, Japan 2) General Atomics, San Diego, California, US

e-mail contact of main author: suzukit@fusion.naka.jaeri.go.jp

Abstract. Results of studies on heating and current drive by the electron cyclotron (EC) waves in JT-60U are presented. Electron temperature up to 26keV was achieved by injecting EC waves in the center of a reversed shear plasma produced by the lower hybrid (LH) waves. The electron temperature $T_e$ exceeds 24keV in a wide range of minor radius ($\rho<0.3$, where $\rho$ is the normalized minor radius). ECCD (Current Drive) efficiency $\eta_{cd}$ was examined at high $T_e$ up to 21keV without using LH waves. The CD efficiency increases with $T_e$ but there is discrepancy from linear dependence on $T_e$ in a high $T_e$ regime of 10-20keV. Dependence of normalized CD efficiency $\xi=\eta_{cd}/e^2kT_e$ on deposition location was also studied to optimize the CD efficiency, since trapped particle effect, which depends strongly on deposition location, is expected to reduce $\xi$. The effect was detected from significant decrease in $\xi$ in the lower magnetic field deposition, which is consistent with linearized Fokker-Planck calculation.

1. Introduction

Electron cyclotron (EC) waves are considered as a strong tool to control electron heating and current profile in plasmas. Since EC waves are absorbed by electron cyclotron resonance, it is considered that absorption location of waves can be easily determined by calculation, so that heating and current drive only close to the resonance is possible. Actually, EC driven current profiles have been measured in DIII-D [1] and in JT-60U [2,3], and it was shown that the measured EC driven current profiles are spatially localized and that they agree with calculations at least in moderate conditions. Owing to such advantages, EC systems are planned to be installed in ITER (International Thermonuclear Experimental Reactor). Main purposes of EC system in ITER were described as follows [4,5]. 1) Steady state current drive capability. 2) Assistance of plasma control, especially to stabilize neo-classical tearing mode (NTM). 3) Wall conditioning. 4) Assistance of the poloidal field system in breakdown and current initiation phase. 5) Heating source to access H-mode.

In this paper, we treat the first and the second ones. The former concerns to on-axis current drive by EC waves (ECCD) in high electron temperature plasmas. Existing data are limited up to $T_e$ ~7keV in T-10 [6] and in JT-60U [2], while ECCD is expected in volume averaged $T_e$ under 12keV in ITER. Electron temperature at CD location in ITER should be much higher than 12keV in on-axis ECCD case. In such high $T_e$ regime, apart from linear dependence of current drive efficiency on $T_e$ is expected [5,7]. The second one concerns off-axis ECCD, since NTMs usually occur around $m/n=2/1$ or 3/2 (poloidal / toroidal mode numbers) surfaces. NTMs on such a major rational surface usually accompany larger islands and affect the plasma confinement. The major rational surface exists in off-axis, where it is considered that trapped particles affect ECCD efficiency, if low-q operation is employed aiming economically preferable high $\beta$ operation. The effect should be studied for applicable use of ECCD for NTM suppression [8].

Section 2 describes recent progress of EC system in JT-60U and production of high $T_e$ plasma using EC waves. On-axis ECCD in such high $T_e$ plasma is analyzed in Section 3. CD efficiency is investigated as a function of electron temperature close to the ITER operation regime. Section 4 describes off-axis ECCD in relation to the trapped particle effect. The effect
is experimentally investigated by changing the deposition location. Dependence of CD efficiencies on electron density and electron temperature is also described. The dependence can help to find optimized plasma parameters for ECCD. EC driven current profile in a large minor radius is also studied, where trapped particle fraction is close to that in ITER. Summary is in the section 5.

2. Progress of EC System in JT-60U and Production of High Electron Temperature plasma

JT-60U has four units of EC systems, each of which has one gyrotron with frequency of 110GHz [9,10]. One unit of them was installed in 2000. EC waves are injected into plasma from lower field side of torus (upper outboard) usually as an O-mode. EC waves are absorbed in fundamental resonance for about 4T of toroidal field. These conditions are same as ITER, except the wave frequency and the toroidal field. Toroidal injection angle of an antenna, which three units of gyrotrons use, is fixed with an angle so that the refractive index parallel to magnetic field \(N_r\) is about 0.5 at the plasma center. The angle is about 60° to the direction of toroidal field at the plasma center. Another antenna for the newly installed unit has a capability to control toroidal injection angle for co-/ctr-ECCD or for just heating. Both antennas can change injection angle in poloidal cross-section to control deposition location. EC system in JT-60U has extended the injection power \(P_{in}\) into the plasma up to 3MW for 2.7s. The pulse duration reached 5s at 1.5MW. Energy injected into the plasma reached 10MJ (2.8MW for 3.6s); see Fig. 1. Output power per a gyrotron was about 1MW, and transmission efficiency was 70-80%. The output power per a gyrotron and transmission efficiency were close to that required for EC system in ITER, although ITER is designed to use more gyrotrons. Not only the additional fourth unit of EC system but also the increases of the output power and of the transmission efficiency contribute the increase of \(P_{in}\).

The progress in the input power made it possible to produce plasmas with high \(T_e\) up to 26keV measured by the ECE diagnostics [11]. Error in ECE measurement was typically \(+2\)keV in this discharge. Temporal evolution of the electron temperature at the plasma center \(T_e(0)\) is shown in Fig. 2 along with the heating power by EC (dashed) and LH (dotted) waves.

FIG. 1. Progress in the EC power injected into plasma and in its pulse duration in JT-60U. Closed circles and stars represent achieved results in 2001 and in 2002, respectively.

FIG. 2. Temporal evolutions of plasma current (I_p), injected power (P_in) (a) and the electron temperature at the plasma center (b). (c) \(T_e\) profile against the normalized minor radius. Open and closed symbols denote \(T_e\) before and during the EC, respectively. Circles and squares represent \(T_e\), measured by ECE and Thomson scattering by YAG LASER, respectively.
The profiles of the electron temperature before and during the EC heating are also shown in Fig. 2. Preheating by the LH waves [12] up to 1.9MW raised\(T_e(0)\) to 12keV. LHCD was applied during the plasma current\(\langle I_p\rangle\) ramp-up in order to produce a negative shear region in the center of the plasma [13,14]. The negative shear within\(\rho<0.4\) is confirmed by the motional Stark effect (MSE) diagnostics [11] even at\(t=8.2s\). When the EC waves of 2.9MW were put in the center of the plasma, electron temperature in the negative shear region exceeded 24keV in a wide range of minor radius\((\rho<0.3)\), and was 26keV at the maximum.

3. On-axis ECCD in a Wide Range of Electron Temperature

Simultaneous current drive by EC waves and by LH waves can produce error in the determination of EC driven current due to the error in LH driven current [15] and synergetic effect with LH and EC waves. Therefore, we prefer discharges without LH waves for on-axis EC driven current measurement. Waveforms of a discharge are shown in Fig. 3(a). EC waves are put at the flat top phase of the plasma current of 0.6MA. Neutral beams (NBs) of about 1.7MW are for MSE diagnostics. Two units of NBs are put in balanced to cancel beam driven current. Central electron temperature measured by ECE increases up to 23keV, by injecting 2.9MW of EC waves. Line averaged electron density is nearly constant during the EC injection. Slight increase of the\(n_e\) after the EC injection is mainly due to the increased fueling by diagnostic NB for MSE as shown in Fig. 3(a). Loop voltage dropped to about 0.1V. We had stable plasma with such high\(T_e\) for about 0.8s until crash in\(T_e\) occurred at\(t=5.96s\). In this series of experiment, duration without instability shortens when strong on-axis ECCD is applied. Therefore in this discharge, one unit of EC (0.6MW) is used for just heating, and three units of ECs (2.3MW) are used (and fixed) for co-ECCD. Measurement of EC driven current is made from\(t=5.5s\) to 5.9s, as hatched in Fig. 3(a). Electron temperature profile during the analysis is shown in Fig. 3(b). Although the central electron temperature is measured by ECE, we confirmed that Thomson scattered spectra of ruby LASER extremely broadened so that the electron temperature near the center is probably no less than 20keV.

Non-inductive current was evaluated by the loop voltage profile analysis [2,16] using the MSE in a high electron temperature plasma of\(T_e(0)=23\)keV with EC heating. The EC driven current profile is shown in Fig. 4(a), in comparison with a result of linearized Fokker-Planck calculation. The measured EC driven current profile is spatially localized even in the high\(T_e\).

![Fig. 3](image)

**Fig. 3.** (a) Waveforms of a discharge; plasma current\(\langle I_p\rangle\), loop voltage\(\langle V_l\rangle\), injected EC power\(\langle P_{EC}\rangle\), NB heating power\(\langle P_{NB}\rangle\), electron temperature at plasma center\(T_e(0)\), line averaged electron density\(n_e\). EC driven current is evaluated during the hatched region\((t=5.5-5.9s)\), where\(T_e(0)\) is nearly constant. (b) Electron temperature profile during ECCD analysis at the\(t=5.7s\). Electron temperature is measured by ECE (circles) and Thomson scattering (squares with errors) diagnostics. (c) Deposition locations of EC waves for current drive (LFS) and heating (HFS). Magnetic axis is between the two locations.
The measured EC driven current $I_{EC}$ was $0.74 \pm 0.06 \text{MA}$, where $T_e$ at the CD location was $21 \text{keV}$. The CD location means a magnetic surface where enclosed EC driven current is half of total EC driven current. Linearized Fokker-Planck code predicts $I_{EC}=1.1 \text{MA}$. The calculation includes a trapped particle effect by toroidicity, but does not include an electric field effect by Ohmic field. The measurement was made in a transient phase ($0.4-0.8 \text{s}$ after the EC injection) during the inductive current diffusion. The resistive diffusion time for the width of the experimental EC driven current profile (standard deviation of Gaussian fit: $0.16 \text{m}$) is about $22 \text{s}$ for $T_e=21 \text{keV}$. Therefore, most of the EC driven current is canceled by the inductive electric field. When the EC driven current modifies the total current profile due to the diffusion of the inductive field, minimum of the safety factor approaches $2.5$ ($0.86 \text{s}$ after the EC injection). Since some instability prevents evolution of the total current profile, apparent EC driven current ($0.74 \text{MA}$) does not exceed plasma current ($0.6 \text{MA}$); loop voltage of the plasma does not get negative as seen in Fig. 3(a). The instability is seen in the sudden decrease in $T_e$ in Fig. 3(a) at $t=5.96 \text{s}$.

Figure 5 shows the measured current drive efficiency $\eta_{CD}=I_{EC}R_n/P_{abs}$ as a function of the electron temperature at CD location. In the above definition, $R_n$ and $P_{abs}$ are the major radius of plasma and absorbed power of EC waves used for current drive, respectively. The absorption power is defined as input power multiplied by absorption fraction of EC waves calculated by linearized Fokker-Planck calculation. Absorption fractions in Fig. 5 were more than $95\%$. All of the CD locations of data in Fig. 5 are $\rho<0.17$. Calculated CD efficiency for the experimental condition is also plotted in the Fig. 5. Measured and calculated $\eta_{CD}$ are found to increase with local $T_e$ at CD location. The range in $T_e$ covers considerable part of the ITER operation regime ($12 \text{keV}$ in volume averaged $T_e$). The highest CD efficiency was $0.42 \times 10^{19} \text{A/Wm}^2$. The experimental CD efficiency was smaller than that of calculation. There are candidates to produce the difference. One of them is the negative electric field that is produced by induction of EC driven current. Negative electric field in plasma is expected to reduce EC driven current and the efficiency [17]. Since the linearized Fokker-Planck calculation do not include the effect, the calculation can be smaller in the experimental condition with negative electric field. In other words, ECCD in a fully steady state, where no electric field remains, will show higher experimental CD efficiency. Such an ECCD experiment will be a demonstration of ECCD in ITER. Since the evolution of the plasma current is limited by an instability as described before, it is important to avoid the instability.

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**FIG. 4.** (a) EC driven current profile of experimental measurement (exp.) and that of linearized Fokker-Planck calculation (calc.). Spatially integrated EC driven currents are also shown for both cases. (b) Safety factor profiles at the start ($t=5.55 \text{s}$) and at the end ($t=5.98 \text{s}$) of the loop voltage profile analysis. Since EC driven current in (a) increases the total current density in $\rho<0.4$, safety factor decreases around the magnetic axis. When $q_{min}$ reaches to $2.5$, some instability occurs.

**FIG. 5.** Measured ECCD efficiency (circle) in JT-60U increases with the electron temperature at CD location. Linearized Fokker-Planck calculations (square) also show the same tendency, while the measurements are smaller than the calculations.
to reduce the EC driven current. Combination of smaller ECCD and larger ECH will provide such a condition. Others are as follows. If distribution function of electron is strongly distorted by the perpendicular heating by EC waves, parallel temperature could be lower than the perpendicular one. Slower parallel velocity shifts the deposition location of EC waves, and hence CD location, more close to the magnetic axis, since ECCD location is LFS of plasma (Fig. 3(c)). Another candidate is an enhanced radial transport of wave coupled electrons [18]. The last one may not the case, since the characteristic scale width of the experimental EC driven current profile does not show broadening. In comparison between the experiment and the calculation, we should also consider non-linearity effect that is off course not included in the linear calculation. To achieve a high Te by ECH, electron density was small and absorption power density was large in the on-axis ECCD. A parameter $p_{abs}/n_{e19}$ exceeds a criterion of 0.5 under the experimental condition, which requires non-linear treatment of Fokker-Planck equation [19]. Notation $p_{abs}$ and $n_{e19}$ are absorbed power density in MW/m$^3$ and electron density in 10$^{19}$m$^{-3}$, respectively. Comparison between the experiment and calculation should be investigated in future, considering the electric field effect and the non-linearity effect.

4. Off-axis ECCD for Trapped Particle Effect Study

Properties of the EC driven current are studied in detail for a higher CD efficiency. Emphases are mainly put on the trapped particle effect, which is expected to reduce the normalized EC driven current efficiency $\zeta = \frac{\epsilon}{\eta_{e19}} \frac{e}{kT_e^*}$. Principal dependencies of CD efficiency on plasma parameters are removed in $\zeta$. Since the reduction of $\zeta$ affects the required EC power for the current profile control, the effect should be investigated. It is expected that the normalized CD efficiency is different between higher field side (HFS) deposition (smaller trapped particle effect) and lower field side (LFS) deposition (larger effect). Since the fraction of the trapped particle is considered to be a function of inverse aspect ratio $\epsilon$, dependence of $\zeta$ on minor radius is also expected. Square root of $\epsilon$ can be a measure of the trapped particle fraction in LFS. Two curves in Fig. 6 show the expected $\zeta$ by the linearized Fokker-Planck calculations including the trapped particle effect. We plotted converted $\zeta$ equivalent to $Z_{eff}$ of unity, assuming a conventional weak $Z_{eff}$ dependence like $\zeta \propto 1/(5 + Z_{eff})$.

The measured effective charge in this experiment was between 1.5 and 1.9 so that the correction was small. The lower curve represents the LFS deposition, showing reduction in $\zeta$ with minor radius. The upper curve is for the HFS deposition, where no significant decrease in $\zeta$. Circles (closed/open) denote measured $\zeta$ in HFS/LFS deposition respectively. They seem to agree with the calculated values. Significant decrease (by a half) in $\zeta$ is seen in the case of LFS deposition compared to that of HFS deposition at $\rho = 0.35$. The reduction in $\zeta$ may show the trapped particle effect. Dependence of $\zeta$ on minor radius (or $\epsilon$) was not clear, since variation of $\epsilon$ was not enough compared to the error in $\zeta$.
We investigated the property of ECCD depending on plasma parameters, such as \( n_e \) and \( T_e \). Since these parameters could affect the absorption of EC waves, we can expect some effect that could not be normalized by the general formulation of CD efficiency \( \zeta \). That is to say, parameter dependencies specific to physics of ECCD. To ensure that the trapped particle effect are same, we had several ECCD discharges with same toroidal field and injection angle of EC waves, but with different \( T_e \) and \( n_e \). The input power of EC waves was increased to keep \( T_e \) constant at higher \( n_e \) (Fig. 7(a)), or electron temperature was changed by input power of EC waves under same \( n_e \) (Fig. 7(b)). We can see experimental \( \zeta \) increases with \( n_e \) for plasmas with nearly same \( T_e \); see Fig. 7(a). The dependence of \( \zeta \) on \( T_e \) was not clear since the variation of \( T_e \) was not enough. The dependence on \( n_e \) is also seen in the linearized Fokker-Planck calculation, but the experimental dependencies are stronger than those of calculations are.

The clear \( n_e \) dependence in the calculation should be explained under the linear theory. Fig. 8(a) shows damping of EC power as a function of major radius of ray trajectory. EC waves propagate from LFS (larger R) to HFS (smaller R) and are absorbed outside of cyclotron resonance due to Doppler shift. Flat \( T_e \) and \( n_e \) profiles are employed to ignore profile effects. CD location in the calculation is in HFS to ignore the trapped particle effect. The electron temperature is set to 5keV, and \( n_e \) is varied by 0.5, 1.0, and 2.0x10^{19}/m^3. The point where the EC power is damped by a half of total absorption power is considered to be a representative damping point. The point moves toward LFS, when \( n_e \) increases. This is because increased deposition of EC power to electrons with faster \( v_\parallel \) component, considering Doppler shift. Normalized ECCD efficiency is written by \( \zeta \cdot Z_{eff} = 3u^2 \), when \( u = (\omega - \omega_{oe})/k_\parallel v_{te} \) is much larger than unity under Lorentz approximation (\( Z_{eff} \gg 1 \)) and without trapped particle effect [20]. Notations \( \omega \), \( \omega_{oe} \), \( k_\parallel \), \( v_{te} \), and \( Z_{eff} \) are angular frequency of wave, electron cyclotron angular frequency, wave number parallel to magnetic field, thermal velocity of electron, and effective charge, respectively. This analytical solution is compared to the calculation result (Fig. 8(b)), by using the \( \omega_{oe} \) at point where the power is damped by a half of total absorption power. Results of linearized Fokker-Planck calculation agree well with the analytical solution. Therefore, the \( n_e \) dependence in calculation is considered to be due to the increase in wave coupling to faster \( v_\parallel \) electrons. When the absorption of EC waves moves faster \( v_\parallel \) in the velocity space, the absorption location apart from the trapping boundary by the toroidal effect. The \( n_e \) dependence in Fig. 7(a) comes from both of the effects. The stronger

![FIG. 7. Normalized CD efficiency by measurement (circle) and by calculation (triangle) as a function of (a) electron density, and of (b) electron temperature. Positive dependence of \( \zeta \) on electron density is seen in (a), while dependence on \( T_e \) (b) is weaker than that on \( n_e \).](image_url)
FIG. 8. Fraction of the EC power damped in the plasma, as a function of major radius. Location where EC waves damps by a half of total absorption power moves LFS of torus, when electron density increases. The shift is due to increased coupling of EC waves to electrons with larger $v_e$. Increase in $\xi$ is consistent with the analytical solution under Lorentz approximation, which increases with $v_e$ of coupled electrons. In calculation, effective charge was set to 1000 for Lorentz approximation.

dependence of experiment than the linear theory is actually not known, so that we need further investigation of the reason. Both of the electric field effect and non-linear effect are not important in these experiments.

Further off-axis ECCD is also investigated, where the trapped particle effect is expected to be much stronger than $\rho=0.35$ in Fig. 6(b). Because the electron density is low ($n_e=0.6\times10^{19}\text{m}^{-3}$) to enlarge the EC driven current density, we cannot compare the CD efficiency in the context of Fig. 6(b). We did not expect the previously described strong $n_e$ dependence at the experiment. Measured EC driven current profiles are compared with calculations in Fig. 9. Figure 9(a) is for HFS deposition. Fig. 9(b) shows LFS deposition case, where the larger trapped particle effect is expected. Large errors in the experimental EC driven current show limitation of detecting a small non-inductive current by the loop voltage profile analysis in JT-60U. Measurement and calculation fairly agree well in CD location for LFS and HFS deposition. Again, EC driven current was smaller in LFS deposition than in HFS deposition, which can be the evidence of the trapped particle effect. Residual current density near the edge ($\rho>0.7$) exists even in a phase without ECCD [2]. The residual current is considered to be due to errors in calibration of absolute angle of MSE diagnostics. Under the configuration of these discharges, normalized minor radius of 0.6 corresponds to trapped particle fraction of $\epsilon^{0.6}=0.4$, which is same to that of $\rho=0.5$ in ITER. ECCD in such a minor radius is important for applicable use on NTM suppression. Such further off-axis ECCD in LFS was

FIG. 9. EC driven current density profiles of measurement and of linearized Fokker-Planck calculation. (a) HFS deposition. (b) LFS deposition.
demonstrated.

5. Summary

Recent progress of EC system in JT-60U enabled production of a high electron temperature plasma, which is close to the ITER operation regime. The electron temperature at the plasma center reached 23keV with 2.9MW of EC waves. Spatially localized EC driven current profile was measured, which do not show significant radial diffusion of the driven current. Measurement of EC driven current at local $T_e$ of 21keV showed that EC driven current is about 0.74MA, which is smaller than that of linearized Fokker-Planck calculation (1.1MA). The CD efficiencies in both experiment and calculation increase with $T_e$. Since the calculation does not include toroidal electric field effect and non-linearity, comparison of the measurement with calculation should be investigated in future considering both of them. Normalized CD efficiency $\zeta$ at $\rho=0.35$ of LFS deposition was about a half of that of HFS deposition, which is consistent with the calculation. The reduction of $\zeta$ can be an evidence of the trapped particle effect. Further off-axis ECCD in the same $n_e$ to extend $\varepsilon^{SS}$ will clarify the trapped particle effect. It was found that the normalized CD efficiency increases with the electron density. A part of the $n_e$ dependence can be explained by a coupling of EC waves to a faster parallel velocity component of electrons. The stronger $n_e$ dependence in experiment than in the calculation should be investigated. Further off-axis ECCD near $\varepsilon^{SS} = 0.4$ was demonstrated to show that the ECCD is effective even under such large trapped particle fraction.

References

Heating and Current Drive by Electron Cyclotron Waves in JT-60U

Japan Atomic Energy Research Institute (JAERI)
C. C. Petty
General Atomics (GA)

Contents of this talk

☆ EC system in JT-60U and its recent progress
☆ CD efficiency
  • On-axis ECCD in the high-\(T_e\) plasmas
  • Off-axis ECCD for trapped particle effect
  • Off-axis ECCD dependence on \(n_e\)
☆ Summary

Introduction

Experimental research on current drive by electron cyclotron waves (ECCD)
On-axis ECCD efficiency in a wide range of \(T_e\)
• ECCD: Drives central current in ITER, where volume averaged \(T_e\) is 12keV.
  ECCD data in high-\(T_e\) regime is necessary.
  \(T_e\)~7keV on T-10 and on JT-60U

Off-axis ECCD in relation to trapped particle effect
• Off-axis ECCD is important for the current profile control to
  suppress neo-classical tearing mode (NTM) in high \(B_e\) plasma.
• Toroidicity induced trapped particles affect the CD efficiency.

Progress of EC system in JT-60U enabled above experiments.

Recent progress of EC system in JT-60U

- 4 units of 1MW 110GHz gyrotrons
- Fundamental O-mode launched from the lower magnetic field side of torus
- 3 units fixed for co-ECCD
- 1 unit steerable for co-/ctr-ECCD or ECH
  \(N/H\) ~ 0.5 at plasma center for CD
EC driven current is experimentally evaluated.

Calc. agrees with the exp. result. Linearized Fokker-Planck calc.
- Including trapped particle effect which is negligible for small \( \kappa \) and \( R \)
- Not including toroidal electric field effect which is negligible for small \( E_y/E_{Dr} \)
- Not including non-linearity which is negligible for small power density

EC driven current is measured at \( T_e=21 \text{keV} \).

- On-axis EC driven current profile is spatially localized, which shows no significant radial diffusion of current.
- Measured EC driven current is 0.74MA.
- \( \eta_{CD} = 4.2 \times 10^{-18} \text{A/m}^2 \)
- \( \eta_{CD} = \frac{i_{EC}}{P_{abs}} \)
- EC driven current is mostly canceled by strong negative OH electric field.
  In an early phase (0.4-0.8s after EC inj.) of current evolution.
  \( \tau_R = 25 \text{s} \) for width of \( i_{EC} \) profile

\( T_e(0) \sim 23 \text{keV} \) was achieved by EC waves.

Objective is to analyze EC driven current in a high-\( T_e \) regime.
Resonance location of ECH/ECCD was optimized to achieve high-\( T_e \) for long enough for ECCD analysis

\( T_e(0) = 23 \text{keV for 0.8s} \)
limited by an instability, which occurred at \( q_{min}=2.5 \).

Measured ECCD efficiency increased with \( T_e \).

- Measured \( \eta_{CD} \) increased with \( T_e \).
- The result should be compared with cal.
  including \( E_y/E_D \) and non-linear effects.

Linear calc. without \( E_0 \)
\( 6.3 \times 10^{-18} \text{A/m}^2 \) for \( T_e=21 \text{keV} \).

The negative electric field can be a candidate for the reduction of \( \eta_{CD} \).

For further study of on-axis ECCD in steady state.
ECCD experiment with high-\( T_e \) under small \( B_{parallel} \),
for power density.
An example is combination of strong ECH and weak ECCD,
for smaller EC driven current.
Significant decrease in $\zeta$ at LFS than at HFS.

Normalized CD efficiency $\zeta$ is compared between HFS and LFS deposition to investigate the trapped particle effect.

$B_l$ and injection angle are changed.
Both $E_\parallel$ effect and non-linear effect are weak.

\[ \zeta = \frac{a_2 \eta_{\text{CD}}}{T_e} = \frac{a_2 \eta_{\text{EC}} n_e R_p}{T_e} \]

- Normalized CD efficiency at LFS deposition is about a half of HFS deposition.
- Dependence of $\zeta$ on $\epsilon^{0.5}$ was not clear in this experiment, while calc. predicts that $\zeta$ decreases with $\epsilon^{0.5}$.
- Extended variation of $\epsilon^{0.5}$ is required.

Summary

- Injection of EC waves of 3MW/2.7s, or for 5s/1.5MW was achieved. Total injection energy reached up to 10MJ.
- High $T_e$ (0)-23keV for 0.8s was achieved by combination of ECH and ECCD.
- On-axis ECCD efficiency $\eta_{\text{CD}}$ increased with $T_e$ and reached $4.2 \times 10^{16} \text{A/W/m}^2$ at $T_e=21\text{keV}$.
- Normalized ECCD efficiency $\zeta$ of LFS deposition is about a half of that of HFS deposition.
- Normalized CD efficiency increased with $n_e$.
  - Density dependence in exp. was stronger than that in calc.

Future issues

- high $T_e$ ECCD experiment with smaller negative $E_\parallel$
- further off-axis ECCD to extend $\epsilon^{0.5}$
- investigation of $n_e$ dependence of $\zeta$
1.9

Property of Alfvén Eigenmode in JT-60U Reversed Shear and Weak Shear Discharges

M. Takechi 1), A. Fukuyama 2), K. Shinohara 1), M. Ishikawa 3), Y. Kusama 1), S. Takeji 1),
T. Fujita 1), T. Oikawa 1), T. Suzuki 1), N. Oyama 1), T. Ozeki 1), A. Morikawa 1), C. Z.
Cheng 4), N. N. Gorelenkov 4), G. J. Kramer 4), R. Nazikian 4) and the JT-60 team 1)

1) Japan Atomic Energy Research Institute, Naka-machi 319-0193, Japan.
2) Department of Nuclear Engineering, Kyoto University, Kyoto 606-8501, Japan.
3) Plasma Research Center, University of Tsukuba, Tsukuba-shi 305-8877, Japan.
4) Princeton Plasma Physics Laboratory, Princeton, NJ 08543, USA

e-mail contact of main author: takechim@fusion.naka.jaeri.go.jp

Abstract. This paper reports activity of Alfvén eigenmode (AE) in the JT-60U reversed shear (RS) and weak shear (WS) plasmas. The AEs with a rapid frequency sweeping and then saturation of frequency as \( q_m \) decreases has been observed in low-\( \beta \), RS discharges with Negative-ion-based NBI (NNBI) or ICRH. We introduce the new type of AE which we call reversed-shear-induced Alfvén Eigenmode (RSAE) near \( q_m \). This puzzling frequency change can be explained by considering the properties of RSAE and their transition to toroidal Alfvén Eigenmodes (TAEs). We verify the existence of RSAEs and their transition to TAEs from magnetic fluctuations and measured q-profile in JT-60U plasmas. The AE amplitude is maximum during this transition, e.g., \( 2.4 < q_m < 2.7 \) for the \( n = 1 \) mode. The recently installed diagnostic of a neutron emission profile measurement reveals the transport of energetic ions associated with MHD modes. Actually, the fast ion loss was observed by neutron profile measurement, when the \( n = 1 \) mode amplitude is large at this transition. It is preferable to operate outside the \( q_m \) transition range of \( n = 1 \) AE to avoid substantial fast ion loss by large amplitude AEs in RS plasmas.

1. Introduction

Magnetohydrodynamics (MHD) instabilities driven by energetic particles such as the Toroidal Alfvén Eigenmode (TAE) [1] and the fishbone [2] have been extensively studied experimentally and theoretically, especially in the positive shear (PS) plasmas. These instabilities can expel energetic particles before they are thermalized with thermal plasmas, possibly leading to the quenching of fusion ignition and damage of the first wall [3]. A reversed shear (RS) plasma is potentially an efficient operation mode for steady state tokamak reactors with good confinement and a large bootstrap current fraction. The stability of TAEs in RS plasmas was first calculated by using the NOVA-K code [4] and the results indicated that TAEs in the RS configuration are more stable than those in the PS configuration [5, 6] and are more stable in the RS plasma with internal transport barrier (ITB) than that without ITB [7] because of the limited gap alignment in the shear Alfvén continuum spectrum. The latter was confirmed by experimental results of AEs driven by energetic particles from ion cyclotron resonant frequency (ICRF) heating in the JT-60U plasmas [7]. The behavior of AEs in RS plasmas has not been systematically studied as those in PS plasmas. Therefore characteristics of AE in RS plasmas is not well-known, e.g., in these ICRF RS plasmas, a puzzling Alfvén Eigenmode (AE) has been observed with large and rapid upward chirping in frequency, which cannot be explained by the change in frequencies of TAE type modes. In this paper we report the AE activity in the RS plasmas in JT-60U negative-ion-based neutral beam injection (NNBI) experiments and the effects of AEs on fast ion loss. We found that large and rapid frequency (both upward and downward) chirping modes exist, and these modes evolve into TAE modes as minimum of \( q \) \( (q_m) \) decreases. We provide a
theory of the reversed-shear-induced Alfvén Eigenmode (RSAE) model to interpret the observed fast frequency chirping AEs. The RSAE model is also employed to explain the previously observed fast upward frequency chirping AEs observed in JT-60U ICRF RS plasma experiments. Recently, we installed six-channel neutron profile monitor to investigate the fast ion behavior. As the AEs evolve from RSAEs to TAEs, the AE amplitude is enhanced and significant fast ion loss is observed by this neutron profile measurement.

2. Property of the AE in a RS plasma

To understand the large and rapid frequency sweeping and its subsequent saturation, we propose a model of RSAE, which is the global AE near the zero shear region of the RS plasma, and its transition to TAE as \( q_{\text{min}} \) decreases. We first note that the Alfvén continuous spectrums in RS plasmas are different from that in PS plasmas. Let us consider the \( n=1 \) AE in the range of \( 2 < q_{\text{min}} < 3 \) as an example. Figure 1 shows two shear Alfvén continuous spectra, frequency vs. normalized plasma radius, in RS plasmas with (a) \( q_{\text{min}} = 2.8 \) and (b) \( q_{\text{min}} = 2.3 \) as examples for \( 2.5 < q_{\text{min}} < 3 \) and \( 2 < q_{\text{min}} < 2.5 \), respectively. In both cases, \( n = 1, q(0) = 4 \) at the plasma center, and \( q(a) = 5 \) at the plasma edge. Frequency gaps outside the region of minimum safety factor are similar in both figures. However, the gap structures around the \( q_{\text{min}} \) region are entirely different. The two gaps shown in Fig. 1(b) are two TAE gaps formed by toroidal coupling of the \( m = 3 \) and \( m = 2 \) harmonics around two \( q = 2.5 \) locations. On the other hand, in Fig. 1(a), there is no \( q = 2.5 \) and the continuum gap is formed due to the reversed shear \( q \)-profile at the zero magnetic shear location and is not induced by toroidal coupling. The lower continuum is due to the \( m = 3 \) harmonic and the upper continuum is due to \( m = 2 \). Around the upper and the lower boundary of this gap, we have found AEs with discrete eigenvalues both theoretically [9] and experimentally [10]. These modes are called Global Alfvén Eigenmode (GAE) in theses papers of [9] and [10] because the resonance condition for this mode is the same as for GAE [11]. It has been generally thought that GAE except for \( n = 0 \) GAE is destabilized only in the plasmas with no magnetic shear like cylindrical configuration or low shear stellarator such as Wenderstein-7AS (W7-AS) [12]. Consequently TAE has been mainly remarked and investigated in many tokamaks. The damping rate caused by bulk plasmas and eigenfrequency of AEs in RS plasmas were calculated with the full wave code TASK/WM [9]. An existence of GAE was predicted and it

![Fig. 1](image1.png)  
Fig. 1 Schematic drawing of the \( n = 1 \) spectrum of AEs in the RS plasma with (a) \( q_{\text{min}} = 2.8 \) and (b) \( q_{\text{min}} = 2.3 \). In both cases, \( n = 1, q(0) = 4 \) at the plasma center, and \( q(a) = 5 \) at the plasma edge.

![Fig. 2](image2.png)  
Fig. 2 The frequencies of RSAE and TAE as a function of \( q_{\text{min}} \). The bold lines denote the frequencies of the \( n = 1 \) AEs and thin lines are for the \( n = 2 \) AEs. Numbers in the parentheses are poloidal mode numbers.
transits to TAE as $q_{mn}$ changes. From calculation of eigenfrequency GAE frequency changes as $q_{mn}$ changes. The results of the calculating damping rate indicated that TAE is more stabilized rather than GAE by bulk plasmas and is stabilized least in the transition from GAE to TAE. On the other hand, in the compact helical system (CHS) heliotron/torsatron plasmas, GAE and that TAE changed to GAE as safety factor changed were observed [10]. In contrast to W7-AS, CHS has a moderate magnetic shear comparable to that in a tokamak. Therefore TAE rather than GAE had been predicted to be destabilized in a CHS plasma and actually TAE was observed and identified. However, in CHS plasma the AEs of which frequency change cannot be explained by that of TAE were observed. However, this frequency change of AE was explained by using that of GAE. From the TASK/WM code calculation, these AE eigenfunctions are rather localized around the $q_{mn}$ location. Because GAEs original defined [11] have very different mode structure, we call these modes as RSAEs, instead of GAEs called previously.

The large and rapid frequency sweeping of AEs can be explained by RSAE and the subsequent frequency saturation by the evolution from RSAE to TAE. The AE frequencies are estimated as follows:

1. As $q_{mn}$ decreases in the range of $(m+1/2)/n + c < q_{mn} < (m+1)/n$ there are two RSAEs: the frequency of the high frequency RSAE (HRSAE) decreases as

$$f_{\text{HRSAE}} \sim \left( n-m/q_{mn} \right) v_A/2\pi R,$$

and the frequency of the low frequency RSAE (LRSAE) increases as

$$f_{\text{LRSAE}} \sim \frac{(m+1)/q_{mn} - n}{n} v_A/2\pi R,$$

2. For $m/n < q_{mn} < (m+1/2)/n + c$, TAE gaps form and TAE frequency is approximately given by

$$f_{\text{TAE}} \sim \frac{v_A}{4\pi q_{TAE}} R,$$

where $c \sim \rho_{m+1}/nR$, $R$ is the major radius, and $q_{TAE} = (m-1/2)/n$. Note that the toroidal effect causes the $m$ and $m+1$ harmonics to couple in the range of $m/n < q_{mn} < (m+1/2)/n + c$. Thus, AEs have changed from RSAEs to TAEs in this $q_{mn}$ range. From eqs. 1-3, we show the change of RSAE frequency and its transition to the TAE by decreasing $q_{mn}$ in Fig. 2. The bold lines denote the frequencies of the $n = 1$ AEs and thin lines are for $n = 2$. For $n = 1$ AEs, the HRSAE and the LRSAE merge and change to TAE when $q_{mn}$ decreases by unity. For $n =

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Fig. 3 The profile of safety factor when NNBI was injected.

Fig. 4 Temporal evolution of plasma current (a), power of PNBI and NNBI (b), and line integral electron density (c).
2, the modes merge and change to TAE twice in the same period, and the frequency change is about two times faster than that for \( n = 1 \).

3. AEs in RS plasmas in JT-60U NNBI Experiments

The accurate reconstruction of \( q \)-profile is required for AE investigation. The \( q \)-profile measurement with MSE including the reconstruction of magnetic surface has recently been greatly improved. Moreover, the \( q \)-profile measurement is more accurate with higher magnetic field. Therefore, the experiments for AEs destabilized by NNBI in JT-60U RS plasmas were carried out with a relatively high 3.73 T toroidal field (\( B_t \)) and 1.3 MA plasma current (\( I_p \)). Furthermore, for RSAE study, the position of \( q_{\text{min}} \) is formed at the more outer region of plasma as possible and the value of \( q_{\text{min}} \) is reduced to below 3. To investigate the effect of \( q \)-profiles on AE stability, the \( q \)-profile is changed during NNBI (Fig. 3) by changing the ramp-up rate of plasma current and the injected power of positive neutral beams (PNB) as shown in Fig. 4. The beta value of energetic particles is relatively low at \( \beta_e \approx 0.1 \sim 0.2 \% \), which is, however, comparable to that expected for ITER. We measured the AE frequency and amplitude and determined the toroidal mode number \( n \) by Mirnov coils with a 500-kHz sampling rate.

In the E40739 shot the hydrogen NNBI of \( E_{\text{NNBI}} \sim 360 \) keV energy and \( P_{\text{NNBI}} \sim 4 \) MW power was injected into hydrogen RS plasmas from \( t = 5.8 \) s to \( t = 7.2 \) s. The ratio of the beam velocity parallel to toroidal magnetic field to the central Alfvén velocity is \( V_b/V_A(0) \approx 0.5 \). \( q \) in the region of \( r/a < 0.65 \) decreased gradually as shown in Fig. 5. Also, \( q_{\text{min}} \) decreased steadily from \( \sim 3.0 \) to \( \sim 2.45 \) in the period from \( 5.8 \) s to \( 7.0 \) s as shown in Fig. 6(a). Magnetic fluctuations with large frequency sweeping in the AE frequency range were observed (shown in Fig. 6(c)). Because \( \beta_e \) is relatively low (\( \beta_e \approx 0.12 \% \)), only the \( n = 1 \) AE is observed. AEs with \( n > 1 \) are barely observed and they can be excited if \( \beta_e \) increases. An \( n = 1 \) AE frequency sweeps up from 40 to 90 kHz over the period \( t = 6.0 - 6.5 \) s. Another \( n = 1 \) mode appears to be sweeping down in frequency from 130 to 90 kHz in the same period. The large upward/downward frequency sweeping from \( t = 6.0 \sim 6.5 \) s cannot be explained by the change of \( V_A \) because the electron density

![FIG. 5 Temporal evolution of the q-profile measured with MSE.](image)

![FIG. 6 Temporal evolution of qmin (a), line averaged electron density (b), a typical behavior of frequency spectrum of the n = 1 AE (c). Broken lines denote estimated frequency from the RSAE model normalized by observed frequency at q = 2.5](image)
changed only less than 5%. In the period $t = 6.5 - 6.8$ s the $n = 1$ AE frequency saturates. After $t \sim 6.8$ s the $n = 1$ AE frequency decreases due to the increase of electron density. The model of RSAE and its transition to TAE can explain the observed upward and downward frequency sweeping and subsequent frequency saturation shown in Fig. 6(c), where the broken lines denote the estimated model frequency calculated with eqs. 1-3 and normalized by the observed frequency at $t = 6.8$ s ($q = 2.5$). The hatched area in Fig. 2 is corresponding to observed AE frequency in Fig. 6(c).

The observed AE amplitude are enhanced during $t \sim 6.65 - 6.85$ s in Fig. 6(c) when the AE frequency is saturated. To investigate the dependence of mode amplitude on the $q$-profile change, we show the mode amplitude versus $q_{\text{min}}$ for three shots in Fig. 7. When NNBI was injected, the values of $q_{\text{min}}$ are about 3.0, 2.8 and 2.6 for the shots of E40739, E40744 and E40743, respectively. For all these cases the $n = 1$ mode amplitude is largest in the range $\sim 2.4 < q_{\text{min}} < \sim 2.7$, which is independent of the time length after NNBI injection. This experimental result is consistent with results of calculation with TASK/WM.

We present a possible reason why AE amplitude is enhanced in the transition from LRSAE to TAE. Let us again consider the $n=1$ AE in the range of $2 < q_{\text{min}} < 3$. For $2.7 < q_{\text{min}} < 3$, the lower frequency LRSAE is close to the adjacent Alfvén continuum and suffers large continuum damping (Fig. 8(a)). For $2.5 < q_{\text{min}} < 2.7$ the toroidal coupling effect of the $m = 2$ and $m = 3$ harmonics modifies the RSAE gap to TAE gap. The mode is a mixture of TAE and RSAE and will suffer weaker continuum damping (Fig. 8(b)). For $q_{\text{min}} < 2.5$ RSAE no longer exists because the RSAE gap has closed up at $q_{\text{min}} = 2.5$ and two TAE gaps open up. TAE at the inner gap is usually destabilized due to large pressure gradient of energetic particles. However, it also encounters continuum damping at the outer TAE gap location because the

![Fig. 8 Schematic drawing of the $n = 1$ spectrum of AEIs (bottom) with $q$-profile (upper) in the RS plasma for RSAE phase of $q_{\text{min}} = 2.8$ (a), transition phase of $q_{\text{min}} = 2.5$ (b), and TAE phase of $q_{\text{min}} = 2.3$ (c). In both cases, $n = 1$, $q(0) = 4$ at the plasma center, and $q(a) = 5$ at the plasma edge. In the case of RSAE and TAE phase AEIs suffer large continuum damping.](image-url)
density at the inner TAE gap is larger than that at the outer gap (Fig. 8(c)). From these considerations, we expect the continuum damping rate of AEs to be smallest in the $2.4 < q_{mn} < 2.7$ range, which was confirmed by the TASK/WM code calculation [9]. Thus, to avoid large amplitude AEs in RS plasmas, it is preferable to operate outside the $q_{mn}$ transition range for $n = 1$.

4. RSAE in RS Plasmas in JT-60U ICRF Experiments

The RSAE model has also been applied to understand AEs with large and rapid upward frequency chirping in previous JT-60U ICRF RS plasma experiments [7,8]. As shown in Fig. 9(a) the frequencies of $n \sim 2$-11 modes observed from 6.0 s to 6.6 s increase $\sim 100$ - $300$ % in a short period of $\sim 200$ ms. The large and rapid change in the mode frequency cannot be explained by a temporal change in plasma density and toroidal flow. The line averaged electron density decreases only $\sim 4$ % during 6.2 to 6.6 s. Moreover, from the profile of the toroidal rotation velocity measured from the charge exchange recombination spectroscopy the Doppler-shift frequency is less than $n$ kHz for the toroidal mode number $n$. Furthermore, these modes have the following features: (1) the $n = 1$ mode was not detected; (2) frequencies of higher $n$ AEs increase more rapidly; (3) the higher $n$ AEs are detected more frequently; and (4) there is a period in which no modes are detected ($t \sim 6.65$ - $6.75$ s). From $t = 6$ to 7.5 s $q_{mn}$ decreases from $\sim 2.4$ to $\sim 1.7$, and we plot the frequencies of $n = 1$ - 9 LRSAEs and TAEs versus $q_{mn}$ in Fig. 9(b), which is calculated by eqs. 1-3, agrees well with the observation shown in Fig. 9(a). The $n = 1$ RSAE cannot be detected because it is predicted that the RSAE suffers large continuum damping due to its low frequency. From the RSAE model the frequency sweeping rate of RSAEs with a toroidal mode number $n$ is $n$ times faster than that with $n = 1$, which is consistent with the observed faster increase in frequency for higher $n$ mode. Also, because they emerge $n$ times during the period $q_{mn}$ decreases by unity, higher-$n$ RSAE should be observed $n$ times more frequently. The $1.95 < q_{mn} < 2.05$ range in Fig. 9(b) is filled with $n \geq 9$ AEs, which are not observed from 6.65 s to 6.75 s in Fig. 9(a).

5. Fast ion transport induced by AE

NNB-AE experiments are performed in JT-60U deuterium RS plasmas with relatively high $\beta_i$.

FIG. 9 Typical temporal evolution of AE frequency changes observed in a JT-60U ICRF heated RS plasma [8] (left), and the frequencies of $n = 1$ - 9 LRSAEs (solid upward-sloping lines) and TAEs (horizontal aligned marks) as a function of the $q_{min}$ decrement (right).
to investigate transport of fast ions. WS deuterium discharge with $B_i \sim 1.2$ T, $V_r/V_A \sim 1$ and $<\beta_i> \sim 0.6\%$ was performed for investigation of fast ion transport induced by large bursting AE. The drop of the neutron emission rate and increase in fast neutral fluxes have been observed as the result of the enhanced radial transport of fast ions during the bursting modes (Fig. 10) \cite{13, 14}. The increased fast neutral flux depending its energy is consistent with wave-particle resonance and loss mechanism \cite{13}. We newly installed six channel neutron profile monitor to measure 2.45 MeV DD fusion neutron profile for investigation of fast ion behavior further \cite{15}. On occurrence of large bursting modes, peripheral signals ($r/a \geq 0.48$) increase and center signals ($r/a \leq 0.34$) decrease, which shows the redistribution of energetic ions. It is estimated by OFMC code that the fast ion profile around $r/a \sim 0.2 - 0.4$ is steep. This suggests that fast ions in such region mainly interact with AE modes and the steeper gradient is relaxed in terms of redistribution of the fast ions to the outer plasma region of $r/a \sim 0.4$.

In the E040739 shot there is no neutron flux data because of the hydrogen plasma. Therefore, we performed RS deuterium discharge with a smaller $B_i$ of 2.1 T or study of fast ion transport with MSE $q$-profile measurement. In this discharge, we observed $n = 2$ and $n = 3$ AEs in addition to the $n = 1$ AE because of relatively higher $\beta_i \sim 0.4\%$. We observed suppression of the total neutron count, when the $n = 1$ RSME and AE in the transition from RSME to TAE were destabilized as shown Fig. 11. After $n = 1$ AE disappeared after $t \sim 5.7$ s, the total neutron count recovered about 20\% even though $n = 2$ and $n = 3$ AEs still remained. This suppression of neutron count is considered due to fast ion loss because the neutron profile monitor suggested that neutron emission is suppressed in all over the plasma region and electron density and ion temperature did not changed from $t = 5.8$ s to 6.2 s. Difference in transport of fast ions seems to be explained by the difference of eigenfunctions of AEs in the WS experiment and in the RS experiment. For the previous WS case, the eigenfunction seems to be localized in the central plasma region, namely such as core localized mode, because of low magnetic shear there. On contrary this, for the RS case, RSME exists around $r/a \sim 0.5$. Therefore, AEs can transport fast ions outer region in the RS plasmas than in the WS plasma.

\begin{figure}[h]
\centering
\includegraphics[width=\textwidth]{fig10}
\caption{Temporal evolution of neutron yields and magnetic fluctuation of AEs during large bursting modes were observed.}
\end{figure}

\begin{figure}[h]
\centering
\includegraphics[width=\textwidth]{fig11}
\caption{Temporal evolution of $q_{min}$ (a), a typical behavior of frequency spectrum of AEs (b), total neutron count rate (c).}
\end{figure}
6. Conclusions

We performed NNB-AE experiment in JT-60U RS plasma with accurate $q$-profile measurement to investigate the property of AEs in RS plasmas. The $n=1$ NNB-AEs with rapid frequency sweeping and then saturation of frequency as $q_{\text{min}}$ decreases were observed in these plasmas. We introduced the RSAE model to explain such AEs with a rapid frequency sweeping observed in RS plasmas. The frequency sweeping AEs and sequent the saturation of frequency can be explained by RSAEs and the transition from RSAE to TAE. The previously observed rapid frequency sweeping of ICRF-AE with $n>1$ in JT-60U RS plasmas can also be well explained by the RSAE model. The AE amplitude is maximum during this transition, e.g., $-2.4 < q_{\text{min}} < -2.7$ for the $n=1$ mode. On the other hand, we newly installed six channel neutron profile monitor to measure 2.45 MeV DD fusion neutron profile for investigation of fast ion behavior. By using this neutron profile detector, we observed the retribution of fast ions in WS deuterium discharge with $B_r \sim 1.2$ T and $\beta_n \sim 0.6 \%$, when large bursting AEs were destabilized. As opposed to this, substantial fast ion loss was observed in RS deuterium discharge with $B_r \sim 2.1$ T and $\beta_n \sim 0.4 \%$, when the $n=1$ mode amplitude is large at transition from RSAE to TAE. Difference in transport of fast ions seems to be explained by the difference of location of eigenfunctions of AEs in WS and RS plasmas. From RSAE model RSAEs with $n$ transit to TAE $n$ times when $q_{\text{min}}$ decrease by unity. Therefore, the $n=1$ RSAE transits to TAE only once. It is preferable to operate RS plasmas outside of the $q_{\text{min}}$ transition range for $n=1$, e.g., $-2.4 < q_{\text{min}} < -2.7$ AE to avoid substantial fast ion loss due to large amplitude AEs.

Acknowledgments

The authors would like to thank the members of JAERI who have contributed to the JT-60 project.

References

Property of Alfvén Eigenmode in JT-60U
Reversed Shear and Weak Shear Discharges

M. Takachi, A. Fukuyama¹, K. Shinohara, M. Ishikawa², Y.
Kusama, S. Takei, T. Fujita, T. Okabe, T. Suzuki, N. Oyama,
T. Ozaki, A. Morikawa, C. Z. Cheng³, K. N. Gorelenkov⁴,
G. J. Kramer⁵, R. Nazikian⁶ and the JT-60 team

Japan Atomic Energy Research Institute, Japan
¹Department of Nuclear Engineering, Kyushu University, Japan.
²Plasma Research Center, University of Tsukuba, Japan.
³Princeton Plasma Physics Laboratory, USA

16th IAEA Fusion Energy Conference
Lyon, France, 14 to 18 October 2002

Introduction

- RS plasma - extensively studied
  An attractive operation for SS tokamak reactors with good confinement
- AE - extensively studied in PS
  Leading to the quenching of fusion ignition and damage of the first wall
- AE in RS - not studied systematically yet
  Characteristics - not well-known
  e.g. puzzling AE with large and rapid frequency sweeping
  - Prediction of frequency sweeping AEs in tokamak RS with TASKWM [Fukuyama, 1997]
  - Explanation of puzzling frequency change of AEs in CHS heliotron
    [Takachi, 1997]

We introduce a reversed-shear-induced AE (RSAE) to interpret frequency chirping AEs observed in tokamak RS.
Fast ion transport Induced by AEs is investigated with the newly installed neutron profile measurement.

Outline

- Introduction
- Property of the AE in RS plasmas
  - Reversed-Shear-induced Alfvén Eigenmodes (RSAE)
- Results of experiment for AE in RS
- Results of experiment for transport of the fast ions
  Induced by AEs
- Summary

Property of the AE in a RS plasma
- RSAE can explain the frequency chirping AE

Let us consider AEs at $2.0 < q_{\text{min}} < 3.0$
Two $q=2.5$ Two TAE gaps
Formed by toroidal coupling
Little frequency change

$2.0 < q_{\text{min}} < 2.5$
Two $q=2.5$
not TAE gap
Formed by RS $q$-profile
Quick frequency change
NNB-AE experiment in JT-60U RS with various q-profiles—high $B_t$ for accurate q measurement—

Ability of various q-profile in JT-60U RS for AE experiment
- by changing $l_p$ Ramp-up rate and preheating power
Accurate q-profile is required for AE experiments -> Higher $B_t$
- Furthermore for confirming RSAE model
- The outer $q_{min}$ position and $q_{min} < 3$
$E_p=3.7T$, $l_p=1.3MA$, $E_{NNBI} \sim 360$ keV, $P_{NNBI} \sim 4$ MW, $\beta_p \sim 0.1-0.2\%$

$n=1$ AE at the transition from RSAE to TAE has largest mode amplitude

To investigate dependence of AE amplitude on q-profile, NNB was injected into RS with various $q_{min}$

Only $n=1$ AE was observed when $\beta_p$ is low (< -0.2%) $n=1$ AE is most unstable
$n=1$ AE amplitude is maximum at $-2.4 \leq q_{min} \leq -2.7$, independent of time length after NNB injection
AE in transition from RSAE to TAE is most unstable

From RSAE model, the $n=1$ RSAEs transits to TAE only once when $q_{min}$ decrease by one

It is preferable to operate RS plasmas outside of the transition range of $q_{min}$ for $n=1$, e.g., $-2.4 < q_{min} < -2.7$
to avoid largest amplitude AE.
RSAE can reproduce rapid frequency chirping AEs with n=2 previously observed in JT-60U ICRF RS

- No n=1 mode
- Frequencies of higher n AEs increase more rapidly
- Higher n AEs are detected more frequently
- A period in which no modes are detected (1–6.65–6.75 s)
- The largest mode amplitude around highest frequency
- Hardly observation of TAE after the transition

Investigation of fast ion transport induced by AEs
- with 6 channel neutron profile monitor

Large bursting AEs and reduction of total neutron count in WS with B_t=1.2 T, B_r=0.6 % and V/V_A=1, [Shinohara, 2000] with neutron profile monitor

Peripheral signals increase and center signals decrease
- Redistribution of fast ions

Summary

NNB-AE experiments in JT-60U RS with various q-profile.
- Observation of puzzling rapid frequency sweeping AEs

To explain this puzzling AEs

Introducing RSAE model

NNB-AE experiments with accurate q-profile measurement
- Identification of RSAE
- Interpretation of the frequency sweeping AEs by RSAE
- The largest n=1 AE amplitude during the transition
- Observation of fast ion loss with neutron profile measurement during the n=1 AE in this transition

It is preferable to operate RS plasmas outside of q_{min} transition range for n=1 AE, e.g., -2.4 < q_{min} < -2.7 to avoid substantial fast ion loss due to large amplitude AEs.
1.10 Driving Mechanism of SOL Plasma Flow and Effects on the Divertor Performance in JT-60U


1) Japan Atomic Energy Research Institute, Naka-machi, Naka-gun, Ibaraki 311-0193, Japan
2) Lawrence Livermore National Laboratory, P.O.Box 808, Livermore, CA 94550, USA

e-mail: asakuran@fusion.naka.jaeri.go.jp

Abstract. The measurements of the SOL flow and plasma profiles both at the high-field-side (HFS) and low-field-side (LFS), for the first time, identified the SOL flow pattern and its driving mechanism. "Flow reversal" was found near the HFS and LFS separatrix of the main plasma for the ion VB drift direction towards the divertor. Radial profiles of the SOL flow were similar to those calculated numerically using the UEDGE code with the plasma drifts included although Mach numbers in measurements were greater than those obtained numerically. Particle fluxes towards the HFS and LFS divertors produced by the parallel SOL flow and ExB drift flow were evaluated. The particle flux for the case of intense gas puff and divertor pump (puff and pump) was investigated, and it was found that both the Mach number and collisionality were enhanced, in particular, at HFS. Drift flow in the private flux region was also evaluated, and important physics issues for the divertor design and operation, such as in-out asymmetries of the heat and particle fluxes, and control of impurity ions were investigated.

1. Introduction

The scrape-off layer (SOL) flow plays an important role in the plasma transport along the field lines. Parallel SOL flow is expected to increase with an intense gas puff and divertor pump (puff and pump), which is proposed to improve impurity shielding from the core plasma. On the other hand, the SOL flow away from the LFS divertor, "flow reversal" (opposite to what one expects from a simple picture of the plasma flow), has been generally observed at various locations around the plasma edge, in particular, for the ion VB drift direction towards the divertor [1-4]. Recently, mechanisms producing parallel SOL flow, resulting from the poloidal variation of plasma drift velocity were investigated [5,6]. Drift flow in the private flux region was proposed as a candidate mechanism to produce in-out asymmetry in divertor particle flux [7,8]. Quantitative evaluation of the drift effects should be established in order to control the SOL and divertor plasmas in magnetic configurations relevant to a tokamak reactor.

Determination of the SOL flow pattern has been recently advanced in JT-60U experiments. Reciprocating Mach probes were installed at the high-field-side (HFS) baffle, low-field-side (LFS) midplane and just below the X-point (Fig.1). SOL flow pattern is shown in Sec. 2. It is compared to UEDGE simulation results with the plasma drifts included in Sec. 3. SOL particle fluxes towards the HFS and LFS divertors are, for the first time, evaluated from components of the parallel SOL flow and perpendicular ExB drift flow. Important physics issues for the divertor design and operation: in-out asymmetry in the ion flux and impurity control by puff and pump, are discussed in Sec. 4 and 5. Summary and conclusions are given in Sec. 6.

2. SOL Flow Measurements at HFS and LFS

Profiles of ion saturation currents at the electron- and ion-drift sides, j_e^e and j_e^i, electron
temperatures, $T_e^{\pm}$ and $T_i^{\pm}$, and floating potential, $V_f$, are measured with spatial resolution of 1-2 mm. The direction of the plasma flow along the field lines and the Mach number are deduced from the ratio of $j_i^{e+}$ and $j_i^{i+}$, using Hutchinson’s formula [9]: $M = 0.35 \ln(j_i^{e+}/j_i^{i+})$, where positive and negative values show the direction towards the LFS and HFS divertors, respectively. Plasma potential, $V_p$, and radial electric field, $E_r$, are calculated from a sheath model ($V_p = V_{Te}$, $2.75 T_e$) and its differentiation. Parameters of the L-mode discharge, $I_p = 1.6$ MA, $B_t = 3.3$ T, $q_{95} = 3.5$ and $P_{NBI} = 4.3$ MW, are fixed. Main plasma density, $\bar{n}_e$, is maintained during probe measurements, and it changes from $1.2 \times 10^{19}$ to $3.9 \times 10^{19}$ m$^{-3}$ ($\bar{n}_e/\bar{n}_e^{GW} = 0.23 - 0.73$, where $\bar{n}_e^{GW} = 5.2 \times 10^{19}$ m$^{-3}$) on a shot-by-shot basis for normal ion $VB$ drift direction.

Both the ion $VB$ drift direction and $\bar{n}_e$ were found to affect the plasma flow velocity [6,10]. Profiles of the Mach number at the three locations are shown in Fig. 2 for the ion $VB$ drift direction towards the divertor and relatively low $\bar{n}_e = 1.6 \times 10^{19}$ m$^{-3}$. Three profiles are mapped to the LFS midplane, and the data in the private flux region are not plotted. For the midplane radius ($r_{mid}$) less or greater than 4 cm, field lines are connected to the divertor plate and baffle, respectively. Results of the LFS flow profiles show that flow reversal occurs near the separatrix of the main plasma with $M = -0.4$. The flow reversal gradually reduces at the outer flux surfaces, whereas fast SOL flow ($M \sim 0.5$) towards the LFS divertor is observed below the X-point.

Characteristics of HFS SOL flow change near the separatrix. The SOL flow away from the HFS divertor with $|M| = 0.1-0.2$ is found at and just outside the separatrix, and width of the flow reversal ($r_{mid} \sim 0.4$ cm) is narrower than that observed LFS midplane ($r_{mid} < 5$ cm). The Mach numbers of the flow reversal decrease gradually with increasing $\bar{n}_e$. On the other hand, at the outer flux surfaces ($l < r_{mid} < 4$ cm), subsonic SOL flow towards the HFS divertor ($M \sim -0.5$) is produced. Mach numbers are similar both at LFS midplane and HFS. Parallel SOL flow may be driven from LFS to HFS on the outer flux surfaces. The Mach numbers increase with $\bar{n}_e$. Particle flux towards the HFS divertor, which is produced by the SOL flow ($n_i V_f$) and is represented by $n_i M C_s \sim M j_i / e$, is also enhanced. Particle flux towards the divertor is investigated in Sec. 4.

For the ion $VB$ drift direction away from the divertor, SOL flow towards LFS divertor is observed both at LFS midplane and near X-point. The SOL flow at HFS is towards the HFS divertor. All Mach numbers near the separatrix are small ($|M| = 0.2-0.3$). From these observations in opposite $VB$ drift configurations, one can conclude that the SOL flow near the
separatrix of the main plasma edge is driven against the $\nabla B$ drift direction. A large influence of the plasma drifts is expected.

3. SOL Plasma Simulation With Drift Effects

Two-dimensional plasma flow pattern was investigated in the toroidal plasma and divertor geometries. Drift effects such as $\mathbf{E} \times \mathbf{B}$, $\mathbf{B} \times \nabla \mathbf{B}$ and diamagnetic drifts have been included in the simulation code of the plasma fluid models, UEDGE [11]. At this stage, plasma calculation mesh covers the edge and SOL area at the LFS midplane radius of $-3 < r_{\text{mid}} < 5 \text{cm}$. Constant diffusion coefficients, $\chi_e = \chi_i = 1 \text{ m}^2\text{s}^{-1}$ and $D = 0.25 \text{ m}^2\text{s}^{-1}$ over the SOL area, are used to reproduce the measured $T_e$ and $n_e$ profiles at LFS midplane.

Figure 3 shows Mach numbers of the parallel SOL flow along the normalized poloidal distance from HFS target to LFS target at $r_{\text{mid}} = 0.6 \text{ cm}$. Here, the calculation with including small drift effects of 55% in iterations is also shown. It is found that the SOL flow towards the HFS divertor, i.e. flow reversal, is produced at LFS SOL, while the SOL flow towards the LFS divertor is seen near the X-point. With increasing drift effects, Mach numbers of the flow reversal increase (in particular, at LFS SOL midplane). Simulation results reveal that the flow reversal is mainly caused by the ion $\mathbf{B} \times \nabla \mathbf{B}$ drift, which increases the ion pressure downward. On the other hand, the Mach number at the X-point does not change near the separatrix at relatively low $\bar{n}$.

Calculated SOL flow profiles are compared to the measured ones as shown in Fig. 4 [12]. At the LFS midplane, the flow reversal near the separatrix ($|/M| = 0.2$) is comparable to the measurement ($|/M| = 0.3-0.4$). The calculated SOL flow at the outer flux surfaces ($r_{\text{mid}} > 3 \text{ cm}$) is towards the LFS divertor: the width of the flow reversal is narrow, while it is observed at further outer radius ($r_{\text{mid}} > 5 \text{ cm}$) in the experiment. Neutral flux from the first wall is larger in experiments, which is not modeled consistently to reproduce the $D_\alpha$ brightness profile. This influence should be simulated in order to understand radial diffusion of the SOL plasma as well as parallel flow pattern.
Calculated Mach numbers above the HFS baffle and near the X-point increase towards the diverter when the drift effects are included. At HFS SOL, the Mach numbers at separatrix and $r_{m_2} \sim 2$ cm are comparable to the measurements, but those within $0 < r_{m_2} < 2$ cm are smaller. On the other hand, near the X-point of LFS, the Mach numbers at $r_{m_2} < 2$ cm (including separatrix) are small compared to the measurements ($\langle M \rangle \sim 0.4$). Mechanism for the subsonic flow in experiments is not understood yet. It could be caused by the fact that distributions of impurity ion densities are not well reproduced to simulate the experimental radiation profile in the divertor. However, near the separatrix, influences of the subsonic flow on net particle flux towards the divertor are relatively small since particle flux caused by the drift flow is larger.

4. Particle Fluxes Towards HFS and LFS Divertors

In this section, net particle fluxes towards the HFS and LFS divertors are investigated [13]. Components of parallel SOL flow ($n_i V_i$) and drift flow ($n_i V_{drift}$) to the particle flux towards the divertor are described as $n_i V_i \Theta$ and $n_i V_{drift} \Phi$, respectively. Here, $\Theta = B_y / B_z$ varies in torus and $\Phi = B_r / B_z (-1)$. Both components change with increasing $n_i$. At the same time, large $E \times B$ drift flow in the private flux region influences the in-out asymmetry in the divertor ion flux [6]. Distributions of the parallel and drift flows upstream of the HFS and LFS divertors and in the private flux region are for the first time established enabling one to understand the particle transport in SOL and divertor.

4.1 Parallel SOL Flow and $E \times B$ Drift Flow

Poloidal components of particle flux densities: parallel SOL flow ($n_i V_i \Theta$) and $E \times B$ drift flow ($n_i V_{drift}^{E \times B} \Phi$) are shown in Fig. 5 for the ion $V \times B$ drift direction towards the divertor. Here, $n_i = n_e$ is assumed. Drift flow has a positive value, which produces particle fluxes away from and towards the divertor at HFS and LFS, respectively. The diamagnetic flow ($n_i V_{drift}^{da}$) does not constitute the particle flux onto the divertor and is not shown. At the HFS and LFS, poloidal flux density of the drift flow is dominant near the separatrix ($r_{m_2} < 0.4$ cm). Poloidal components of particle flux density in the private flux region are shown in Fig. 5(b), which is produced by negative $E_r$ in the plasma under the detached divertor condition. The $E_r E \times B$ drift flux is very large and it contributes to particle transport from LFS to HFS divertor.

In the detached divertor, negative $E_r$ appears near the boundary of detached and attached plasmas in the common flux region. This $E \times B$ drift would produce particle flux away from the LFS divertor to the X-point. On the other hand, parallel SOL flow towards the LFS divertor increases up to the sonic level since the ionization front moving from the target plate to the X-
point [4]. The net particle flux is carried towards the LFS divertor.

4.2 Influence of ExB Drift on Particle Flux

Influences of the drift flow on total particle fluxes, \( \Gamma_p^{HFS} \) and \( \Gamma_p^{LFS} \), are investigated. \( \Gamma_p^{HFS} \) and \( \Gamma_p^{LFS} \) are calculated by integrating \( n_e V_{\parallel} \Theta + n_e V_{\parallel} \Phi \) across the SOL along the probe scan from the separatrix \( (r=0) \) to the most outer radius \( (r=r_{div}) \), where field line is connected to the divertor, as follows,

\[
\Gamma_p^{HFS/LFS} = \int_0^{r_{div}} 2\pi R n_e (V_{\parallel} \Theta + V_{\parallel} \Phi) V\Phi \cdot dr,
\]

where positive value shows particle flux towards the LFS divertor. Figure 6 shows \( |\Gamma_p^{HFS}| \) and \( \Gamma_p^{LFS} \), and the ratio of \( E \times B \) drift flux to parallel flux as a function of \( \pi_e/n_G^W \). Equation (1) is written as \( \Gamma_p^{HFS/LFS} = \Gamma_p^{HFS/LFS} + \Gamma_p^{drift} \), where \( \Gamma_p^{HFS} \) and \( \Gamma_p^{LFS} \) have negative and positive values, respectively. Total particle flux at the private flux region, \( \Gamma_p^{prv} \), is calculated from Eq.(1), where two components are integrated in the private flux region \((-2 < r < 0 \text{ cm})\).

At low \( \pi_e \) (\( \pi_e/n_G^W =0.24-0.34 \)), \( \Gamma_p^{HFS} \) of \((-3.0-4.0) \times 10^{21} \text{ s}^{-1} \) is larger than \( \Gamma_p^{drift} \) of \((0.9-2.1) \times 10^{21} \text{ s}^{-1} \). \( \Gamma_p^{HFS} \) is 30-50% of \( |\Gamma_p^{HFS}| \). Thus, the direction of \( \Gamma_p^{HFS} \) is towards the HFS divertor, and \( \Gamma_p^{HFS} \) is \((-1.5-2.5) \times 10^{21} \text{ s}^{-1} \). On the other hand, \( \Gamma_p^{LFS} \) and \( \Gamma_p^{drift} \) are \((3.0-4.0) \times 10^{21} \) and \((1.6-2.9) \times 10^{21} \text{ s}^{-1} \), respectively. \( \Gamma_p^{drift} \) corresponds to 45-80% of \( \Gamma_p^{LFS} \), and \( \Gamma_p^{LFS} \) is \((5.9-7.0) \times 10^{21} \text{ s}^{-1} \). As a result, \( \Gamma_p^{LFS} \) is larger than \( \Gamma_p^{HFS} \), and the asymmetry is produced mostly by the drift flow in SOL. On the other hand, \( \Gamma_p^{prv} \) of \((-3.7-3.8) \times 10^{21} \text{ s}^{-1} \) is large, and it should contribute to in-out asymmetry in the divertor ion flux.

When the detachment occurs at LFS divertor, however, \( \Gamma_p^{prv} \) decreases to zero. At the same time, with increasing \( \pi_e \), both \( |\Gamma_p^{HFS}| \) and \( \Gamma_p^{prv} \) increase largely and become dominant in particle transport towards the divertor. Here, \( \Gamma_p^{drift} \) changes the direction, as mentioned in Sec. 4.1.

4.3 In-Out Asymmetry in Divertor Ion Flux

Influences of \( \Gamma_p^{HFS} \), \( \Gamma_p^{LFS} \) and \( \Gamma_p^{prv} \) on in-out asymmetry in the divertor ion flux are discussed. We make the following assumption: a part of \( \Gamma_p^{LFS} \) is exhausted into the private flux region by diffusion and radial drifts before reaching the target plate: such particle flux would be comparable to \( |\Gamma_p^{prv}| \). Total particle fluxes towards the HFS and LFS divertors are estimated as \( \Gamma_p^{HFS} + \Gamma_p^{prv} \) and \( \Gamma_p^{LFS} - |\Gamma_p^{prv}| \), respectively. \( |\Gamma_p^{HFS} + \Gamma_p^{prv}| \) and \( \Gamma_p^{LFS} - |\Gamma_p^{prv}| \) are shown in Fig. 7 as a

![FIG. 6. Net poloidal fluxes towards divertors, \( \Gamma_p^{HFS} \) and \( \Gamma_p^{LFS} \), and ratio of drift flux to parallel flux, \( \Gamma_p^{drift}/\Gamma_p^{prv} \), as a function of \( \pi_e/n_G^W \). (a) for HFS and (b) for LFS near X-point. Divertor flux from LFS to HFS, \( \Gamma_p^{prv} \), is also shown.](image-url)
function of \( \bar{n}_p/n^{GW}_p \). For the attached divertor conditions \((\bar{n}_p/n^{GW}_p = 0.24 - 0.45)\), \(|\Gamma_p^{HFS} + \Gamma_p^{Prv}|(5.4 - 12.6) \times 10^{21} \text{s}^{-1}\) is a factor of 2-3 larger than \(\Gamma_p^{LFS} - |\Gamma_p^{Prv}|(2.2 - 4.4) \times 10^{21} \text{s}^{-1}\). Large contribution of \(\Gamma_p^{Prv}\) to the HFS-enhanced asymmetry in the divertor ion flux is determined.

When the detachment occurs at the LFS divertor at \(\bar{n}_p/n^{GW}_p > 0.46\), \(\Gamma_p^{Prv}\) disappears and net particle fluxes towards the HFS and LFS divertors are described as \(\Gamma_p^{HFS}(-\Gamma_p^{HFS})\) and \(\Gamma_p^{LFS}(-\Gamma_p^{LFS})\), respectively. At the same time, \(\Gamma_p^{LFS}\) becomes larger than \(\Gamma_p^{HFS}\). As a result, \(\Gamma_p^{LFS}\) becomes a factor of 1.3-1.8 larger than \(\Gamma_p^{HFS}\). The asymmetry in \(\Gamma_p\) is small compared to that in the attached divertor. Similar HFS-enhanced asymmetries in the particle recycling and neutral pressure have been generally observed [10]. When the divertor detachment occurs such asymmetries become small and reversed. These characteristics of the particle transport in the divertor are determined by changes in \(\Gamma_p^{HFS}\), \(\Gamma_p^{LFS}\), and \(\Gamma_p^{Prv}\).

Contributions of drift effects to the particle transport would be an important question for the divertor design and operations in tokamak reactor such as ITER. Although the high density plasma \((\bar{n}_p/n^{GW}_p \sim 0.85)\) is sustained, collisionality of the SOL plasma is relatively low \((v_e^* = L_e/\lambda_e \sim 5-10)\) since \(T_e\) at separatrix \((-150\text{ eV})\) would be high at \(n_e \sim 3.5 \times 10^{19} \text{ m}^{-3}\). Relatively large \(E_x\) is expected in such low \(v_e^*\), which corresponds to database at \(\bar{n}_p/n^{GW}_p \sim 0.4\) in Fig.6. Then, \(\Gamma_p^{drift^{HFS}}/\Gamma_p^{HFS}\) of \(-30\%\) and \(\Gamma_p^{drift^{LFS}}/\Gamma_p^{LFS}\) of \(-60\%\) would be anticipated. At the same time, \(E_xB\) drift flow in the private flux region is expected since the divertor detachment is localized near the strike-points: private plasma below the X-point is attached. Particle flux towards the divertor would be influenced by these drifts, and a design work with including these drifts will be useful to optimize the divertor and pump geometries.

5. Effect of Gas Puff Location on SOL Flow

SOL flow pattern is modified depending on the gas puff location, and a technique of the puff and pump was demonstrated to exhaust impurity ions (helium, neon and carbon) from the core plasma [14,15]. However, the SOL flow pattern has not been determined experimentally. The SOL plasma flow was investigated during strong gas puff from the plasma top ("main puff") and divertor ("divertor puff") as shown in Fig.8(a). Mach probes locations are downstream and upstream from the main puff and divertor puff locations, respectively. Constant puff rate is kept during the measurements under the attached divertor condition, and it changes from 20 to 60 Pam s\(^{-1}\) \((\Gamma_p^{puff} = 1.1 - 3.2 \times 10^{22} \text{ Atoms s}^{-1})\) on a shot-by-shot basis. Hydrogen L-mode plasma with normal ion \(VB\) drift direction was used in these experiments.
Profiles of $M$ and $n_e$ are compared for the two cases, where $\bar{n}$, are comparable, $(2.6$-$2.7) \times 10^{19} \text{ m}^{-3}$ ($\bar{n}/n_{GW}^m \sim 0.52$). For the main puff, enhancements of $M$ and $n_e$ are observed both at HFS and LFS, compared to those for the divertor puff. At HFS, $M$ increases by 20% in the wide region between the separatrix and the outer flux surfaces, and $\bar{n}$ also increases by 20-50%. As a result, parallel flux density component ($n_i V_i / \Theta$) increases as shown in Fig.8 (b). On the other hand, at LFS near the X-point, an increase in $M$ (less than 25%) is observed only at $1 < r_{mid} < 2 \text{ cm}$, while $n_e$ is comparable. Thus, small increase in $n_i V_i / \Theta$ is seen only at the outer flux surfaces as shown in Fig.8 (c).

Total parallel flux components, $\Gamma_{p||}^{\text{HFS}}$ and $\Gamma_{p||}^{\text{LFS}}$, which are dominant in the particle transport, are compared between the main puff and the divertor puff. $\Gamma_{p||}^{\text{HFS}}$ is shown in Fig.9. For the main gas puff, large enhancement of $\Gamma_{p||}^{\text{HFS}}$ (a factor of 1.5-2) is found, while $\Gamma_{p||}^{\text{LFS}}$ is 1.1-1.3 times larger. These results suggest that the bulk of the SOL plasma generated by the main puff is brought towards the HFS divertor. This SOL flow direction is consistent with simulation result as shown in Fig.3.

This experiment also demonstrated that electron pressure profiles are comparable for different gas puff cases, and that $T_e$ at HFS SOL and $T_i$ (measured at the LFS plasma top) decrease with the increase in $n_e$ at the HFS SOL, in particular, at the outer flux surfaces. As a

![FIG. 8. (a) Locations of main puff and divertor puff. Poloidal components of parallel flux (b) at HFS, and (c) at LFS, are compared at the same $n_e$.](image)

![FIG. 9. Total parallel particle fluxes towards HFS divertor. Horizontal bars show increment of $n_e$ during gas puffing.](image)

![FIG. 10. Ion collisionalities at separatrix and outer flux surfaces ($r_{mid} - 3 \text{ cm}$) in HFS.](image)
result, electron and ion collisionalitites, $\nu_e^*$ and $\nu_i^*$, are increased largely at the outer flux surfaces of HFS SOL for the main puff case. Figure 10 shows that $\nu_i^*$ is enhanced by the factor of 2-3 at $r_{\text{mid}} = 3$ cm, while an increase in $\nu_i^*$ near the separatrix is small. Values of $\nu_i^*$ (~3-5) at LFS ($r_{\text{mid}} = 0$ and 3 cm) are comparable. A friction force on the impurity flux is enhanced by the increases in $\nu_i^*$ and $M$, which would overcome the thermal force (producing impurity flux towards the main plasma). $Z_{\text{eff}}$ (the main contribution of carbon) is reduced to 1.3 during the strong main puff cases, comparing to 1.4 for the divertor gas puff. This result suggests that the impurity reduction in the main plasma is caused by enhancement of the SOL flow at HFS.

6. Summary and Conclusions

Measurements of the SOL flow both at the HFS and LFS of the JT-60U tokamak, and UEDGE simulation revealed the SOL flow pattern and effects of the plasma drifts on the SOL flow. Drift flow was dominant near the separatrix, which contributes up to 50% (HFS) and 80% (LFS) to net particle transport at relatively low density. At the same time, $E \times B$ drift flow in the private flux region was found to be comparable to the net particle flux towards the HFS divertor, which produces the HFS-enhanced asymmetry in the divertor ion flux under the attached divertor condition (for ion $\nabla B$ drift direction towards the divertor).

A strong gas puff from the main plasma top increased the SOL flow, in particular, at HFS by the factor of 1.5-2 under the attached divertor condition. At the same time, $\nu_i^*$ and $\nu_e^*$ increased in particular at the outer flux surfaces. Increments of both $M$ and $\nu_i^*$ at HFS SOL enhance the friction force on impurity flux, resulting in reduction of $Z_{\text{eff}}$.

In a tokamak reactor such as ITER, drift effects in the particle flux transport would be expected since collisionality of the SOL plasma is relatively low. At the same time, $E \times B$ drift flow in the private region will exist, producing in-out asymmetry in divertor particle flux. Particle flux towards the divertor will be influenced by these drifts, and the design work including the drift effects will be useful to optimize the divertor and pump geometries.

References
Driving mechanism of SOL plasma flow and effects on the divertor performance in JT-60U


Naka, JAERI

Lawrence Livermore National Laboratory, USA

In 19th IAEA Fusion Energy Conference

1. Introduction: SOL flow plays an important role on determining particle flux to the divertor and impurity shielding/exhaust.

Simple SOL plasma model:
Parallel SOL flow is produced towards divertor.

Tokamak experiments:
Puff & Pump reduced impurities in main plasma.

"flow reversal" (flow away from the divertor) was measured by Mach probe (ion flow) and spectroscopy (impurity flow).

Plasma drifts produce ion flux, and the poloidal variation influences parallel SOL flow:
Drift effects (ExB, BxVd and diamagnetic) have been investigated by models and 2D-simulations (ULMGE code).

Determinations of Parallel SOL flow and Drift effects should be established in order to control plasma and impurity transport.

CONTENTS

1. Introduction
2. Plasma flow pattern in SOL: Experiment and UEDGE Simulation
3. Ion flux towards divertor and in-out asymmetry
4. Puff and Pump effects on SOL flow
5. Summary and Conclusions
2. Plasma flow pattern in SOL: Experiment and UEDGE Simulation

- Flow reversal occurs at the main plasma edge.
- High-Field-Side: Flow reversal occurs near separatrix ($r_{mid} < 0.4 cm$), at outer flux surfaces, SOL flow is towards HFS divertor.
- Low-Field-Side: Flow reversal is observed at midplane ($r_{mid} < 5 cm$), just below X-point, SOL flow is towards LFS divertor.

3. Ion flux towards divertor and in-out asymmetry
Drift flow ($n_i V_{drift}$) is dominant near separatrix ($r_{min} < 4$ mm). Pantisol SOL flow ($n_i V_i$) extends to the outer flux surfaces.

Poloidal ion flux: $n_i V_i = n_i (V_i + V_{drift})$, where $V_i = n_i \Phi_i$. $V_{drift} = E \times B$ parallel.

$E \times B$ drift constitutes net mass flow. $i = B_\theta / B_\phi < 0.1$, $\Phi_i = B_i / B_\phi > 1$.

Poloidal flux density at HFS SOL

4. Puff and pump effects on SOL flow

ExB drift flow in the private region produces HFS-enhanced asymmetry.

When LFS divertor is detached, $E \times B$ drift appears in the LFS divertor. $I_{FS}$ is enhanced due to large increase in parallel flux in outer flux surfaces.

Puff and Pump* increases parallel flux Friction force on impurity ions. Effects on SOL flow are distinguished at HFS and LFS, comparing Main (Puff and Pump) and Divertor puffing (for the same $r_s$). At HFS, Mach numbers increase (max. 25%) over wide SOL region.
5. Summary and Conclusions

Puff and Drift flows were investigated both at HFS and LFS SOLs experimentally, and numerically (for the ion VB drift towards the divertor):

- Reversal of Parallel SOL flow direction at the main plasma edge was consistent with UEDGE including drift effects (mostly due to ion VB drift).
- Subsonic flow \( \langle \delta \rangle \) in measurement was larger than simulation should be distinguished by improving simulation and Mach probe models.
- E x B drift flux in SOL reached 30-50% of the parallel flux.
- \( \beta \) x B slab flux in private flux region was comparable to parallel flux in SOL. It produces in-out asymmetry in the attached divertor.
- Parallel SOL flow was controlled by Puff and Pump:
  - dominant effects (flow velocity, density, friction force) appeared at HFS.
- Design work including drift effects will be important to optimize the divertor and pumping geometries relevant to a tokamak reactor.

This work contributed to JFA Divertor and SOL Physics, N. Asakura et al. CT/R01
1.11 Enhanced Pedestal Pressure and Parameter-Linkages Determining Edge Pedestal Structure in JT-60U Type I and Type II ELM My H-mode

Y. Kamada 1), H. Takenaga 1), H. Urano 2), T. Takizuka 1), T. Hatake 1), Y. Miura 1)
1) Japan Atomic Energy Research Institute, Naka Fusion Research Establishment, Ibaraki-ken, Japan
2) Max-Planck-Institut fur Plasmaphysik, Garching, Germany

e-mail contact of main author: kamada@naka.jaeri.go.jp

Abstract. Based on the expanded H-mode operational regimes in JT-60U utilizing the improved capability of high triangularity (δ) operation, the multiple pellet injection and high power heating including the negative ion based NB (NNB), we have clarified the pedestal parameter-linkages determining the pedestal structure (i.e. the pedestal width) increases with $\rho_p$ and does not depend on $\beta_p$ [16]. The normalized pressure gradient $\alpha'$ is almost constant at low $\delta$, while $\alpha \propto \beta_p$ at high $\delta$. Based on these results, we have enhanced the pedestal pressure of the ELMy H-mode by factors of 2-2.5 at the same plasma current and plasma shape, and extended the high confinement regime to a high density. We have found that the pedestal stored energy $W_{ped}$ increases with the core energy (or $\beta_c$-core) at high $\delta$ (>0.3-0.4), while $W_{ped}$ is low and almost constant independent of, for example, heating power at low $\delta$ (<0.2). In addition, we have expanded the type II (grassy) ELMy high confinement regime with a small heat load on to divertor plates to the low-q ($q_{95}$<4) regime, and demonstrated successful compatibility of the type II ELMs with the pellet injection. Based on a variety of JT-60U experiments, possible linkages among the pedestal and the core parameters has been proposed.

1. Introduction

The H-mode edge pedestal condition determines the burning plasma performances, such as the fusion gain, since it determines the core confinement as the boundary condition and affects the stable $\beta$ limit through the global current and pressure profiles. Therefore, study on the pedestal structure and its response to external controls has been raised as one of the urgent research subjects in the world tokamak research activities [1-4]. The pedestal structure [5] is determined by 1) formation and dynamics of the transport barrier [6-8] governed by transport bifurcations [9,10] and 2) by appearance of the edge localized modes (ELMs) [11,12]. Based on these physics understandings, we need a global treatment of transport [9,13] and stability [12] characteristics including core, pedestal, and SOL regions in order to expand the H-mode operational regimes [14] towards the high integrated fusion performance [15] in ITER and DEMO reactors. From this point of view, JT-60U has devoted large efforts on understandings of the edge pedestal and expanded the operational regimes of the ELMy H-mode [16]. This paper reports recent results of JT-60U contributing to the two urgent research subjects: extension of the high confinement regime to high density, and expansion of the small (Type II) ELM regime towards low $q_{95}$. In addition, based on a variety of JT-60U experiments, we propose possible linkages among the pedestal and the core parameters.

2. Density Regime Extended by Enhanced Pedestal Pressure

We applied multiple pellet injection into the high-$\beta_p$ ELMy H-mode discharges and the high confinement regime was extended to $n_e/n_{cw}$ ~0.7 (Fig.1(a)) [17]. In these cases, the pellets penetrated just inside the pedestal width $\Delta_{ped}$ and the electron density $n_e$ was increased gradually (~20$\tau_e$) so as not to decrease the pedestal temperature (since $\Delta_{ped}$ increases with the thermal ion poloidal gyro radius $\rho_{pi}$ as described below). In addition, we increased $\beta_p$ above 2 with an optimum heating profile consisting of the positive and negative ion source NBs to keep MHD stability. Consequently, we have enhanced $W_{ped}$ by factors of 2-2.5 for the type I ELMy edge at the same plasma current and plasma shape ($I_p$=1MA, $\delta$=0.44-0.5; Figs.1(b),(c) and 2), and achieved $H_{95}=2.1$ ($H_{92}=1.1$) at $n_e=0.7n_{cw}$. At the same density, $H_{95}$ was 1.3 in the gas-fuelled reference cases. Figure 1(b) and (c) compares time evolution of the representative discharges with pellet injection and gas fueling, respectively. Figure 2(a) shows that the pedestal ion temperature is high (by a factor of 2.5) and $\Delta_{ped}$ is wide in the pellet injection case compared with gas fueling at the same pedestal density. Figure 2(b) (treating discharges at 1MA and $\delta$=0.44-0.50) shows that the pedestal pressure ($P_e^{ped}=n_e^{ped}T_e^{ped}$) stays roughly constant for the standard ELMy H-mode with type I ELMs (open circles). While in the high $\beta_p$ ELMy H-mode
FIG. 1: (a) $H_{90PL}$ vs. $n_e/n_{GW}$ for low $\delta$ (0.15 - 0.2) ELMMy H-mode, high $\delta$ (0.45 - 0.55) H-mode and high $\delta$ (0.45 - 0.55) high-$\beta_p$ ELMMy H-mode. The multiple pellet injection extended the density range with $H_{90PL} > 2$ up to 0.7 $n_{GW}$. (b&c) Time evolution of the two discharges at $I_p = 1$ MA and 0.7 $n_{GW}$ shown in (a) by black circles; one (b: E37413) with pellet injection and the other (c: E32398) with gas puffing.

(closed circles), $p_e^{PED}$ can be higher. In the pellet injected cases, $p_e^{PED}$ increases gradually (see the time evolution of E37413), and reaches high values. On the other hand, in the gas-fueled case (E32398), $T_e^{PED}$ decreases with increasing $n_e^{PED}$. The pedestal temperature in E37413 is higher than that in E32398 by a factor of 2.3. We have also achieved high pedestal pressure with the type II ELMs (crosses in Fig.2(b)). Such type II ELMs were obtained without pellet injection and in the relatively high $T_e^{PED}$ regime as shown in Fig.2(b).

3. Pedestal Structure

Figure 3 shows dependence of the pedestal parameters on the total $\beta_e$ values ($\beta_e$-tot). Figure 3(a) shows the pedestal $\beta_e$ ($\beta_e$-ped) increases with $\beta_e$-tot at high $\delta$ -0.44 -0.50. This relationship appears independent of existence of the ITB, which means that this relation does not come from the profile stiffness. On the other hand, $\beta_e$-ped is almost constant at low $\delta$.

The pedestal pressure is sustained by both the width $\Delta r_{ped}$ and the gradient $V_p$. Figure 3(b) shows that $\Delta r_{ped}$ is independent of $\beta_e$-tot (and also $\beta_e$-ped, since $\beta_e$-tot = $\beta_e$-ped as shown in Fig.3(a)). Figure 3(c) shows that the pedestal width follows the scaling $\Delta r_{ped} \sim 5 \rho_{i0}^{0.3}$ [18]. Previously, $\beta_e$-ped and $p_{ei}$ had a strong correlation experimentally. However, recently, the pellet injection and the NNB injection

FIG. 2: (a) Ion temperature profiles for pellet and gas fueled type I ELMMy H-mode discharges at the same $n_e^{PED}$ ($I_p = 1$ MA, $\delta = 0.46$). (b) Pedestal electron temperature $T_e^{PED}$ vs. density $n_e^{PED}$ at $I_p = 1$ MA and $\delta = 0.44 - 0.50$. 

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enabled these two parameters to be decoupled.

Figure 3(d) shows that, at high δ (circles), the normalized pressure gradient, the α-parameter, increases with βP-tot. On the other hand, at low δ (squares), it is almost constant at a low value. This βP-dependence may be due to increasing Shafranov shift [19] or radially increasing filed line pitch at the low field side.

Figure 4 shows time evolution of the high-δ high βP type I ELMy H-mode discharge E32358. In this discharge, βP-ped increases with βP-core, and after saturation of βP-core, βP-ped increases gradually with decreasing I, with a slow time constant of ~2 sec (~10τE) which is comparable with the edge current diffusion time over the pedestal layer.

In Fig.3, the type II ELM data follow the similar dependence. Based on these observation, the pressure gradient of the type I and type II ELMy edge is determined by δ, βP-tot (or βP-core) and the edge current driven mainly by the bootstrap current.

4. Pedestal Stored Energy and Parameter-Linkages

The dependence of βP-ped on βP-tot was different between low and high δ discharges as shown in Fig.3(a). In order to clarify the δ-dependence of the pedestal stored energy WPED, Fig.5 shows WPED normalized by the pedestal term of the offset-nonlinear scaling proposed in ref.20. (WPED-κRAF,B) at a fixed q95 = (3.0 – 3.8). We have found that upper boundary of WPED increases with δ (Fig.5(a)), and that WPED increases with the core energy at high δ (Fig.5(b)). In the previous scalings of WPED [3,20,21], WPED for the type I ELMy H-mode was expressed as Pped(Wped/V) = Bf(shape), where V is the plasma volume, B is the magnetic field (combination of B, and Ip/a), and f is the function of the plasma shape.

FIG.5: Pedestal stored energy normalized to the pedestal term of the Offset-nonlinear scaling: dependence on triangularity and βP-core.

FIG.6: Comparison of the profiles of electron density n_e, ion temperature T_i, heat flux Qheat and effective heat diffusivity Xeff at two different heating power for (a) the low δ (=0.16) and (b) the high δ (=0.45) configurations.
FIG. 7: Possible correlations (feedback loops) among pedestal and core parameters: Loop (1); Improved pedestal stability (high $\delta$, high $\beta_p$) magnetic shear) steepens pedestal pressure gradient and enhances pedestal pressure ($p_{\text{PED}}$). High $p_{\text{PED}}$ allows high pedestal temperature ($T_{\text{PED}}$). High $T_{\text{PED}}$ improves core confinement when core kinetic profile shapes are stiff. High core confinement increases $\beta_p$. Loop (2); High $T_{\text{PED}}$ widens pedestal width and then enhances $p_{\text{PED}}$. Loop (3); Pedestal pressure gradient drives bootstrap current and affects edge magnetic shear.

It should be noted that this expression is independent of the heating power. This power-dependence holds at low $\delta$ ($<0.2$) in JT-60U. (As shown in Fig.3(a), $\beta_p$-$\text{ped}$ is almost constant and independent of $\beta_p$-core). On the other hand, at high-$\delta$, $W_{\text{ped}}$ increases with $\beta_p$-core. This result means that $W_{\text{ped}}$ increases with heating power at high-$\delta$. Accordingly, we conclude that the pedestal term of the confinement scaling law should be a non linear function on $\delta$ depending on $\beta_p(\delta)$.

The degradation of the H-mode confinement at high density is explained that $T_{\text{PED}}$ decreases with increasing $n_{\text{e, PED}}$ because $p_{\text{PED}}$ is constant, and temperature in the core region decreases in proportion to decreasing $T_{\text{PED}}$ due to appearance of the profile stiffness [22]. Figure 6 shows that the scale length of $T_i$ is almost constant and independent of the heating power for both low-$\delta$ (Fig.6(a)) and high-$\delta$ (Fig.6(b))[23]. However, at high-$\delta$, $T_{\text{PED}}$ becomes high at high heating power $P_{\text{he}}$, while $T_{\text{PED}}$ is almost unchanged at low-$\delta$. Consequently, $\chi_{\text{el}}$ in the core region increases largely with increasing $P_{\text{he}}$ at low-$\delta$.

Figure 7 summarizes possible correlations among pedestal and core parameters based on the observations in JT-60U. In order to achieve a steep pedestal pressure gradient, high $\delta$, high $\beta_p$ and edge magnetic shear control are required [24-27]. Effects of the edge magnetic shear on the edge turbulence suppression (thus on the pedestal width) [28] has not been clarified in JT-60U. The steep $Vp$ enhances pedestal pressure. The high pedestal pressure allows a high pedestal temperature at a given pedestal density. The high pedestal temperature widens the pedestal width ($r_p$, dependence [18,29]). The wide

FIG. 8 (a) Three cases with different behavior of ELMs in pellet fuelled plasmas: top two columns: pure grassy ELMy at $q_{95}=3.8$ and $\delta=0.58$, middle two columns: mixture of grassy (type II) and giant (type I) ELMs at slightly lower $q_{95} =3.7$ and $\delta=0.56$, bottom two columns: giant ELMy at $q_{95} =3.6$ and $\delta=0.45$. (b) Grassly ELMy and giant ELMy regimes on the $\delta$ - $q_{95}$ plane. (c) Typical equilibrium with $\delta=0.6$. 

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pedestal width enhances the pedestal pressure and the pedestal temperature. High pedestal temperature improves the core confinement for the standard ELMy H-mode [22,23,30]. It has not been clarified that the high edge temperature helps the ITB formation. However, at least for the high \( \beta_p \) mode, high-\( \delta \) plasmas seem to have relatively lower threshold heating power for ITB formation with a clear electron temperature internal barrier [31]. And then, if the core confinement (or \( \beta_p \)) is improved, the pedestal stability is improved. According to Fig.4 and ref.[18], the time constant required for this positive feedback cycle (loop '1' in Fig.7) seems to be \( \sim 2 \text{sec} (10 \tau_\phi) \) at \( I_p=1\text{MA} \). Therefore, when we increase density, we need to fit the rise time to this time scale. In practice, in the pellet injected discharge E37413 (Figs.1-3), the slow density rise over 3sec was successful. On the other hand, a strong gas puff decreases the pedestal temperature which may force the plasma to follow the negative feedback loop.

5. Extension of the type II ELM regime to \( q_{\text{eff}} \leq 4 \)

The type II ELM appears at high-\( \delta \) and high-\( q_{\text{eff}} \) [17,24,25,32]. The high-\( \delta \) operation capability extended recently in JT-60U enables \( \delta=0.6 \) at \( I_p=1\text{MA} \). With \( \delta=0.58 \), we have sustained the type II ELMy H-mode at \( q_{\text{eff}}=3.8 \) (Fig.8), and expanded the operational regime having high confinement with an ITB (\( H_{\text{95}}>1.1 \)), high \( \beta_p>2.8 \) and small peak heat load (\( \sim 1/5 \) of the type I ELMs) to this low-\( q_{\text{eff}} \) regime. In addition, we demonstrated favorable compatibility of the type II ELMs with pellet injection (Fig.8(a)) when the pellet penetration is deeper than the pedestal width. At a smaller \( \delta \) or \( q_{\text{eff}} \), type I ELMs appear after each pellet. At a medium \( \delta \) (\( \sim 0.45 \)), type I ELMs govern the discharges.

Acknowledgments
The authors would like to thank the JT-60 Team and the members of the ITPA Edge & Pedestal Group.

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Enhanced Pedestal Pressure and Parameter-Linkages Determining Edge Pedestal Structure in JT-60U Type I and Type II ELM My H-mode

Y. Kamada 1), H. Takenaga 1), H. Urano 2), T. Takizuka 1), T. Hatae 1), Y. Miura 1)

1) Japan Atomic Energy Research Institute, Naka Fusion Research Establishment, Ibaraki-ken, Japan
2) Max-Planck-Institut fur Plasmaphysik, Garching, Germany

E-mail contact of main author: kamada@naka.jaeri.go.jp

--- Abstract ---

Based on the expanded H-mode operational regimes in JT-60U utilizing the extended capability of high triangularity (δ) operation, the multiple pellet injection and high power heating including the negative ion based NB (NNB), we have clarified the pedestal parameter-linkages determining the pedestal structure:

'Pedestal width' increases with \( \rho_p \) and does not depend on \( \beta_p \).

'The normalized pressure gradient \( \alpha \)' is almost constant at low \( \delta \) and \( \alpha = \beta_p \) at high \( \delta \).

Based on these results, we have enhanced the pedestal pressure of the ELM My H-mode by factors of 2-2.5 at the same plasma current and plasma shape, and extended the high confinement regime to a high density.

We have found that the pedestal stored energy \( W_{ped} \) increases with the core energy (or \( \beta_p \)-core) at high \( \delta (>0.3-0.4) \), while \( W_{ped} \) is low and almost constant independent of, for example, heating power at low \( \delta (<0.3) \).

We have expanded the type II (grassy) ELM My high confinement regime with a small heat load on to divertor plates to the low-\( \alpha \) (\( \alpha_{\psi<4} \)) regime, and demonstrated successful compatibility of the type II ELMs with the pellet injection.

Based on a variety of JT-60U experiments, possible linkages among the pedestal and the core parameters has been proposed.

--- Introduction ---

JT-60U Objectives / Strategy

- ITER Physics R&D
- Advanced Tokamak Concepts for ITER & DEMO

Integration of the key elements

\[ \begin{align*}
\bar{n}_e & \quad n_{W} \\
\rho_N & \quad n_{rad} \\
f_{BS} & \quad P_{rad} \\
f_{CD} & \quad P_{heat}
\end{align*} \]

C: Heat confinement as B.C. \( \rho \)-limit through \( \mu(r) \) heat/particle pulse to Div.

Long Sustained Exposure to reactor relevant regime

- small \( \nu_{e}^* \)& \( \rho_{p}^* \)
- \( T_e - T_i \)
- Repeated ELMs.

--- Approach ---

Pedestal Pressure
Pedestal Width
Pedestal grad-p
Pedestal Core
Core -> Pedestal ELM types

ELM -> SOL -> Div.

Linkages among Pedestal and Core
Parameters incl. time constant
Controlability

response to external controls
Shape, Heating, Fueling...
high integrated performance
Extended density region by Pellet + high $\delta$

High confinement regime ($H_{89PL}=2.1$, $H_{89}=1.1$) has been extended to $n_e/n_{GW} \approx 0.7$. ($I_p=1$MA, $\delta\approx 0.44-0.5$)

Pedestal Temperature: Pellet > Puff

$E_37413$ $E_32398$

Pellet gas puff
$t=6.5s$ $t=9.4s$

Pedestal Pressure: Enhanced by $x 2-2.5$ at the same $I_p$ & shape

Type I: High $\beta_p$, ELM H-mode > Standard ELM H-mode
Type II: Higher T-ped regime

Confinement: Pellet > Puff

The pellets penetrate just inside the pedestal width $\delta_{GW}$ and the electron density $n_e$ was increased gradually ($-20\%$) so as not to decrease the pedestal temperature (since $\delta_{GW}$ decreases with $n_e$). In addition, we increased $P_e$ above $2$ with an optimum heating profile, sustaining the positive and negative ion source NBs to keep MHD stability.
At high-δ, pedestal $\beta_p$ & $\alpha$ increases with $\beta_p$-total
At low-δ, pedestal $\alpha \sim$ const

Pedestal Stored Energy: $\sim$ const at low-δ
$= f(\delta, \beta_p(\delta))$ at high-δ

Previous scalings of $W_{ped}$ (type I ELMMy): $W_{ped} \sim kR_{L_i}\beta_i$ (independent of the Pheet)
We found that
- upper boundary of $W_{ped}$ increases with $\delta$,
- $W_{ped}$ increases with the core energy at high $\delta$,
- while $W_{ped} \sim$ const. at a low value independent of, for example, heating power at low $\delta$ ($\leq 0.2$).

Shafarevich shift or radially increasing field line pitch at the low field side?

High Triangularity: Gradual Pedestal Evolution
$T_{ped}$ & $T_e^ped$ and $\Delta_{ped}$ increase gradually ($\sim 2s - 10s$).
$\beta_p$-ped increases with increasing $\beta_p$-tot
& decreases with decreasing $I_i$

Low triangularity
$T_{ped}$ becomes high at high $P_{abs}$

High triangularity
Core stiffness & confinement degradation (reminder)

low triangularity:
Degradation of HH at high ne → Core degradation
Pedestal Pressure (nT) – const for type I ELM
Core HH decreases with decreasing T-ped
T(\theta), T(\phi) - shapes are stiff.

high triangularity:
Pedestal pressure is enhanced.
Core LT: low-\beta → high-\beta

Comparison with the multi-machine scaling laws

JT-60U: Parameter Linkages

P_{pl} → High T-ped
High P-ped

1. wide pedestal width

2. steep grad-p

3. high Np

low-recycling

High Triangularity

JT-60U
New high $\delta$ (0.6) configuration expands grassy ELMy regime

Grassy ELM was achieve at $q_{95} = 3.8$ at $\delta = 0.6$ & compatible with pellet (inj. depth $\sim \Delta_{\text{ped}}$)

Summary

- HFS Pellet inj. into high triangularity high $\beta_p$ ELMy H-mode extended the density regime with high confinement.
- Pedestal Pressure was enhanced by $> 2$ at fixed $\ell_p$ & shape
- At high-$\delta$, pedestal $\beta_p$ & $\alpha$ increases with $\beta_p$ total
  - At low-$\delta$, pedestal $\alpha$ $\sim$ const
  - both giant & grassy ELMy pedestal width $\Delta_{\text{ped}}$ $\sim \beta_p q_{95}^{-0.3}$ (not $\sim \beta_p$ or $\beta_p$-ped)
- Pedestal Stored Energy: $\sim$ const at low-$\delta$
  - $\sim f (\delta, \beta_p(\delta))$ at high-$\delta$
- Possible linkage among pedestal & core parameters was proposed.
- New high $\delta$ (0.6) config. expanded grassy ELMy regime ($q_{95} > 3.8$).
  - & compatible with pellet (inj. depth $\sim \Delta_{\text{ped}}$)
- Full-CD in Grassly ELMy Advanced Operation was demonstrated.

Pedestal Width: on the $\rho_{pl} q_{95}^{-0.3}$ scaling
1.12

Observation of High Recycling Steady H-mode Edge and Compatibility with Improved Core Confinement Mode on JFT-2M

Naka Fusion Research Establishment, Japan Atomic Energy Research Institute
E-mail: kamiya@axjft3.tokai.jaeri.go.jp

Abstract. A new operational regime has been discovered on JFT-2M under the boronized first wall condition to produce High Recycling Steady (HRS) H-mode, which is characterized by good energy confinement ($H_{\text{exp}}$-1.6) at high density around 70% of the Greenwald density ($n_e/n_{\text{GW}}$), low radiated power fraction, and the complete disappearance of large (Giant) ELMs. Accompanying the HRS H-mode, a coherent magnetic fluctuation is observed at around 50-150kHz, whose characteristics are similar to the Enhanced D$_e$ (EDA) H-mode reported from Alcator C-Mod. The H$^+$-mode, previously observed on JFT-2M, had common features with EDA-mode in terms of the coherent density fluctuation, but it appeared transiently. The HRS operating regime is also similar to EDA-mode, except that HRS is seen even at low $q_{95}$ ($2 < q_{95} \leq 3$). The most important feature of HRS H-mode edge condition is the compatibility with an improved core confinement mode at high density without large ELMs. We have demonstrated that an internal transport barrier (ITB) can be produced with the HRS H-mode edge condition, achieving $\beta_p H_{\text{exp}}$-6.2 at the $n_e/n_{\text{GW}} \sim 70\%$, transiently.

1. Introduction

Recent experiment in many tokamaks, including ASDEX Upgrade, DIII-D, JET, and JT-60U, have concentrated on advanced scenarios, such as H-mode with type II and grassy ELMs, QH-mode, etc [1-4]. These discharges show a strong reduction of the ELM activity, eliminating pulsed heat loads on the divertor target, and having substantial potential for higher performance by combination of an internal transport barrier (ITB) plus steady H-mode edge. However, it is not well understood which types of advanced H-mode edge condition is more favorable for next-step devices. Also the exact relationship among these different regimes is presently unclear. In the JFT-2M tokamak (major radius $R=1.31$ m, minor radius $a = 0.35$ m, elongation $\kappa \leq 1.7$) [5], an attractive new High Recycling Steady (HRS) operating regime has been discovered after boronization of the first wall of the vacuum vessel. This new regime has following important features, (1) the steady-state H-mode edge condition at high density with good energy confinement, (2) the complete disappearance of Giant ELMs, and (3) the compatibility with ITB.

2. Discovery of High Recycling Steady (HRS) H-mode edge

On JFT-2M, the HRS H-mode edge condition was obtained with co-, counter-, and balance-NBI heating under the strong wall fueling from the boronized first wall and the saturated pumping capability with deuterium gas. Figure 1 (a) and (b) show time history of the co-NBI heating plasmas with $P_{\text{in}} \sim 0.7$ MW, comparing between before and after boronization at identical experimental conditions ($I_p = 0.2$ MA, $B_T = 1.3$ T, and $q_{95} \sim 3.1$). The plasma configuration for these experiments has a standard shape for JFT-2M, with triangularity ($\delta$) $\sim 0.4$ and elongation ($\kappa$) $\sim 1.4$. Before boronization (#97009), the plasma made a transition into the standard ELM-free H-mode triggered by a sawtooth crash at $\sim 656$ ms as seen by a sharp drop of the $D_e$ signal and increase in the electron density, stored energy, and especially in edge soft X-ray (SXRF) intensity passing through just inside separatrix. The radiation loss power ($P_{\text{rad}}$) and core SXRF intensity passing through the magnetic axis are also increasing during ELM-free period, showing impurity concentration to the plasma core. And at the onset of second Giant ELM, the plasma makes a back-transition into L-mode. On the other hand, in
the case after boronization (#97156), the plasma behavior exhibits a considerable change from pre-boronization, especially in the impurity, wall pumping and fueling (i.e., recycling rate). As shown in FIG. 1 (b), the radiation loss power is about 1/3 as large as #97009 during Ohmic heating period, resulting in a significant reduction of the threshold power for L/H transition ($P_{th}$) to about a half of pre-boronization ($P_{th}$ at post-boronization ~ 0.4 MW). In this case, although the pumping capability of boronized first wall is saturated with deuterium gas, the plasma makes a transition into the standard ELM-free H-mode at 10 ms after additional heating is applied, as seen by sharp drop in the $D_0$ light and at the same time, both electron density and stored energy begin to rise. But at ~ 625 ms, the plasma makes a second transition into a HRS H-mode, indicating a rise in the $D_0$ signal, which is the source of the name for this operating regime. In contrast to the ELM-free case, the density reaches a new plateau value, and the radiation loss power also stops increasing. The stored energy keeps constant value of ~ 23 kJ for about 7 time global energy confinement time ($\tau_E$ ~ 27 ms), corresponding to the confinement enhancement factor, $H_{95}$ ~ 1.5.

![FIG. 1. Time history of the co-NBI heating plasmas with $P_{NB}$ ~ 0.7 MW, comparing between (a) before and (b) after boronization at identical experimental conditions ($I_p$ = 0.2 MA, $B_T$ = 1.3 T, and $q_{95}$ ~ 3.1).](image)

Figure 2 (a) and (b) show the spectral time history of the magnetic fluctuation (dB/dt) measured by the magnetic probe at the outer midplane. It is noted that two types of small ELMs are seen on top of the enhanced $D_0$ signal, (a) grassy-like and (b) dithering ELMs. As shown in FIG. 2 (a), the coherent mode appears at around ~ 150 kHz after a brief ELM-free period, coincident with the increase in the $D_0$ signal and decrease in dB/dt, indicating enhancement in particle transport. It is considered that HRS H-mode may be associated with the coherent fluctuation, which is similar to EDA-mode observed on Alcator C-Mod [6]. The H'-mode, previously observed on JFT-2M [7, 8], had common features with EDA-mode in terms of the coherent density fluctuation, but it appeared transiently. In this study, the coherent magnetic fluctuations with various frequencies between 50-200kHz have been observed in the different experimental conditions. As shown in FIG. 2 (b), the coherent mode is also seen at around 100kHz. However, its spectrum seems to be made somewhat confusing by the dithering ELMs. The relation among ELMs, coherent-mode, and confinement will be discussed later in this paper.
3. HRS H-mode operational space

Even after boronization, the ELMy/ELM-free H-mode is also obtained when the pumping capability of boronized first wall is not saturated. To investigate the condition under which either ELMy or HRS H-modes are obtained, a series of experiments was performed scanning $I_p$, $B_T$, (i.e. $q_{95}$) at fixed triangularity ($\delta$) $\sim 0.4$ and $P_{NB} \sim 1.4$ MW (balance injection). Most ELMy/ELM-free H-modes are clearly classified as one or the other. But an ambiguity exists near the operational boundary between HRS and other small ELMy regime, namely, A “Mixture” regime. Considering the disappearance of Giant ELMs is one of important features in HRS H-mode, we chose to use the amplitude of ELMs ($I_{ELM}$) normalized by enhanced D$_x$ level during H-mode ($I_{ew}$) as the best indicator (FIG. 3 (a)). Figure 3 (b) and (c) show the results of $I_p$ and $B_T$ scan. It has been found that HRS H-mode is observed widely at $q_{95}$-$I_p$ and $q_{95}$-$B_T$ space, although at higher $q_{95}$ (>3.7), no large ELM (i.e. ELMy regime) appears. On the contrary, considerable overlap with the ELMy regime is seen at lower $q_{95}$ (<3.7), except for higher $I_p$ region at $\sim$300kA. Figure 3 (d), (e), and (f) show the operational space of HRS H-mode in relative with the line averaged electron density ($n_e$), Greenwald fraction ($n_e/n_{GW}$), and neutral pressure during $q_{95}$ scan described above. It has been found that HRS and ELMy regimes can be separated by $n_e/n_{GW}$ at low $q_{95}$ region, while considerable overlap exists in $q_{95}$ vs. $n_e$ space. Also the mixture regime is seen at HRS/ELMy boundary at $n_e/n_{GW} \sim 0.35-0.45$. In addition, two types of HRS regimes are separated by neutral pressure, indicating that the dithering ELMs are more dominant at higher neutral pressure. As shown in FIG. 3 (g) and (h), it is found that the $H_{app}$ is slightly degraded as the $n_e/n_{GW}$ increases. It is noted that the $H_{app}$ value is comparable between ELMy and HRS at same $n_e/n_{GW} \sim 0.4$. But, at higher $n_e/n_{GW}$ around 0.5-0.7, the $H_{app}$ in the HRS regime with dithering ELMs is systematically lower than the HRS regime with grassy ELMs. Returning to FIG. 2, it is considered that the degradation in the $H_{app}$ may be connected to the large magnetic fluctuation, which appears slowly after the L/H transition. But the exact causality is still unclear due to lack of data, such as pressure and current profiles at the pedestal. Further understanding is required for obtaining good confinement and steady H-mode without large ELMs.
FIG. 3. Results of global parameter scan to determine ELMy/HRS boundary.

4. Compatibility of HRS H-mode edge with Internal transport barrier

Recent experiments on JFT-2M, the discharges having ITB have been produced with steady H-mode edge condition. As shown in FIG. 4 (#97229, I_p = 0.15 MA and B_t = 1.0 T), the co-NBI with 0.7MW is applied as the pre-heating during I_t-ramp-up at 350 ms, and after the plasma current reaches flattop, the ctr-NBI with 0.7MW is added at 450 ms. During the pre-heating phase, the plasma makes a transition into H-mode at ~390 ms as seen by a drop in the D_n signal and an increase in the electron density. After a brief ELM-free period, the plasma makes a second transition into the HRS H-mode at ~ 425 ms as seen a rise in the D_n signal and decrease in d n/dt. During main-heating phase with balance-NBI of 1.4 MW, the plasma exhibits a bifurcation in the particle and energy confinement from the previous plasma state in the plasma core region, keeping the HRS H-mode edge condition, as seen in continuous increasing in the electron density, SXR (core), and β_n. It is suggested that these confinement improvement results from an ITB formation within the inner half radius (normalized ITB radius, ρ.ITB ~ 0.1-0.2), which is characterized by the peaked ion temperature profile. The IL-mode, previously observed on JFT-2M [9], has common features with this ITB, except for edge condition (L-mode edge in the previous case). The q-profile also changes from monotonic to zero/weak shear in the plasma core region. It has been found that the HRS H-mode edge is compatible with an improved core confinement mode. Just before collapse, the product β_N H_{95} ~ 6.2 (β_N ~ 3.1, H_{95} ~ 2.0) is achieved at n_e/n_{c95} ~ 0.7, transiently. It is believed that combination of an ITB plus steady H-mode edge has substantial potential for higher performance.
FIG. 4. Time history of ITB plus HRS H-mode discharge (left) and radial profiles of \( T \), and \( q \) (right).

5. Summary

An attractive new “High Recycling Steady” (HRS) operating regime has been discovered on JFT-2M after boronization, where a coherent magnetic fluctuation is associated with its steady-state characteristics. An operational regime is similar to EDA-mode, except that HRS is seen even at low \( q_{95} \) (2<\( q_{95} \)<3). The most important feature of the HRS H-mode edge condition is the compatibility with an improved core confinement mode at high density without large ELMs. We have demonstrated the product \( \beta_n H_{\text{app}} = 6.2 \) at the \( n_i/n_{\text{GW}} \sim 70\% \) in a combination of ITB plus HRS H-mode edge condition, transiently.

Acknowledgement

The authors are much indebted to Drs. T. Fujita, T. Fukuda, Y. Kusama, K. Shinohara, H. Takenaga, and T. Takizuka for helpful suggestions and discussions. We would like to thank Dr. M. Bakhtiar for providing the FFT analysis code of the magnetic probe data. We also thank Dr. K. Ida (NIFS) for his careful check of the charge exchange recombination spectroscopy (CXRS) data. Thanks are also due to Drs. A. Kitsunezaki, H. Ninomiya, and M. Kikuchi for continuous encouragement.

References

INTRODUCTION
Recent experiment in many tokamaks (AUG, C-Mod, DIII-D, JET, and JT-60U) have concentrated on advanced scenarios, (1) Strong reduction of the ELM activity, eliminating pulsed heat loads on the divertor target (2) ITB plus steady H-mode edge

Which types of advanced H-mode edge condition is more favorable for next-step devices? - the exact relationships among these different regimes is presently unclear

Attractive new "High Recycling Steady" (HRS) operating regime discovered after boronization of JFT-2M vacuum vessel:
• Steady-state H-mode edge at high density with good confinement
• Complete disappearance of giant ELMs
• ITB-compatible

ABSTRACT
A new operational regime has been discovered on JFT-2M under the boronized first wall condition to produce High Recycling Steady (HRS) H-mode, which is characterized by good energy confinement ($H_{E98}=1.6$) at high density around 70% of the Greenwald density ($n/n_{GW}$), low particle confinement, low radiated power fraction, and the complete disappearance of large ELMs (Giant). Accompanying the HRS H-mode, a coherent magnetic fluctuation is observed at around 50-150kHz, whose characteristics are similar to the Enhanced D$_{e}$ (EDA) H-mode reported from Alcator C-Mod. The H'-mode, previously observed on JFT-2M, had common features with EDA-mode in terms of the coherent density fluctuation, but it appeared transiently. HRS operating regimes are also similar to EDA-mode, except that HRS is seen even at low q ($2<q_{98}<3$). The most important feature of HRS H-mode edge condition is the compatibility with an improved core confinement mode at high density without large ELMs. We have demonstrated that an internal transport barrier (ITB) can be produced with HRS H-mode edge condition, achieving $\beta_{p,H_{E98}}\sim 6.2$ at the $n/n_{GW}\sim 70\%$, transiently.

Shot comparison between pre- and post-boronization at $I_{P}/B_{T}=0.2MA/1.3T$ ($\phi_{B}=3.1$)
• HRS H-mode edge condition was obtained with co-, ctr., and balance-NBI under the wall fueling from the boronized first wall ($\text{B}(\text{CH}_{3})_{3}$, tri-methyl-boron), the pumping capability of which was saturated with deuterium gas.

-1-

<table>
<thead>
<tr>
<th>[MW]</th>
<th>$F_{NB}$</th>
<th>$F_{rad}$</th>
</tr>
</thead>
<tbody>
<tr>
<td>[a.u.]</td>
<td>SXR</td>
<td>edge</td>
</tr>
<tr>
<td>$10^{53}$</td>
<td>$n_{e}$</td>
<td>$D_{e}$ [a.u.]</td>
</tr>
<tr>
<td>[kJ]</td>
<td>$W_{ASID}$</td>
<td>ELM-free/ELM</td>
</tr>
</tbody>
</table>

-2-
HRS H-mode is associated with the coherent fluctuation

- Coherent-mode appears at around 150kHz after a brief ELM-free period,
  - coincident with the increase in $D_\alpha$ signal and decrease in $dn_{t}/dt$
  - enhancement in particle transport

Similar to EDA-mode (C-Mod) and H'-mode (JFT-2M), though $H'$-mode appeared transiently (0/n)


Operational boundaries in parameter space

- HRS is observed widely at $q_{95}$-1T and $q_{95}$-B_T space, although
  - at high $q_{95}$ (>3.7), no Giant ELM (ELMg) appears
  - EDA plasmas (C-Mod) are more likely at low I_T ($q_{95}$>3.7)
  - at low $q_{95}$ (<3.7), considerable overlap with ELMg, except for
    high I_T region (~300kA) - other parameters must be included

Complete disappearance of Large ELMs (Giant) at HRS operational regimes

- It makes possible to eliminate transient heat loads on the divertor target
  - Standard ELM/ELM-free H-mode is also obtained even after boronization,
    although only at the pumping capability of boronized first wall is not saturated
  - Two types of ELM activities on top of enhanced $D_\alpha$ signal - grassy-HRC ELMs
    and dithering ELMs

HRS and ELMg regimes can be separated by $n_e/n_{GW}$ at low $q_{95}$

- Mixture exists at ELMg and HRS operational boundary at $n_e/n_{GW}$=0.35-0.45
- HRS(with dithering ELMg) regime is more dominant at very high neutral pressure
Energy confinement time enhancement factor ($H_{98p}$) is slightly degraded as $n_e/n_{GW}$ increases.

- Well known result in tokamaks, though...
  \[ H_{98p}^{ELMy} \approx H_{98p}^{HRS(gray)} \approx 1.6 \text{ at } n_e/n_{GW} \approx 0.4, \text{ extending to } \approx 0.7 \]
  \[ H_{98p}^{HRS(dith+1)} \approx 1.3 \text{ at } n_e/n_{GW} \approx 0.5-0.7 \]

**Relation among coherent mode, ELMs, and confinement in HRS (dither) regimes**

Magnetic fluctuations appear slowly during L/H transition. Degradation in $\tau_e$ at this HRS regimes (under very high neutral pressure) may be related to this MHD activity. Its spectrum seems to be made somewhat confusing by the dithering ELMs.

Further understanding and controlling are required for good confinement and steady H-mode without large ELMs.

**HRS H-mode edge condition is compatible with improved core confinement mode**

- $\beta_p H_{98p} = 6$ is achieved at $n_e/n_{GW} \approx 0.7$, transiently.

**Combination of an ITB plus HRS H-mode edge has substantial potential for higher performance**

- $P_{TH}$ for ITB formation may exist - depending on $B_p$, strongly.
- Operation at higher density ($n_e/n_{GW} \approx 1.0$) with good confinement ($\beta_p H_{98p} = 6$) makes possible by combination of ITB plus steady H-mode edge.
Summary and future directions

An attractive new "High Recycling Steady" (HRS) operating regime discovered after boronization:

- Steady-state H-mode edge at high density ($n_e/n_{ew}=0.7$) with $H_{tor}=1.6$
- Complete disappearance of Large ELMs - It makes possible to eliminate transient heat loads on the divertor target
- HRS is associated with a coherent fluctuations
- HRS and ELMy regimes are not only separated by $q_{95}$ (HRS is seen even at low $q$, $2<q_{95}<3$) but also $n_e/n_{ew}$ (0.35-0.45) and/or neutral pressure

HRS H-mode edge condition is compatible with improved core confinement mode, having substantial potential for higher performance

- $\beta_p, H_{tor}=6$ is achieved at $n_e/n_{ew}=0.7-1.0$, transiently

Priority for future is to investigate scaling HRS transition using pedestal parameters and control of HIB
1.13 Properties of Internal Transport Barrier Formation in JT-60U

Y. Sakamoto 1), T. Suzuki 1), S. Ide 1), Y. Koide 1), H. Takenaga 1), Y. Kamada 1), T. Fujita 1), T. Fukuda 1), T. Takizuka 1), H. Shirai 1), N. Oyama 1), Y. Miura 1), K. W. Hill 2), G. Rewoldt 2) and the JT-60 Team 1)

1) Naka Fusion Research Establishment, Japan Atomic Energy Research Institute, 801-1 Mukouyama, Naka-machi, Naka-gun, Ibaraki-ken, 311-0193, Japan
2) Princeton Plasma Physics Laboratory, Princeton, New Jersey, U.S.A

e-mail contact of main author: sakamoty@fusion.naka.jaeri.go.jp

Abstract. The dependence of the ion thermal diffusivity ($\chi_i$) on the radial electric field ($E_r$) shear has been investigated in JT-60U plasmas. In positive magnetic shear (PS) plasmas, $\chi_i$ in the core region generally increases with the heating power, similar to the L mode at low heating power. However, as a result of the intensive central heating, which is relevant to the enhancement of the $E_r$ shear, a weak internal transport barrier (ITB) is formed, and $\chi_i$ in the core region starts to decrease. Corresponding to a further increase of the heating power, a strong ITB is formed and $\chi_i$ is reduced substantially. In the case of reversed magnetic shear (RS) plasmas, on the other hand, no power degradation of $\chi_i$ is observed in any of the heating regimes. The electron thermal diffusivity ($\chi_e$) is strongly correlated with $\chi_i$ in PS and RS plasmas. There exists a threshold in the effective $E_r$ shear to change the state from a weak to a strongly ITB. It is found that the threshold of the effective $E_r$ shear in the case of a PS plasma depends on the poloidal magnetic field at the ITB. There are multiple levels of reduced transport in the strong ITB for RS plasmas.

1. Introduction

The ITBs observed in JT-60U [1] can be categorized into two groups, i.e., weak and strong ITBs [2]. The weak ITB has lower diffusivity in the wide core region, compared to L-mode, while the strong ITB exhibits a large reduction in thermal diffusivity in a narrow layer. It is known that the strong ITB often develops into a so-called box-type ITB in RS plasmas. The threshold power ($P_{th}$) for the ITB formation is one of the critical issues for the application of ITBs to ITER. It is important to investigate the dependence of diffusivity ($\chi$) on the heat flux in the weak and strong ITB plasmas. On the other hand, in order to maintain high stability and high confinement, ITBs have to be actively controlled. Active control of the ITB quality has been demonstrated in JT-60U by changing the toroidal momentum input or heating power, the implication of which was that $E_r$ shear plays an important role [3]. Since the $E_r$ shear is one of the key factors, the dependence of $\chi$ on $E_r$ shear is required to develop the ITB control [4]. This type of study leads to a deeper understanding of the mechanism of ITB formation and also helps clarify the source of anomalous transport. To clarify these points, the dependence of $\chi$ on heating power and $E_r$ shear has been studied in PS and RS plasmas.

2. Dependence of diffusivity on heat flux in PS and RS plasmas

The threshold power for strong ITB formation was investigated in PS plasmas with the local magnetic shear (s) of around unity [5]. In order to investigate the properties of the ITB formation, including those of weak ITBs, the power of perpendicularly injected neutral beams ($P_{ne}$) was scanned in a detailed manner for the PS (s<1) and RS plasmas at fixed plasma parameters ($B_t=3.7T$, $I_p=1.3MA$, target line averaged $n_e=1.0x10^{19} m^{-3}$, triangularity of about 0.2, balanced toroidal momentum input). Figure 1 shows the strong ITB formation in a PS plasma. Changes in $T_e$ and $T_i$ profiles were observed simultaneously, accompanying the bipolar transition, which indicated a rapid local drop in $\chi$, and then the strong ITBs were formed. However, $\chi$ at the ITB is three times as high as the neoclassical level. Figure 2 shows the weak ITB formation in a PS plasma. With increasing heating power, $T_e$ and $T_i$ increased gradually in each radial location, and then the weak ITB appeared in the temperature profiles.
FIG. 1. The strong ITB formation in a PS plasma. (a) Waveform of the injected NB power. Time evolution of (b) $T_1$ and (c) $T_e$ in each radial location. Profiles of (d) $T_1$ (closed circles) and $T_e$ (closed squares), (e) $\chi_i$ (solid curve), $\chi_i$ (dotted curve) and $\chi_i^{\text{neq}}$ (bold curve) at $t=8.15$sec. The strong ITB appeared with a clear transition.

FIG. 2. The weak ITB formation in a PS plasma. (a) Waveform of the injected NB power. Time evolution of (b) $T_1$ and (c) $T_e$ in each radial location. Profiles of (d) $T_1$ (closed circles) and $T_e$ (closed squares), (e) $\chi_i$ (solid curve), $\chi_i$ (dotted curve) and $\chi_i^{\text{neq}}$ (bold curve) at $t=7.08$sec. The weak ITB appeared without a clear transition.

without a clear transition. The reduced transport level of the weak ITB is higher by a factor of 10 than the neoclassical level.

The dependence of $\chi$ on heat flux in PS plasmas is shown in Fig. 3. Below an absorbed $P_{\text{N}}$ ($P_{\text{N}}^{\text{abs}}$) of 2MW, the increment of the ion temperature gradient ($-\nabla T_i$) at $r/a=0.46$ is very small with increasing ion heat flux ($Q_i$) divided by ion density ($n_i$), as shown in Fig. 3(c), and $\chi_i$ increases with heating power. This indicates the L-mode transport without an ITB. Indeed, the confinement enhancement factor over the L-mode scaling in this case was around unity. In the range of $P_{\text{N}}^{\text{abs}}=3-5$MW, the increasing rate of $Q_i/n_i$ against $-\nabla T_i$ is slightly reduced, which is indicative of the weak ITB formation. In the case of $P_{\text{N}}^{\text{abs}}=8$MW, further reduction of $\chi_i$ is observed at the formation of the strong ITB. The relation between $Q_i/n_i$ and $-\nabla T_i$ at the strong ITB formation is interpreted as a bifurcation in transport. On the other hand, the increasing rate of $Q_i/n_i$ in the outer region ($r/a=0.7$) stays large in the L-mode state. It should be noted that the dependence of $\chi_i$ on heat flux is similar to that of $\chi_e$, as shown in Fig. 3(b), suggesting a strong correlation between electron and ion transport. Figure 4 shows the dependence of $\chi_i$...

FIG. 3. Dependence of $\chi$ on heat flux in PS plasmas. (a) Comparison of $T_1$ profiles at each $P_{\text{N}}^{\text{abs}}$ (b) $\chi_i$ and $\chi_e$ as a function of $P_{\text{N}}^{\text{abs}}$. With increasing heating power, the transport properties change from no ITB to the weak ITB and to the strong ITB. (c) Relation between ion heat flux divided by ion density and ion temperature gradient indicates a bifurcation in transport.
on heat flux in RS plasmas. Below a $P_{\text{NB}}^{\text{abs}}$ of $\sim$5MW, $\chi_{t}$ stays constant with increasing heating power. In the range of $P_{\text{NB}}^{\text{abs}}$>7.5MW, $\chi_{t}$ starts to decrease when the strong ITB is formed. It is noteworthy that no power degradation in $\chi_{t}$ is observed for RS plasmas, suggesting the absence of a $P_{a}$ for the weak ITB formation in RS plasmas.

In order to investigate the condition of $T_{e}$-ITB formation in the absence of $T_{e}$-ITB, the power of ECH was scanned in the range $P_{\text{ECH}}$=0.6-3MW in PS and RS plasmas under the condition of $B_{r}$=3.7T, $I_{p}$=1MA, and line averaged $n_{e}$=0.75x10^{13}m^{-3}. Changes in the scale length of $T_{e}$ show the absence of a $P_{a}$ for ECH with RS, while an ITB is difficult to form without $T_{e}$-ITB with PS. These results indicate that magnetic shear might play a significant role in the $T_{e}$-ITB formation.

3. Dependence of diffusivity on the $E_{r}$ shear

Figure 5 shows profiles of $T_{e}$, the $E_{r}$ shear ($dE_{r}/dr$) and $\chi_{t}$ in a PS plasma with a strong ITB ($P_{\text{NB}}^{\text{abs}}$~8MW). The strong $E_{r}$ shear formed near the ITB layer, where a large reduction in $\chi_{t}$ was observed. It is seen that the ITB layer is located between the positions of the maximum and the minimum values of the $E_{r}$ shear. The $E_{r}$ shear becomes zero around the minimum of $\chi_{t}$. In the balanced toroidal momentum injection experiment, the $E_{r}$ profile has a local minimum at around half the minor radius, where the sign of the $E_{r}$ shear changes. Moreover, the experiments on the toroidal momentum input [3] indicate that a reduction in the maximum and/or the minimum values of $E_{r}$ shear leads to degradation of the ITB. These effects indicate the non-locality in the relation between the $E_{r}$ shear and the reduction of transport. Actually, abrupt variation of the diffusivity around the ITB (ITB event) was observed over a wide interval (30% of the minor radius) [6]. By considering the non-locality, we define the effective $E_{r}$ shear near the ITB as $(dE_{r}/dr)_{\text{eff}} = (|(dE_{r}/dr)_{\text{max}}| + |(dE_{r}/dr)_{\text{min}}|)/2$. 

**FIG. 4.** Dependence of $\chi$ on heat flux in RS plasmas. (a) Comparison of $T_{e}$ profiles at each $P_{\text{NB}}^{\text{abs}}$. (b) $\chi_{t}$ and $\chi_{\phi}$ as a function of $P_{\text{NB}}^{\text{abs}}$. No L-mode phase, where $\chi$ increases with heating power, was observed, suggesting the absence of a threshold power. (c) Relation between ion heat flux divided by ion density and ion temperature gradient.

**FIG. 5.** Non-locality in the relation between $E_{r}$, shear and $\chi_{t}$. (a) $T_{e}$ profile in the strong ITB. (b) Profiles of $E_{r}$ shear (solid curve) and $\chi_{t}$ (dotted curve).
FIG. 6. (a) $\chi_i$ and (b) $L_{\text{T}_i}$ as a function of the effective $E_z$ shear for PS plasmas. The dotted line indicates the time trace. There exists a critical $(dE_z/dr)_{\text{eff}}$ to change the state from a weak to a strong ITB. (c) The critical $(dE_z/dr)_{\text{eff}}$ depends on the poloidal magnetic field at the ITB.

The dependences of $\chi_i$ and the scale length of $T_i$ ($L_{\text{T}_i}$) on $(dE_z/dr)_{\text{eff}}$ in PS plasmas are shown in Figs. 6(a) and (b). The same cases shown in the previous figures are plotted. The value of $\chi_i$ increased with $(dE_z/dr)_{\text{eff}}$ for the cases with no ITB, whereas $\chi_i$ decreased for the data with weak and strong ITBs. There exists a critical value of $(dE_z/dr)_{\text{eff}}$ to change the state from a weak to a strong ITB. The value of $L_{\text{T}_i}$ was constant for the cases with no ITB, and gradually decreased with increases in $(dE_z/dr)_{\text{eff}}$ for the cases with a weak ITB. A drop in $L_{\text{T}_i}$ was observed at the strong ITB formation. The possible physical processes involved in the formation of weak and strong ITBs are considered as follows. The $E_z$ shear increased with an increase in heating power due to the increase in the pressure gradient. The core confinement was improved, which corresponded to the formation of a weak ITB. Once the plasma state changes in acquiring the weak ITB, the $E_z$ shear is enhanced by an increase in the heating power, and $\chi_i$ is gradually decreased. The growth of a weak ITB due to the gradual reduction in $\chi_i$ leads to an increase in the $E_z$ shear. The transport properties change according to the transition from a weak to a strong ITB when the $E_z$ shear exceeded the critical value. It is found that the critical value of $(dE_z/dr)_{\text{eff}}$ correlates with the poloidal magnetic field ($B_p$) at the ITB as shown in Fig. 6(c).

The dependences of $\chi_i$ and $L_{\text{T}_i}$ on $(dE_z/dr)_{\text{eff}}$ in RS plasmas are shown in Figs. 7(a) and (b). In the weak ITB plasmas, $\chi_i$ stays constant and $L_{\text{T}_i}$ slightly decreases with an increase of $(dE_z/dr)_{\text{eff}}$. The first drop in $\chi_i$ and $L_{\text{T}_i}$ was observed at $(dE_z/dr)_{\text{eff}} \sim 70 \text{kV/m}^2$, which is smaller than that in the case of a PS plasma. The second drop in $\chi_i$ and $L_{\text{T}_i}$ is observed at a larger value of $(dE_z/dr)_{\text{eff}}$ when $\chi_i$ is reduced to the neoclassical level. Discontinuous evolution of the strong ITB was sometimes observed in high performance RS discharges as shown in Fig. 8. In this discharge, the transition for strong ITB formation occurred at $t=4.2 \text{sec}$, and then a box-type ITB was already formed by $t=4.9 \text{sec}$. Rapid increases of $T_i$ and $T_e$ inside the ITB layer was observed at $t=4.9 \text{sec}$ and $t=5.5 \text{sec}$. These results suggest that there are
multiple levels of reduced transport in the strong ITB in the RS plasma. Comparison between the results shown in this paper and the local linear analysis [7] of microinstabilities is future work.

4. Summary and discussion

In order to address the $P_w$ and control of ITBs, the dependence of $\chi$ on the $E_s$ shear was investigated in JT-60U experiments with a dedicated scan of the NB heating power. In PS plasmas, $\chi$ in the core region generally increases with the heating power, similar to the L mode at low heating power. However, as a result of the intensive central heating, which is relevant to the enhancement of the $E_s$ shear, a weak ITB is formed, and $\chi$ in the core region starts to decrease. Corresponding to a further increase of the heating power, a strong ITB is formed and $\chi$ is reduced substantially. In RS plasmas, however, no power degradation of $\chi$ is observed, suggesting absence of a $P_w$ for the weak ITB formation. The $T_r$-ITB strongly correlates with the $T_r$-ITB in PS and RS plasmas. Furthermore, the results of the $T_r$-ITB formation experiment performed using EC heating in the absence of a $T_r$-ITB indicate that magnetic shear may be very influential for the $T_r$-ITB formation. On the other hand, $E_s$ shear is one of the key factors for the $T_r$-ITB formation. In weak ITB plasmas, $\chi$ decreases gradually with increasing $E_s$ shear for both PS and RS plasmas. There exists a critical value of $(dE_s/dr)_{\text{crit}}$ to change the state from a weak to a strong ITB. It is found that the critical value of $(dE_s/dr)_{\text{crit}}$ in PS plasmas correlates with $B_s$ at the ITB. There are multiple levels of reduced transport in the strong ITB for RS plasmas.

As for the $P_w$, it is difficult to define the $P_w$ for a weak ITB due to the absence of a clear transition, particularly since the $P_w$ for RS is very small, suggesting the absence of a $P_w$. On the other hand, the $P_w$ for a strong ITB is defined unambiguously both for PS and RS. As for the controllability of the ITB quality, a weak ITB is favorable for continuous control, while a strong ITB seems to be unfavorable due to the rapid drop in $\chi$. Since there are multiple levels of reduced transport in the strong ITB for RS, however, the plasma can move between several levels discontinuously.

Acknowledgement

The authors would like to thank the members of JAERI who have contributed to the JT-60 project.

References

Properties of Internal Transport Barrier Formation in JT-60U

Naka Fusion Research Establishment, Japan Atomic Energy Research Institute,

K. W. Hill, G. Rawlott
Princeton Plasma Physics Laboratory

19th IAEA Fusion Energy Conference
14 to 19 October 2002
Lyon, France

Introduction

- Threshold power for the ITB formation is one of the critical issues for the application of ITB to ITER.
- It is important to investigate the dependence of diffusivity on heat flux for the weak and strong ITB formation.
- It is important to clarify the relation between \( T_e \)-ITB and \( T_i \)-ITB.
- In order to maintain high confinement and high stability, ITBs have to be actively controlled.
- Active control of ITB strength based on modification of \( E_i \) shear profile. (Y. Sakamoto, 2000 IAEA)

Contents
1. \( T_e \)-ITB formation by ECH in PS and RS plasmas. (Effect of magnetic shear)
2. \( T_i \)-ITB formation by HB in PS and RS plasmas. (Effect of \( E_i \) shear)

Magnetic shear is a key factor for \( T_e \) ITB formation

- Inverse scale length of \( T_e \) \( \lambda(T_e) \) for PS plasmas stays constant with increasing \( P_{EC} \).
  \( T_e \)-ITB was not formed without ITB in PS region.
  The threshold power for PS is large compared with that for RS.
- Inverse scale length of \( T_e \) \( \lambda(T_e) \) for RS plasmas increased gradually with \( P_{EC} \).
  \( T_e \)-ITB was formed without ITB in RS region.
  Gradual increase suggests that absence of threshold power for \( T_e \) ITB formation.

Experimental condition: \( B_t \sim 3 \text{ T}, T_e\sim 1 \text{ eV}, n_e\sim 0.8 \times 10^{19} \text{ cm}^{-3}, T_i\sim 0.75 \text{ eV} \text{cm}^{-2} \text{ s}^{-1} \)
- On-axis EC power \( P_{EC} \) was varied from 0.8 to 3 \text{ MW} in PS and RS plasmas.
- EC deposition and \( T_i \) profile (without ITB) are similar in each case.
- Change in the \( T_i \) profiles are significantly different in each case.
$T_e$ ITB appeared in PS region when $T_i$ ITB was formed

- $P_{\text{eb}}$ was raised during ECH.
- Strong $T_i$ ITB appeared in PS region, then $T_e$ ITB was formed.
- $T_e$ ITB formation correlates with $T_i$ ITB in time and space.

Strong ITB appeared with clear transition in $T_i$ and $T_e$

- High heating power cases with $P_{\text{eb}} = 15$ MW (PS), 17 MW (RS).
- Sudden change in $T_i$ and $T_e$ was simultaneously observed at the same location.
- $T_e$ ITB strongly correlates with $T_i$ ITB in both PS and RS.

$T_i$ ITB formation – NB power scan –

Experimental condition:
- $B_T = 3.7$T, $I_T = 1.3$MA, $q_{95} = 5$, $\delta = 0.2$, $n_T = 1.0 \times 10^{19} \text{cm}^{-3}$, Balanced momentum injection.
- NB power ($P_{\text{eb}}$) was varied from 2 to 17 MW in PS and RS plasmas.
- Change in $T_i$ profiles are significantly different in both PS and RS plasmas.

Strong ITBs possess the local minimum of $\chi_i$ and $\chi_e$

Strong ITBs, which are characterized by local drop of $\chi_i$, are formed in both PS and RS plasmas.

Reduced transport level is different form each case.

- In PS plasmas,
  $\chi_i^{\text{itb}} - \chi_i^{\text{itb}} = 3 \times \chi_i^{\text{noo}}$.
- In RS plasmas,
  $\chi_e^{\text{itb}} - \chi_e^{\text{itb}} = 3 \times \chi_e^{\text{noo}}$. 
Dependence of $\chi$ on heat flux

- With increasing $P_{\text{inj}}$, no ITB -> weak ITB -> strong ITB.
- For the no ITB, $\chi$ increases with $P_{\text{inj}}$ in PS, i.e., L-mode, while no L-mode phase in RS.
- In the weak ITB, $\chi$ decreases gradually with $P_{\text{inj}}$.
- In the strong ITB, $\chi$ drops remarkably with $P_{\text{inj}}$.

Strong correlation between $\chi$ and $\chi_{\text{th}}$ (similar response to $P_{\text{inj}}$).

Relation between heat flux and $T_e$ gradient shows a bifurcation in transport.

Non-locality in relation between $E_t$ shear and $\chi$

- Strong $E_t$ shear is formed near the ITB layer.
- ITB layer locates between the positions of the maximum and the minimum values of the $E_t$ shear.
- The $E_t$ shear becomes zero around the minimum of $\chi$.
- ITB event (S.V.Neudatchin et al., in Fusion Energy 2000, EXP5/01)
  - The abrupt variation of the diffusivity around the ITB was observed in wide space (30% of minor radius).
- ITB degradation by momentum injection
  - $\chi$ is important for the sustainment of ITB.
  - $\chi_{\text{th}}$ is important for the sustainment of ITB.

Local description is insufficient. Non-locality is important.

Strong ITB was weakened by toroidal momentum injection

Non-locality in relation between $E_t$ shear and ITB degradation

- CO injection
  - Right before the degradation, $\left(\frac{dE_t}{dr}\right)_{\text{min}}$ became small.
- CTR injection
  - Right before the degradation, $\left(\frac{dE_t}{dr}\right)_{\text{min}}$ became small.

Dependence of $\chi_t$ on the effective $E_t$ shear

By considering the non-locality, we define the effective $E_t$ shear near the ITB as

$$\left(\frac{dE_t}{dr}\right)_{\text{eff}} = \frac{1}{2} \left( \left| \frac{dE_t}{dr} \right| + \left( \frac{dE_t}{dr} \right)_{\text{min}} \right)$$

$\chi_t$ is correlated with $\left(\frac{dE_t}{dr}\right)_{\text{eff}}$ rather than $\left(\frac{dE_t}{dr}\right)_{\text{min}}$.
Critical (dE/|dr|)_{eff} depends on B_p at ITB

- For the no ITB, \( x \) increases with (dE/|dr|)_{eff} and L_m stays constant.
- In the weak ITB, \( x \) and L_m decrease gradually with (dE/|dr|)_{eff}.
- In the strong ITB, \( x \) and L_m drop remarkably with (dE/|dr|)_{eff}.

There exists a critical (dE/|dr|)_{eff} to change its state from weak to strong ITBs.

Critical (dE/|dr|)_{eff} increases with B_p at ITB.

### Threshold power for the ITB formation and Controllability of the ITB

Threshold power (P_r):

- For a week ITB, it is difficult to define P_r due to the absence of a clear transition.
- For a strong ITB, P_r is defined unambiguously.

Controllability of the ITB:

- A weak ITB is favorable for continuous control.
- A strong ITB seems to be unfavorable due to the rapid drop in \( x \).
- Possibility of a strong ITB control is the use of discontinuous transport level.

Multiple levels of reduced transport in RS plasmas

- In the week ITB, \( x \) and L_m stay constant with (dE/|dr|)_{eff}.
- In the strong ITB, \( x \) and L_m decrease stepwise with (dE/|dr|)_{eff}.

Discontinuous evolution of the strong ITB was observed in several RS discharges.

Summary

- \( T \_r \), ITB formation
  - Magnetic shear is a key factor.
  - In RS plasmas, absence of threshold power of EC.
  - In PS plasmas, no ITB in the range of P_{ordib} - 3MW.
  - \( T \_r \) ITB appeared in PS region when \( T \_r \) ITB was formed.
  - \( T \_r \)-ITB formation strongly correlates with \( T \_r \)-ITB in time and space.

- \( T \_r \), ITB formation
  - E_shear is a key factor.
  - With increasing E_shear, ion transport property changes
    - L-mode \to weak ITB \to strong ITB (no L-mode phase in RS plasmas)
    - In weak ITB plasmas, \( x \) decreases gradually with increasing E_shear .
    - Critical E_shear for the strong ITB in PS plasmas depends on B_p at ITB.
    - There are multiple levels of reduced transport in the strong ITB of RS.

- Threshold power (P_r):
  - It is difficult to define P_r for a weak ITB due to the absence of a clear transition, while P_r for a strong ITB is defined unambiguously.
  - Controllability of the ITB quality:
    - A weak ITB is favorable for continuous control, while a strong ITB seems to be unfavorable due to the rapid drop in \( x \).
    - Possibility of strong ITB control is the use of discontinuous transport level.
1.14 Effects of complex magnetic ripple on fast ions in JFT-2M ferritic insert experiments


1) Japan Atomic Energy Research Institute (JAERI), Tokai-mura, Japan
2) Princeton Plasma Physics Laboratory (PPPL), Princeton, USA

e-mail contact of main author: shinohak@fusion.naka.jaeri.go.jp

Abstract. In JFT-2M, the ferritic steel plates (FPs) were installed inside the vacuum vessel all over the vacuum vessel, which is named Ferritic Inside Wall (FIW), as the third step of the Advanced Material Tokamak Experiment (AMTEX) program. A toroidal field ripple was reduced, however the magnetic field structure has become the complex ripple structure with a non-periodic feature in the toroidal direction because of the existence of other components and ports that limit the periodic installation of FPs. Under the complex magnetic ripple, we investigated its effect on the heat flux to the first wall due to the fast ion loss. The small heat flux was observed as the result of the reduced magnetic ripple by FIW. Additional FPs were also installed outside the vacuum vessel to produce the localized larger ripple. The small ripple trapped loss was observed when the shallow ripple well exist in the poloidal cross section, and the large ripple trapped loss was observed when the ripple well hollow out the plasma region deeply. The experimental results were almost consistent with the newly developed Fully three Dimensional magnetic field Orbit-Following Monte-Carlo (F3D OFMC) code including the three dimensional complex structure of the toroidal field ripple and the non-axisymmetric first wall geometry. By using F3D OFMC, we investigated the effect on the ripple trapped loss of the localized larger ripple produced by FPs in detail. The ripple well structure, e.g., the thickness of the ripple well, is important for ripple trapped loss in complex magnetic ripple rather than the value defined at one position in a poloidal cross section.

1. Ripple reduction by Ferritic Inside Wall

A potentially significant heat load to the first wall can occur in ITER-size machines due to toroidal field ripple arising from the discreteness of the TF coils (TFCs). The ripple reduction by using ferritic steel plates (FPs) under TFCs was demonstrated in JFT-2M [1]. In ITER, the ripple reduction method by using FPs is planed to be applied. However, it would be difficult to install FPs with perfect toroidal symmetry because of interference with other components and ports, such as neutral beam injection ports. In this situation, “N”-fold toroidal symmetry is broken, a toroidally localized larger ripple cannot be reduced and the structure of the TF ripple becomes complex. Here “N” is the number of TFCs.

As the third stage of AMTEX program [2,3], we have installed ferritic steel F82H all over the vacuum vessel inside the vacuum vessel in F.Y. 2001 and have started the experiments from April in F.Y. 2002 [4]. This configuration is a demonstration of the ferritic steel blanket wall. Most of FPs under TF coils outside vacuum vessel was removed. We reduced the magnetic ripple by optimizing the thickness of FPs. Though the FPs are installed inside vacuum vessel,
the FPs do not touch plasma directly with carbon tile limiters and divertor. We refer this condition as Ferritic Inside Wall (FIW).

Figure 1 shows the strength of the toroidal magnetic field for only TFC, FIW(calculation), and FIW(experiment) at the position of \((R=1.6m, Z=0m)\), which is referred as "mid" below, in the case of \(B_0=1.3T\). In JFT-2M, the number of toroidal field coils is \(N=16\) and the period of the TF ripple in the case of only TFC was 22.5 degree. The ripple amplitude defined as \(\delta=(B_{\text{max}}-B_{\text{min}})/(B_{\text{max}}+B_{\text{min}})\) for only TFC was about \(2\%\). The ripple structure of FIW has no periodicity in the toroidal direction and is complex because of the limitation of the installation of the FPs. The limitation comes from the compatibility with the other facilities such as the neutral beam injection system, the antenna of fast wave injection, the toroidal insulation structure, and the system of the plasma diagnostics. The ripple amplitude defined as \((B_{\text{max}}-B_{\text{min}})/(B_{\text{max}}+B_{\text{min}})\) is not a good global indicator in this situation because \(B_{\text{max}}\) and \(B_{\text{min}}\) are not typical values of the complex ripple structure. Here we use the standard deviation normalized by the average, defined as \(\delta_{\text{nd}}=\sqrt{\langle B^2 \rangle - \langle B \rangle^2} / \langle B \rangle \), as an indicator of a ripple amplitude, where \(\langle . . \rangle\) means the average over the toroidal direction. The value of \(\delta_{\text{nd-mid}}\) is \(0.47\%\) for \(B_0=1.3T\) in FIW. The ratio of \(\delta_{\text{nd-mid}}\) after the installation of FIW to that of only TFCs is less than \(1/3\) for \(B_0=1.0 - 1.9T\).

The measured field structure is almost consistent with that calculated by FEMAG code. It is considered the difference between the measured value and the calculated one comes from the error of the installation of FPs and the error of the measurement.

In the toroidal ripple structure produced only by TFCs, only one ripple well with a sinusoidal shape exists between two consequent TFCs. However the toroidal ripple structure formed with FPs is not simple as shown in FIG. 1. We cannot use the well-known \(\alpha\) parameter, \(\alpha=r/NRq\delta\), simply to determine the ripple well structure, where \(r\) is the minor radius, \(N\) is the number of toroidal symmetry, \(R\) is the major radius, \(q\) is the safety factor, and \(\delta\) is the ripple amplitude defined above. Here, we define the ripple well as the existence of minimum \(B\) along a field line. At first, we determine a toroidal angle, \(\phi_0\), at which we want to know the existence of the ripple well. Secondly we trace one of magnetic field lines from \(\phi_0\) to +/- \(\Delta\phi =11.25(=22.5/2)\) degree. We consider the magnetic field line has a ripple well structure at the toroidal angle, \(\phi_0\), when we encounter the stronger field than that at \(\phi_0\). The magnetic field is determined by the following way; an axisymmetric MHD equilibrium with FPs is calculated by MEUDAS code. The magnetic field produced by TFCs and FPs is calculated by FEMAG code with magnetic field from TFCs, and poloidal field from plasma and vertical coils calculated by MEUDAS code. Figure 2 shows the ripple well structure obtained in this way and shows the case of \(B_0=1.3T\). The ripple well exists only in a very small region of plasma reflecting the small value of toroidal ripple.

**FIG.2. Ripple well structure of FIW**

![Ripple well structure of FIW](image)

**FIG.3. Heat flux normalized by NB power \(P_{NB}\) at mid-plane estimated by IRTV measurement versus \(B_0\)**

![Heat flux normalized by NB power](image)
FIG. 4. Ripple well structure of L EFP, (a) and M+L EFP, (b)

We injected neutral beams (NB) to this plasma configuration of FIW in the co-direction to the plasma current and the toroidal field tangentially. The power of NB (P_{NB}) was \sim 0.5MW. The plasma was L-mode and the plasma parameters are $n_e \sim 2 \times 10^{19} \text{m}^{-3}$, $T_{e0} \sim 1 \text{keV}$, $T_{i0} \sim 1 \text{keV}$, $q_{95} \sim 4$, the slowing-down time is \sim 30 ms. We measured the temperature increment on the first wall by the IRTV camera. The temperature increase was observed only near mid-plane. This is consistent with the ripple well structure. Because it was expected most loss be the banana drift loss and the direct loss from the ripple well structure.

Ferritic steel F82H produces the saturated magnetic field of 1.96 T at the external field of 0.25 T. Thus the ripple amplitude is changed when the magnetic field produced by TFCs is changed. The value of $\delta_{\text{std-mid}}$ is about 1.21% and 0.4% for $B_{t0} = 0.85T$ and $B_{t0} = 1.6T$, respectively. Figure 3 shows the heat flux normalized by $P_{NB}$ at the mid-plane estimated by the IRTV measurement for $B_{t0} = 0.85T$, 1.3T, and 1.6T. The value of $\delta_{\text{std-mid}}$ is also shown in FIG. 3. The heat flux at the mid-plane is small at $B_{t0} = 1.3T$ and 1.6T, around which FIW is optimized, and is large when $B_{t0} = 0.85T$ which is out of the optimum toroidal field.

2. Effect of Local Ripple

Localized larger ripple structure is expected to be used for the ripple-injection [5], the ripple-fueling [6] or He exhaust control [7] in the theoretical studies. However the experiment under the localized larger ripple was few. In the PLT tokamak, it was found that a localized larger ripple had little effect on the global confinement of tangentially injected beam ions in the experiments with the accident to one of TFCs [8]. However, the effect of fast ions on the first wall was unclear. We have produced the localized larger ripple by installing the external ferritic plate (EFP). The EFIs have been installed outside the vacuum vessel under TFC-3 ($\phi_i = -45$ degree). The poloidal section is inside the viewing area of the IRTV camera. We have produced two types of ripple structures. In the first case, EFP is installed on a lower-shoulder part (L EFP). In the second case, EFP is installed on a Mid-plane part and a Lower-shoulder part (M+L EFP). Figure 4 shows the ripple well structure defined above around TFC-3. We can see the poloidal structure of ripple well have been modified by EFIs. Both cases produce the ripple well mainly below the mid-plane. And the ripple well structure for M+L EFP is similar to that proposed for the ripple-injection and the ripple-fueling.

The direction of the grad-B drift of ions is downward in experiments, thus the heat flux at the downward position of the plasma corresponds to the ripple trapped loss. Figure 5 shows the heat flux normalized by $P_{NB}$ estimated by the IRTV measurement at the downward position of the plasma. In response to the modified ripple well structure, we observed the separated hot spot due to the ripple trapped loss, which was not observed in FIW. The ripple trapped loss
FIG.5. Heat flux normalized by $P_{NB}$ estimated by the IRTV measurement at the downward position of the plasma for L EFP and M+L EFP. $\delta_{\text{loc-low}}$ and $\delta_{\text{std-low}}$ are the value at the lower shoulder part of plasma, $R = 1.55$ m and $Z = -0.2$ m. $\Delta_{\text{ripple}}$ is the thickness of the ripple well at $Z \sim -0.15$ m.

increase with $\delta_{\text{loc-low}}$ and $\delta_{\text{std-low}}$. However it looks the loss suddenly start to increase at $\delta_{\text{loc-low}} \sim 0.5\%$ and $\delta_{\text{std-low}} \sim 0.4\%$. In other words, the effect of localized larger ripple on the heat flux cannot be observed till $\delta_{\text{loc-low}} \sim 0.5\%$ and $\delta_{\text{std-low}} \sim 0.4\%$ in this configuration. Figure 5 (c) shows the heat flux of the ripple trapped loss versus the thickness of the ripple well at $Z \sim -0.15$ m ($\Delta_{\text{ripple}}$ shown in FIG.4). It looks the thickness of the ripple well is a more appropriate parameter for the ripple trapped loss in the complex magnetic ripple than $\delta_{\text{loc-low}}$ and $\delta_{\text{std-low}}$ in this configuration. I will discuss this issue by using simulation code later.

We also compared the toroidal rotation ($V_t$) and the ion temperature ($T_i$) among the FIW, L EFP, and M+L EFP configurations by using the charge exchange recombination spectroscopy. We cannot see a clear effect of the localized larger ripple on $V_t$ and $T_i$ for OH and L-mode plasma. This result is consistent with the result observed on PLT through Fe XX [8].

3. Comparison with Fully three Dimensional magnetic field (F3D) OFMC

We have improved OFMC code in order to understand the heat flux to the first wall due to the fast ion loss in the complex toroidal magnetic field. So far, OFMC assumed N-fold toroidal symmetry; the toroidally periodic boundary condition was used and the area of calculation was between two TFCs. After the installation of FPs on JFT-2M, 16-fold toroidal symmetry of magnetic field is broken and the magnetic field structure is complex. We need a new tool that can treat the complex magnetic structure in order to compare the experimental observation with the classical theory that is used in the above OFMC. We have developed OFMC without N-fold toroidal symmetry. Here, we call this code Fully three Dimensional magnetic field (F3D) OFMC.

The poloidal shape of the first wall is also not axisymmetric in JFT-2M. Banana particles hit the first wall non-axisymmetrically. In F3D OFMC, we can also allow for the non-axisymmetric first wall as the boundary to estimate heat flux. As the result of including the non-axisymmetric first wall, we observed that the limiter and the limiter-like structure cut banana ions and passing ions around the mid-plane at the low field side in the results of F3D OFMC calculations. The heat flux is large and localized at the limiter and the limiter-like structure. This new feature of F3D OFMC is useful for the estimation of the heat flux due to alpha particles to the irregular structure, such as ICRF antennas, in the fusion reactor.

We compared the heat flux of the ripple trapped loss between experiments and F3D OFMC calculations. The results are shown in FIG.6. The heat flux of experiments is almost consistent with that of F3D OFMC calculations. We also compared the poloidal structure of the heat flux of the ripple trapped loss between experiments and F3D OFMC calculations. The poloidal structure of the heat flux of experiments is also almost consistent with that of F3D OFMC calculations.
FIG. 6. The comparison of the heat flux of the ripple trapped loss between experiments and F3D OFMC calculations.

FIG. 7. Ripple well structures for the various thickness of M EFP. The M EFP is divided to nine plates. The number near ripple wells means the numerator; namely 9 means full thickness of M EFP. 3 means 1/3 thickness of full M EFP.

FIG. 8. The heat flux of the ripple trapped loss versus δ_{loc-low}. (a), and the thickness of the ripple well at Z~ -0.15 m, (b). Closed circle:F3D OFMC, Open markers: Experiments.

We compared the total loss of fast ions between FIW and only TFC cases at B_{o}=1.3T by using F3D OFMC. The total loss in the case of FIW is about 1/3 as large as that in the case of only TFCs in responding to the reduction of the toroidal magnetic ripple.

In the experiments, it looks the ripple trapped loss suddenly start to increase at δ_{loc-low} ~ 0.5. We expected the shape of the ripple well is an important feature. We have run F3D OFMC in the various thickness of M EFPs in order to change ripple well structure in a poloidal cross section. Figure 7 shows the ripple well structures for the various thickness of M EFP. Figure 8(a) shows the heat flux of the ripple trapped loss versus δ_{loc-low}. The heat flux start to increase at δ_{loc-low} ~ 0.5 %. The experimental results are consistent with F3D OFMC calculations. Figure 8(b) shows the heat flux of the ripple trapped loss versus the thickness of the ripple well at Z~ -0.15 m. The ripple trapped loss is almost proportional to the thickness of the ripple well of the localized larger ripple. The population of fast ions is larger at a smaller minor radius. Therefore it is considered that a larger amount of fast ions can be trapped in the localized ripple when the thickness of the ripple well is thicker. From this results, the ripple well structure, e.g. the thickness of the ripple well, is important for ripple trapped loss in complex magnetic ripple rather than the value defined at one position in a poloidal cross section, which is applied in simple magnetic ripple produced by TFC. And it is also considered the FP installation near mid-plane might be enough for the reduction of the ripple induced loss of ITER through the reduction of the thickness of the ripple well.

Effects of complex magnetic ripple on fast ions in JFT-2M ferritic insert experiments

JAERI
¹PPPL

Contents

- Heat flux of energetic ions in optimized ferritic-steel covering (called: Ferritic Inside Wall, FIW)
- Heat flux with local toroidal ripple by using extra ferritic-steel plate
- Comparison with new F3D OFMC code

Introduction

Ferritic Steel F62H is a leading candidate as the structural material for fusion reactor because of low activation, low swelling, high heat resistance. Ferritic Steel is ferromagnetic.

• Can be used to reduce toroidal ripple, especially when ferritic steel is placed under toroidal field coil.
• N-fold toroidal symmetry of magnetic field will be broken and complex magnetic ripple will be produced because of limitation of installation, e.g. existence of tangential beam port.
• Localized larger ripple of magnetic field can be produced. Local ripple structure is expected to be used for ripple-injection, ripple-fueling, or He exhaust control. However, experiments and simulation in complex magnetic ripple were few so far.

Need understanding of complex magnetic ripple.

Ferritic-steel installation with Optimized thickness - Ferritic Inside Wall, FIW -

![Diagram](image)

\[ \text{Calculated value} \]
Ripple well exist only in a very small region of plasma corresponding to small value of toroidal ripple.

*We cannot use well-known $\alpha$ parameter, $n/Nk\delta$, to determine ripple well structure in complex magnetic ripple.

*Here ripple well is defined as existence of minimum $B$ along field line around one toroidal section.

**Experimental Setup**

*IRTIV is used in order to estimate heat flux.
*Here NB is injected in co-direction to $I_p$ and $B_t$ tangentially.
$P_{NB} = 0.5 \text{ MW}, E_0 = 36 \text{ keV}, R_{in}=1.03 \text{ m (R_0=1.31 \text{ m})}$
*L-mode: $n_e = 2 \times 10^{19} \text{ m}^{-3}, T_{e0}=1 \text{ keV}, T_{i0}=1 \text{ keV}$ during NB injection.

**Heat flux had small value at the toroidal field, at which FIW design was optimized**

*Banana drift loss increase with $\delta_{\text{std,mid}}$
Toroidally localized larger ripple is produced with external ferritic plate (EFP).

- Localized larger ripple structure is expected to be used for ripple-injection, ripple-fueling, or He exhaust control in theoretical studies.

- Characteristics of heat flux caused by local ripple is measured by using EFP under toroidal field coil (TFC).
- Experiments were performed in two configurations:
  1. On Lower-shoulder part (L)
  2. On Mid-plane part (M) and Lower-shoulder part (L)

Poloidal structure of ripple well have been modified by EFPs.

One toroidal section around TFC-3

B₀₀=1.3T

Magnetic field pattern used in experiment
- Measured -

Heat flux induced by EFP

\( B₀₀=1.3T \)

\( d₀₀=4 \)

w/o EFP
(Optimized ferritic wall)

\( \Delta T \) [deg.]

with M+L EFP
under TFC

\( \Delta T \) [deg.]

Banana drift loss and Ripple trapped loss increase with EFP on mid-plane and lower shoulder part under TFC.
Thickness of ripple well, $\Delta_{\text{ripple}}$, looks a more appropriate parameter.

Effect of local ripple on $V_t$ and $T_i$ is negligible for OH and L-mode plasma.

Dependence of Ripple trapped loss on $B_{10}$ with EFP.

Development of OFMC w/o n-fold toroidal symmetry

- So far Orbit-Following Monte-Carlo code (OFMC) assumed N-fold toroidal symmetry; toroidally periodic boundary condition was used and its area of calculation was between 2 TFCs.
- On JFT-2M, 16-fold toroidal symmetry of magnetic field is broken and magnetic field structure is complex because of installation of FPs. We needed new tool.

- Fully three Dimensional magnetic field OFMC (3D OFMC) was newly developed.
  - Axisymmetric MHD Equilibrium with ferritic Wall is calculated by modified MEUDAS code.
  - Magnetic field produced by TFC and ferritic wall is calculated by FEMAG code with magnetic field from TFC and poloidal field from plasma and vertical coils.
Complex first wall geometry is considered

- Poloidal shape of first wall is not axisymmetric. Banana particles can be cut unaxisymmetrically.
- In JFT-2M, there exist nine types of shapes of first wall.

In F3D OFMC, we can also use unaxisymmetric first wall as boundary to estimate heat flux.

Comparison between EXP. and F3D OFMC

- Heat flux and peak position of experiments are almost consistent with F3D OFMC calculation.

Effect of complex first wall geometry was clearly observed in F3D OFMC result

- Large heat flux is observed at limiter and limiter-like structure around mid-plane at low field side in F3D OFMC calculation.
- This feature is useful to estimate heat flux of fast particles on irregular first wall structure, e.g., ICRF antenna, in reactor design.

Ripple trapped loss increase with thickness of ripple well of localized larger ripple

- Thickness of M EFPs is changed in order to change ripple well structure in a poloidal cross section in F3D OFMC.
- Heat flux start to increase at \( \Delta_{\text{ripple}} \approx 0.5 \text{ %} \) Exp., is consistent with F3D OFMC.
- FP installation on mid-plane is effective and enough for reduction of heat flux in ITER through reduction of thickness of ripple well.
Summary

- For the first time, we have demonstrated toroidal magnetic ripple reduction by the method of the optimization of thickness of FPs in FIW. We have observed the reduction of heat flux of energetic ions by using IRTV camera in FIW.

- We have performed experiments with localized larger toroidal ripple by using EFP. Heat flux of L EFP was about 3% of that of M+L EFP, though difference of δp,low and δsat,low is about 40% and 70%, respectively.

- It is considered ripple well structure, e.g. Δripple, is important for ripple trapped loss in complex magnetic ripple rather than value defined at one position in a poloidal section, e.g. δp,low, δsat,low

- Effect on Vt and Ti of L EFP and M+L EFP is negligible in OH and L-mode phase

- We have newly developed F3D OFMC, in which complex magnetic field and complex first wall is considered. F3D OFMC calculations are almost consistent with experiment results.
1.15 Characterization of Axisymmetric Disruption Dynamics toward VDE Avoidance in Tokamaks

Y. Nakamura 1), R. Yoshino 1), R.S. Granetz 2), G. Pautasso 3), O. Gruber 3) and S.C. Jardin 4)

1) Naka Fusion Research Establishment, JAERI, Naka-machi, Ibaraki 311-0193, Japan
2) Plasma Science and Fusion Center, MIT, Cambridge, Massachusetts 02139, USA
3) MPI für Plasmaphysik, EURATOM Association, D-85748 Garching, Germany
4) Princeton Plasma Physics Laboratory, Princeton University, New Jersey 08543, USA

e-mail contact of main author: nakamura@fusion.naka.jaeri.go.jp

Abstract Experiments and axisymmetric MHD simulations on tokamak disruptions have explicated the underlying mechanisms of Vertical Displacement Events (VDEs) and a diversity of disruption dynamics. First, the neutral point, which is known as an advantageous vertical plasma position to avoiding VDEs during the plasma current quench, is shown to be fairly insensitive to plasma shape and current profile parameters. Secondly, a rapid flattening of the plasma current profile frequently seen at thermal quench is newly clarified to play a substantial role in dragging a single null-diverted plasma vertically towards the divertor. As a consequence, the occurrence of downward-going VDEs predominates over the upward-going ones in bottom-diverted discharges. This dragging effect is absent in up-down symmetric limiter discharges. These simulation results are consistent with experiments. Together with the attractive force that arises from passive shell currents and essentially vanishes at the neutral point, the dragging effect explains many details of the VDE dynamics over the whole period of the disruptive termination.

1. Introduction

With regard to an area concerned for future tokamak reactors, the Vertical Displacement Events (VDEs) and its concatenate generation of halo-currents and their associated vessel forces is recognized as a crucial issue, and one that must be dealt with by disruption mitigation. Therefore, an underlying mechanism of the VDE was investigated by means of axisymmetric MHD simulations [1, 2]. For a tokamak with an up-down asymmetric passive shell, the simulation using the Tokamak Simulation Code (TSC) [3] clarified that the eddy currents induced by the plasma current quench give rise to a vertical imbalance of forces, causing a VDE [4]. If the current centroid of the pre-disruptive plasma is chosen to be close to the neutrally balanced vertical position (the so-called neutral point), the attractive forces due to the eddy currents will cancel at that point, and thus the VDE may be avoided. However, if the pre-disruptive plasma is positioned away from the neutral point, it will exhibit either an upward- or downward-going VDE, according to the initial position, consistent with JT-60U experiments [5]. It has also been demonstrated that for plasmas initially positioned at the neutral point and with feedback position control activated, an almost VDE-free and halo-current-free disruption can be made to occur [6, 7].

While such progress has been made, some recent observations of the plasma equilibrium response to transient disturbances due to ELMs and sawteeth indicate that the precise position of the neutral point are somewhat sensitive to profile parameter variations, and raise into question the practicality of controlling VDEs through the neutral point in future reactors [8]. This paper first describes the sensitivity of the neutral point to a variety of plasma shape and current profile parameters. We then describe a generalization of the VDE modeling to include the change of the plasma current profile during the disruption.

2. Neutral Point

The location of the neutral point is supposed to be linked up closely with a geometry of shell structures and probably with an arrangement of plasma shaping coils which differs from
tokamak to tokamak. Therefore, much different VDE characters from the JT-60U can appear in the Alcator C-Mod and ASDEX-Upgrade.

2.1. Validation Experiment on Alcator C-Mod

In Alcator C-Mod, disruption experiments conducted by injecting killer-pellets into plasmas with five pre-disruption equilibria were carried out to identify the location of the neutral point (Fig. 1). The disruptive plasmas initially positioned around the numerically determined neutral point \( Z = + \) a few cm, taking the much different shell-geometry from the JT-60U into consideration) exhibit upward- or downward-going VDEs as predicted by the TSC. And further, the plasma positioned close to the numerically determined neutral point stayed for \(~ 40\) msec. It thus follows that the neutral point is experimentally confirmed to widely exist as the TSC prediction as well as the JT-60U.

2.2. Neutral Point of ASDEX-Upgrade

The TSC simulation, which models a plasma current quench without the current profile change, reproduced a large variety of VDEs according to respective bottom-diverted ASDEX-Upgrade equilibria prior to the current quench (Fig. 2). Although the VDE rate significantly depends on the plasma shape and current profile parameters, neutral points that are seen to exist at \(~ +5\) cm above the horizontal midplane are not sensitive to those parameters with elongation of \( \kappa = 1.5, 1.6, 1.7 \), triangularities of \( \delta = 0.1, 0.25, 0.4 \), and a range of current profiles characterized by \( l_i = 0.7, 0.9, 1.1, 1.3 \) [9].

The ASDEX-Upgrade experiment (see Fig. 3) illustrates a new feature that was not

FIG. 1. Disruption dynamics of relevant VDEs to plasma current decay forced by killer-pellet injection in Alcator C-Mod. Neutral point is experimentally confirmed to exist as TSC prediction.

FIG. 2. TSC vertical excursions versus initial vertical positions of bottom-diverted ASDEX-Upgade plasmas. Neutral points are found at \(~ +5\) cm above horizontal midplane, being insensitive to plasma shape and current profile.

FIG. 3. Experimental excursions versus initial position. Neutral point of limiter plasmas is seen at \(~ 5\) cm above horizontal midplane as Fig. 2, whereas bottom-diverted plasmas exhibit downward (#14048) or upward VDEs (#14042).
observed in the TSC simulations shown in Fig. 2. The vertical excursions of the plasma current centroid due to the disruption induced VDE depend on whether the initial plasma configuration is diverted or not. For limiter discharges, which are essentially up-down symmetric (closed circles), the neutral point can be seen at $\sim +5$ cm above the horizontal midplane, consistent with the simulation results shown in Fig. 2. On the other hand, diverted discharges exhibit a wide variety of VDE behaviors, implying that a unique neutral point does not exist in ASDEX-Upgrade, as it does in the JT-60U [4] and Alcator C-Mod. For the bottom-diverted configuration (open circles), notice that even the discharges initially positioned considerably above the numerically determined neutral point exhibit downward-going VDEs, contrary to the TSC prediction (e.g. #14048). However, several exhibit upward-going VDEs in accordance with the TSC (e.g. #14042) [9].

3. Dragging effect due to profile change of plasma current

Both initial equilibria of the shots #14042 and #14048 are very similar (see Fig. 4), e.g. $\kappa = 1.6$, $\delta = 0.25$, and they are positioned at $Z = 19.8$ cm ( #14042) and at $Z = 17.3$ cm ( #14048), being well above the TSC neutral point ($\sim 5$ cm). A remarkable disparity between these two disruptions is the different change of the internal inductance, $\Delta l$, which is often observed during the thermal quench stage prior to the plasma current quench [10]. Note the large, rapid ($< 1.0$ msec) decrease of $\Delta l \sim 0.7$ at the thermal quench stage of #14048 ($t = 4.922$ sec in Fig. 4(a)) and the relatively small, slow ($\sim 5.0$ msec) decrease of $\Delta l \sim -0.3$ at the thermal quench stage of #14042 ($t = 4.901$ sec in Fig. 4(b)).

The TSC simulation of a rapid change of the current profile clarifies a vertical dragging of single null-diverted plasmas, which predominates over the growth of vertical instabilities (Fig. 5): as the current profile becomes broad ($\Delta l < 0$), the plasma tends to drag itself toward divertor, whereas a current peaking ($\Delta l > 0$) pulls the plasma out of the divertor [9]. Those substantially depend on a measure of the up-down asymmetry $\gamma = Z_f/Z_o$. A value of $\gamma = 1$ denotes an up-down symmetric, double null-divertor configuration, while $\gamma > 1$ ($\gamma < 1$) denotes a bottom (top)-divertor.

It thus follows that the significant change of $\Delta l \sim -0.7$ may drag the bottom-diverted

![Diagram](image)

**FIG. 4.** Discharges of (a) downward (#14048) and (b) upward (#14042) going VDEs. However, both equilibria prior to disruptions are similar, e.g. bottom-diverted, $\kappa = 1.6$, $\delta = 0.25$, and closely positioned as $Z = 19.8$ cm of #14042, while $Z = 17.3$ cm of #14048 (much above the TSC neutral point of $\sim 5$ cm). Note a large decrease of $\Delta l \sim 0.7$ at 4.922 sec (#14048), whereas a small decrease of $\Delta l \sim -0.3$ at 4.901 sec (#14042).
plasma (#14048) below the neutral point to cause a downward VDE due to the following plasma current quench. Meanwhile, the small change of $\Delta I \sim -0.3$, being insufficient to dragging the plasma (#14042) downward, leaves the plasma much above the neutral point even after the thermal quench. Consequently, it undergoes an upward VDE. In case of the limiter or double null-divertor ($\gamma = 1$) (such as the closed circles, Fig. 3), the dragging effect is always absent, consistent with the simulation (Fig. 2).

![Diagram](image)

4. Behavior details of axisymmetric disruption dynamics

The present VDE modeling now enables us to explain the precise details of axisymmetric disruption dynamics in ASDEX-Upgrade [9]. As an illustration, we consider in detail the shot #12086, shown in Fig. 6, which exhibits many of the characteristics generally important for disruption dynamics. Note that switchovers between the bottom-diverted (marked with gray) and top-diverted (not marked) configurations took place. At the pre-disruption phase (a), the bottom-diverted plasma positioned at $\sim 10$ cm above the horizontal midplane, a little higher than the neutral point ($\sim 5$ cm), starts to flatten the plasma current profile.

For the duration of the following thermal quench phase (b) that lasts for 2 msec, a decrease of the internal inductance of $\Delta I \sim -0.5$, i.e., a rapid flattening of the current profile caused by a minor collapse of the highly peaked current profile, appears together with an associated positive current spike of $\Delta I_p \sim 40$

![Diagram](image)

**FIG. 5.** Current profile changes ($\Delta I$) and vertical dragging effect ($dZ$), which substantially depends on up-down asymmetry $\gamma (= Z/Z_u)$ of single null-divertor, and is absent in double null-divertor ($\gamma = 1$). $Z_u$ or $Z_l$ means respective vertical distance between magnetic axis and bottom or top edge of outermost flux surface. Dragging toward divertor at flattening ($\Delta I < 0$), whereas pulling out of divertor at peaking ($\Delta I > 0$).

**FIG. 6.** Disruption dynamics of #12086. (a) : pre-disruption plasma positioned a little above neutral point ($\sim 5$ cm). (b) : first thermal quench with $\Delta I \sim -0.5$ drags bottom-diverted plasma downward. (c) : attractive force due to plasma current quench along with current sharpening drags the plasma positioned above neutral point upward. (d) : after switchover from bottom to top-divertor at 4.020 sec, sharpening begins to drag plasma downward, and ceases upward VDE. (e) : second thermal quench with $\Delta I \sim -0.75$ drags top-diverted plasma upward. (f) : plasma positioned much above neutral point undergoes upward VDE due to current quench.
kA and an inward radial shift of ~ 5 cm. Simultaneously, as expected, the bottom-diverted plasma exhibits a small vertical dragging toward the divertor. In the phase (c), the bottom-diverted plasma that still stays above the numerically determined neutral point exhibits an upward-going drift due to the combination of the attractive force of the induced eddy current and the upward dragging effect of the profile sharpening.

At 4.020 sec of the phase (d), the bottom-diverted configuration switches over to the top-diverted. The profile sharpening now begins to drag the plasma away from the top-divertor (downward), while the current quench continues to pull up the plasma that remains above the neutral point. Consequently, the upward-going VDE ceases in phase (d). Within a short duration of 1 msec in phase (e), a considerable decrease of $\Delta l \sim -0.75$ and an associated positive spike of $\Delta I_p \sim 100$ kA again occur. This drags the top-diverted plasma toward the divertor, i.e., upward, contrary to the downward in the previous phase (b). Finally, the current quench starts again in phase (f), and the plasma, which is now well above the neutral point, exhibits an upward-going VDE, regardless of the top- or bottom-diverted configuration.

5. Summary and conclusions

Concerning the VDE avoidance, any tokamak has been verified to possess its own advantageous neutral point which the present study revealed to be fairly insensitive to plasma shape and current profile parameters. And furthermore, a vertical dragging effect that arises from the plasma current flattening has been newly introduced to successfully explicate the predominant occurrence of the disruptive VDEs toward divertor, specific to the ASDEX-Upgrade. It is also demonstrated how the dragging effect, together with that of the imbalanced attractive force that may vanish at the neutral point, can explain the precise details of VDE dynamics.

It turned out that the new concept of the dragging effect strongly depends on a measure of the up-down asymmetry of the single null-diverted plasmas which closely connects with how apart a divertor coil is standing from the plasma, e.g., a far divertor coil of the ASDEX-Upgrade (outside the toroidal field coils) in contrast to near ones of the JT-60U and Alcator C-Mod (inside the toroidal field coils). As a consequence, it has been clarified that the dragging effect is more remarkable in the ASDEX-Upgrade than the others, and that such various dragging effect brings the disrupting plasmas a diversity of VDE dynamics.

In a future advanced tokamak like the ITER-FEAT phase, the disruptions will be associated with an advanced performance plasma operation regimes with reversed magnetic shear [11]. Therefore, the destruction of such a magnetic shear profile is expected to sharpen the current profile and consequently cause a VDE motion away from the divertor, in contrast to disruptions of normal shear plasmas with significant flattening. An integrated study on such details of the VDE in a reversed shear plasma is now under investigation.

References
Characterization of Axiallyymmetric Disruption Dynamics toward VDE Avoidance in Tokamaks

Y. Nakamura 1), R. Yoshino 1), R.S. Granetz 2), G. Pautasso 3), C. Gruber 3) and S.C. Jardin 4)
1) Naka Fusion Research Establishment, JAERI, Japan
2) Plasma Science & Fusion Center, MIT, USA
3) Max-Planck-Institut für Plasmaphysik, EURATOM Association, Germany
4) Princeton Plasma Physics Laboratory, Princeton University, USA

[ Aim ]
Avoidance of VDE (Vertical Displacement Event), Suppression of Divertor Halo-Current, and Consequent Soft-Landing Disruption

[ This Work ]
Advantageous "Neutral Point" (first found in JT-60U) to VDE Avoidance, Common or Specific Characters of VDE Dynamics to Tokamaks

Why VDE Avoidance for Soft Landing Disruption?

Typical Disruption Behavior

- VDE coincides with ip Quench.
- Large vertical shift toward divertor
- Formation of circuit going around plasma boundary and divertor structure

If VDE avoided, we can expect suppression of divertor Halo-current?
Soft Landing Disruption

"Neutral Point" as a "Watershed" of VDEs

JT-60U Disruption Database
(N. Fusion, Vol.36, No.3 (1996) 296.)

Numerical Model of Free Boundary Axiallyymmetric Plasma (Tokamak Simulation Code: TSC)

- Force Balance of Plasma Momentum Density:
\[ \frac{d}{dt} \left( \rho \mathbf{v} \right) = \nabla \cdot \mathbf{F} \]
- \( \mathbf{F} = -\mathbf{\nabla} \cdot \mathbf{P} \mathbf{v} + \mathbf{\tau} - \mathbf{b} \cdot \mathbf{v} \mathbf{\nabla} \rho \mathbf{v} \) : Viscosity Operator
- Faraday's and Ohm's Laws yield evolution equations for the poloidal flux \( \psi \) and toroidal field function \( g \):
\[ \frac{3}{2} \frac{d}{dt} \psi = \frac{1}{2} \int_{\Omega} \mathbf{B} \cdot \nabla \mathbf{B} \, \mathrm{d}V + \int_{\partial \Omega} \mathbf{B} \cdot \mathbf{v} \, \mathrm{d}A \]
at grid-point \( \mathbf{a} \) of solid conductor

TSC Simulation

- Up-down imbalanced attractive force that arises from eddy-current causes VDE during ip quench.
- Attractive force is always absent at "Neutral Point" (Z = ± 15 cm).

Stable Plasma Current Termination Attained at Neutral Point

(Y. Neytstent et al., N. Fusion 50 (1999) 555.)

In JT-60U, positioning plasma at TSC-estimated Neutral Point before ip quench and with weak feedback position control activated results in VDE avoidance and consequent Halo-current suppression.

Further Questions?

- Does N.P. exist widely in Alcator C-Mod, ASDEX-Upgrade as well as JT-60U?
- How sensitive is N.P. to plasma shape, current profile, etc?
Alcator C-Mod Disruption Experiment by Killer-Pellet Injection

"Neutral Point" consistent with TSC was confirmed as well as JT-60U.

- Positioned below N.P: Downward VDE
- Positioned around N.P: No VDE
- TSC "N.P." 

Neutral Point, VDE Behavior Detail of ASDEX-Upgrade

- much different shell-structure from JT-60U and Alcator C-Mod (up-down asymmetric passive stabilizer) (tor divertor coil from plasma +s near in JT-60U and Alcator C-Mod)
- wide parameters of plasma shape and p-profile $\kappa = 1.5 - 2.0$, $\delta = 0.1 - 0.5$, $\lambda = 0.7 - 1.3$
- flexible plasma configuration fixed limiter, top/bottom divertor

Vacuum Volume

Gushter Coil

TSC Neutral Points

- Although VDEs depend on plasma elongation and current profile, N.Ps gather around Z = +5 cm.
- Neutral Point is less sensitive to plasma shape and current profile.

Vertical Dragging due to Current-Profile Change

- up-down asymmetric, single null-diverted plasma ($\gamma \neq 1$) vertical dragging toward divertor
- up-down symmetric, DN diverted or limiter plasma ($\gamma = 1$) no dragging, as experiments.

Complex VDE Behavior on Disruption Database

Neutral Point of ASDEX-Upgrade? Downward VDE (#14048) Upward VDE (#14042)

- similar equilibria ($\kappa = 1.6$, $\delta = 0.25$, $\lambda = 1.4$, bottom-diverted) and both positioned at $Z_0 = +18$ cm > N.P.
- big difference in flattening of p-profile at thermal quench: $\Delta \lambda = 0.7$ (#14048), $\Delta \lambda = 0.3$ (#14042)

Vertical jump $\Delta Z$ due to $\Delta \lambda$?
Underlying Mechanism of VDEs during Disruption

<table>
<thead>
<tr>
<th>Disruption Event</th>
<th>Thermal Quench</th>
<th>Current Quench</th>
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<tbody>
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<td>Up-down Asymmetry (e.g. SN-div. Plasmas)</td>
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Conclusion

Main results obtained from Japan-US-EU collaboration:

- Neutral Point
  - widely exists, and strongly depends on shell structure.
  - confirmed advantageous to avoid VDEs.
  - greatly mitigates hard disruptions via Halo-current reduction.
- VDE Behavior Detail
  - now well understood by the VDE Mechanisms:
    - up-down imbalanced force of eddy-current (absent at N.P.),
    - dragging effect due to Ip-profile change (absent in DN-divertor).

Such strong dependence on up/down asymmetry of SN-diverted plasmas brings tokamaks a variety of different character of VDE detail.

Future Work
- Integrated study on VDE avoidance modelling of reversed shear plasmas in ITER advanced operation regime

Combination Effect of VDE Mechanisms on Disruption Dynamics
1.16 Interaction among different spatio-temporal scale fluctuations through zonal flows

Y. Kishimoto,\textsuperscript{1)} J.Q. Li,\textsuperscript{1,2)} Y. Idomura,\textsuperscript{1)} and A. Smolyakov\textsuperscript{3)}

\textsuperscript{1)} Naka Fusion Research Establishment, JAERI, Naka, Ibaraki 311-0193, Japan
\textsuperscript{2)} South Western Institute of Physics, Chengdu, People’s Republic of China
\textsuperscript{3)} University of Saskatchewan, Saskatoon, S7N 5E2 Canada

E-mail: kishimoy@fusion.naka.jaeri.go.jp

Abstract. Zonal flows generated by fluctuations on some scale are expected to affect fluctuations on other different spatial-temporal scales in addition to regulating their own fluctuation. As an example, here we discuss ITG turbulence embedded in a small scale ETG-driven zonal flow based on 3-dimensional gyro-fluid simulations. At first, we identified the ETG-driven zonal spectrum and found that the zonal flows are enhanced in higher pressure and lower magnetic shear plasma, leading to a self-organized high energy state. Further, we found a new suppression mechanism of the long wavelength ITG mode by the small-scale ETG-driven zonal flows, namely the radially non-local mode coupling and the associated energy transfer. Besides the role of suppression of ITG turbulence, we have for the first time observed intermittent and/or bursting behavior in the ion heat transport originating from the complicated mutual interaction among ETG-driven zonal flows, ITG turbulence, and the associated ITG-driven zonal flows.

1 Introduction

It is widely recognized that various profile formation in tokamaks is tightly coupling with various radial electric fields and related flow generation\cite{1}. Besides neoclassically driven equilibrium flows, turbulence self-generated $E \times B$ zonal flows have been shown to play an efficient role in regulating the turbulence structure and suppressing the heat transport in magnetized plasmas\cite{2-4}. In tokamaks, there exist many types of fluctuation with a spatial scale from micro-scale electron gyro-radius to macro-scale machine size. Since the zonal flows are generated through nonlinear interaction of the turbulence, different scale zonal flows may be simultaneously generated in the plasma. Since the zonal flows are characterized by large spatial structures in both the toroidal and poloidal directions and long auto-correlation times, it is expected that the zonal flows generated by fluctuations on some scale may affect fluctuations on other different scales in addition to regulating their own fluctuation level \cite{5,6}. The interaction mechanism, nonlinear dynamics and/or turbulent transport affected by such small-scale flows are open issues. Note that a direct interaction has been investigated based on the viscosity damping mechanism\cite{7}.

In this paper, we exploit a new role of zonal flows in regulating different spatio-temporal scale turbulence and the relevant transport, and further aim at studying the indirect interaction between the different scale turbulences through zonal flows. As an example, here we investigate the ion temperature gradient (ITG) turbulence embedded in the micro-scale zonal flows driven by electron temperature gradient (ETG) turbulences based on our theory and gyro-fluid simulations. This topic involves the interaction between small-scale sheared flows, for example, with a spatial scale on electron gyro-radius or the collisionless skin depth, and the large-scale turbulence. Such effects may sensitively depend on the zonal flow spectrum, which is typically characterized by the auto-correlation length (or
typically the radial wavelength) and auto-correlation time. In order to identify the spectrum, at first we perform the gyro-fluid simulations of the ETG turbulence-zonal flows system. The ETG-driven zonal flows are believed to be generally very weak compared with the background turbulence and hardly work for suppressing the turbulent electron transport contrary to the ITG counterpart[8]. However, we found that weak magnetic shear induces the large amplitude zonal flows in steeper electron temperature gradient regime. As a result, the ETG turbulence-zonal flow system may be self-organized to a higher energy state with lower electron transport level. We have developed a theoretical model based on the modulational instability analysis and found that the weak shear is favorable to the zonal flow instability and higher saturation level.

Based on the knowledge of the ETG-driven zonal flows, we performed the ITG turbulence simulation including the effect from such micro-scale zonal flows. So far, discussions for flow shearing stabilization is focused on the sheared flows with \( k_x^{(e)} \leq k_x^{(turb)} \) and \( k_x^{(e)} \) are typical radial value number of zonal flows and turbulence, respectively)[9]. We have found a new suppression mechanism of the long wavelength ITG mode by the small-scale flows through the radially nonlocal mode coupling[6]. Besides the role of suppression of ITG turbulence, we have for the first time observed intermittent and/or bursting behavior in the ion heat transport originating from the complicated mutual interaction among ETG-driven zonal flows, ITG turbulence, and the associated ITG-driven zonal flows.

2 Zonal flow controlo by magnetic shear and electron transport reduction

The generation of ETG-driven zonal flows has been found to be a slower process[8]. However, here we demonstrate that the zonal flows can be enhanced by controlling the magnetic shear and pressure gradient. Our simulations are based on a 3-dimensional gyro-fluid electrostatic slab model with the Landau damping and proper adiabatic ion response[8,10], considering that the electron transport is essentially electrostatic even though the electromagnetic effect seems an important component [11,12]. In order to understand the electron transport with electron ITB, the simulations are focused on the experimental observations, which are characterized by the steep electron temperature gradient, \( \eta_e = L_n/L_Te \), and weak magnetic shear \( \dot{s} = L_n/L_s \) [13,14]. The typical parameters are \( \mu_s = \eta_s = \chi_s = 0.5, L_s = 100 \rho_e, L_n = 10 \pi \rho_e, L_T = 2 \pi L_n, m_{Max} = 24 \), and a periodic (twisting) boundary condition is employed in the radial direction.

Figure 1 shows the results illustrating the time history of electron heat diffusivity \( \chi_e \) and corresponding zonal flow energy \( < |d\phi_{(e)}/dz|^2 > \) for different magnetic shears in the case of \( \eta_e = 6 \). ETG fluctuations exponentially grow up and surely saturate at different levels with corresponding zonal flows as the magnetic shere decreasing. In the moderate
magnetic shear case, i.e. $\hat{s} = 0.4$ and 0.2, the zonal flows are weak compared with the dominant turbulent component and has no effect on the saturation and suppression of the ETG turbulence. When the magnetic shear is further weakened to $\hat{s} = 0.1$, the linear ETG-drive and then the saturation level of the fluctuations increase. However, after the saturation, the heat diffusivity is found to be gradeally decreased, reaching to a lower quasi-steady state level. The tendency is more prominent in the case of $\hat{s} = 0.05$. Noticeably, the zonal flows undergo a growing phase (shadow part) in accordance with the decrease of the heat diffusivity, approximately exponential, after the saturation. This growing phase shows a zonal flow instability, namely, turbulent fluctuations could be converted to large amplitude zonal flows. The turbulence $(k_x, k_y)$ and zonal flow $k_{z}^{(e)}$ spectra are shown in Fig. 2. It is clearly seen that the zonal flow spectrum reveals a narrow peak around $k_{z}^{(e)} = 0.2 - 0.3$ with high amplitudes, which are almost same level as that of background turbulences, implying that the turbulences are strongly self-regulated by the zonal flows. The frequency power spectra of the zonal flows are also shown in Fig. 3 for moderate and weak shear cases, i.e. $\hat{s} = 0.4$ and $\hat{s} = 0.1$. The power spectra show a strong peak at $\omega = 0$ with the typical auto-correlation frequency $\Delta \omega$ that is much lower than the electron diamagnetic frequency $\omega_{ce}$, typically $\Delta \omega / \omega_{ce} \leq 0.01$. It is also found that the auto-correlation time becomes longer for the weaker magnetic shear, suggesting that the coherence of the zonal flows is more pronounced at higher amplitude regime.

We now present a simple analysis based on Hasegawa-Mima (H-M) equation [15]

$$
(1 - \nabla^2) \partial_t \phi = \partial_y \phi + [\phi, \nabla^2 \phi],
$$

which can demonstrate that the weak shear is favorable to the zonal flow instability. Zonal flows can be generated in turbulence only through the nonlinear interaction [15-17]. Previous modulational analyses were mainly based on an universal treatment by assuming a monochromatic wave packet. Actually, the different turbulence is characterized by the specific eigen-mode structure and fluctuating property, which involve the essentially parametric dependence. Hence, we should take the slab ETG eigen mode as a pump to drive zonal flows. In a general slab drift wave theory, the eigen structure of the lowest order radial mode in the fluid limit is described as $\phi(x) \sim \exp(-i \sigma z^2)$ (here $\sigma = L_{\phi} |s|/2 \Omega R q$) with the normalized (by $\omega_{ce}$) complex eigen value $\Omega \sim (-1, 1) \sqrt{\hat{s}}$ for given $\eta_e$ and $k_y$. 

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FIG. 2. Instantaneous spectra of both ETG turbulence and zonal flows at $t = 600$ in the case of $s = 0.1$ and $\eta_e = 6$.

FIG. 3. Frequency power spectrum of zonal flows for different magnetic shears in the case of $\eta_e = 6$. 

---
FIG. 5. Scan of zonal flow energy (a) and electron heat diffusivity \( \chi_e \) (b) for different \( \eta_e \) and \( \hat{s} \).

This pumping wave is written as [8]

\[
\phi_p(t, x, y) = \phi_0(t) \exp(-i\sigma x^2) \cos(k_y y) .
\]

(2)

It seems difficult to get an analytical dispersion relation and evolution equation of zonal flow instability similar to the previous derivations [8,16] due to the radial mode structure of pumping wave. Considering that the modulational processes are the same, we could use a summed dispersion relation for all \( k_x \) components and given zonal flow \( k_q \) [8]

\[
\omega^2 = \sum_{k_x} \frac{k_x^2 k_q^2 (k_x^2 + k_q^2 - k_y^2)(3k_x^2 - k_y^2 - k_q^2) \phi_x^2}{4\Lambda_y [\Lambda_y + (k_x + k_q)^2][\Lambda_y + (k_x - k_q)^2]} ,
\]

(3)

with \( \Lambda_y = 1 + k_y^2 \) to approximately understand the shear dependence of zonal flow instability. The increase of the radial mode width as the shear decreasing corresponds to a shrinking structure in Fourier \( k_x \) space. Small \( k_x \) can contribute to \( \omega^2 < 0 \), namely, excite zonal flow instability. We have directly calculated the H-M equation (1) by employing the pumping wave, Eq.(2). The modulational analysis can be also applied to the saturation mechanism of zonal flows. In order to probably compare the saturation levels of zonal flows for different shears, we fix the same total energy \( \int dx \, dy |\phi|^2 + (\nabla \phi)^2/2 \) of pumping waves in the calculations. Fig.4 plots the initial evolution of zonal flows for different shears. A zonal flow instability can be excited as the shear decreases. This is in well agreement with the above simple analysis. Further, the zonal flows also saturate at higher level as the shear becomes weak.

Combined with the results shown in Fig.1, the parametric scan of the magnetic shear and \( \eta_e \) dependence of the zonal flows and electron heat conductivities are studied as plotted in Fig.5. As \( \eta_e \) increases at the moderate magnetic shear \( \hat{s} = 0.2 \), the heat diffusivity increases together with the zonal flows, implying that the zonal flows are insufficient to suppress the electron transport. On the other hand, in the weak shear case around \( \hat{s} \lesssim 0.1 \), the heat diffusivity reaches the top at around \( \eta_e = 4 \) and then shows a flat or rather decreasing tendency as \( \eta_e \) increases, exhibiting some transition nature. Note that the total fluctuation energy, i.e. the sum of turbulence part and zonal flow part keeps roughly constant or rather increasing, but the rate between turbulence part and zonal flow part is reversed, so that the turbulent dominated plasma is self-organized to a lower transport state with a higher zonal flows level.
This picture may be helpful to understand the characters of electron ITB behaviour observed in tokamak discharges with pure electron heating [13,14], although there also exist other mechanisms like MHD stability and TEM for the electron ITB physics [18,19]. The steeper electron temperature gradient, which may be locally driven by ECRH or other electron heating methods, can excite stronger ETG fluctuation. When the central $q$ profile is controlled to become flat or reversed, the large amplitude zonal flows can be generated by the ETG turbulence and reduce the turbulences and electron heat transport.

3 ITG turbulent dynamics embedded in ETG-driven zonal flows

Interaction mechanism: From the analysis in Sec.2, large zonal flow potentials such as Mach number being about 0.5% with long-lived coherent structure have been observed. Those zonal flows developed with small spatio-temporal scale may be regarded as quasi-steady sheared fluid flows for ITG fluctuations. They can interact with large-scale turbulence on a gyro-phase averaged and velocity space averaged level from the view of fluid. Ignoring the weak time dependence, the coherent structure of the small-scale zonal flows are typically modeled by a circular function(sine or cosine) as follows

$$v_{\perp p}(x) \propto d\phi_{\perp x}(x)/dx = A(k_{\perp x})\cos(k_{\perp x}x),$$

(4)

when they are externally embedded in ITG fluctuations. Here, $k_{\perp x}$ is the normalized wave-number of small-scale flows by ion gyro-radius, which is larger than 1. It is assumed that the factor $A(k_{\perp x})$ has included both gyro-phase and velocity space averaging for the convenience of fluid treatment, in which $k_{\perp x}$ dependence should be involved in Bessel functions. A rigorous gyro-kinetic integral calculation was done in Ref.[5].

Analyses and simulations are based on a gyro-fluid model of electrostatic slab ITG turbulence [6,8], which includes the small-scale flows as an external source by adding $\partial_x \phi_{\perp x} \partial_y \int$ term to the corresponding moment equations. For usual ITG-driven zonal flows with $k_{\perp x} \leq k_{\perp x}^{(ITG)}$, the Doppler shift dominantly leads to local shearing of fluctuating potential structures. However, a different interaction mechanism for small-scale zonal flows can be advisedly revealed by a perturbation analysis. Keeping the lowest order effects of a small amplitude flow and Fourier transforming the perturbed eigen-mode equation from real space to wave-number $k$ space, i.e., $\phi(x) = \phi_k \exp(-ikx)$, it can formally yield

$$(\mathcal{L} + U_k)\phi_k = \Lambda \theta(\Lambda - \mathcal{L})(\phi_{k+k_{\perp x}} + \phi_{k-k_{\perp x}}).$$

(5)

Here, $\mathcal{L} \equiv d^2/dk^2$, $U_k = (L_n \Omega / L_m)^2(k_y^2 - (1 - \Omega)/(\Omega + K) + k_z^2)$, $\Lambda = L_n^2 \Omega^2 (1 + K)/2L_m^2(\Omega + K)^2$, $\theta = L_m/\Omega$ with $\Omega = \omega/\omega_e$ and $K = 1 + \eta$. This is a coupling equation system of different radial components $k$, $k + k_{\perp x}$, and $k - k_{\perp x}$. It shows that small-scale sheared flows can produce radial mode coupling of different harmonics, which is formally similar to the well-known poloidal coupling of drift wave in a toroidal configuration. However, the new feature of this model is that the coupling is non-local in the radial spectral space due to $k_{\perp x} > 1$, generally. It may directly transfer the fluctuating free energy from the unstable longer wavelength region, to stable or damped components at shorter wavelengths. Then, the ITG mode is expected to be stabilized. This non-local coupling sensitively depends on two factors: 1) the intensity of small-scale flows including gyro-phase averaged effects, which may strongly reduce the coupling intensity due to in the Bessel function dependence of $k_{\perp x}$, i.e., $A(k_{\perp x}) \sim A_0 \Gamma_0^{1/2}$, $\Gamma_0 = \Gamma_0(k_{\perp x}^2) \exp(-k_{\perp x}^2)$ [20], and 2) the spectral structure of ITG fluctuations between different radial modes $k$, $k + k_{\perp x}$, and $k - k_{\perp x}$. The latter means that the smaller the decay rate of ITG turbulence spectrum is in the inertial range, the
stronger the coupling is for a given $k_{cz}$. It can be expected that the zonal flows generated by the mesoscale turbulence on the collisionless skin depth size may more effectively interact with ITG turbulence.

Nonlinear simulations have been performed by initially including small-scale flows for different flow intensities or wave-numbers $k_{cz}$ [6]. The typical parameters are $\eta_i = 2.5$, $\hat{s} = L_n/L_s = 0.2$, $\mu_1 = \eta_1 = \chi_1 = 0.5$, $L_s = 50\pi_i$, $L_{d_s} = 10\pi\rho_i$, $L_z = 2\pi L_n$, $m \leq 15$. Figure 6(a) shows the time evolution of the space averaged energy-like quantity $\langle \phi^2 \rangle / 2$ in the earlier linear phase and the corresponding instantaneous radial spectra without and with small-scale zonal flows. An initial slowdown of the time evolution of the potential fluctuations is observed because of the stabilization role of small-scale flows. Most importantly, a spectral prominence with width $\Delta k_z \sim 2$ clearly appears near $k_z = k_{cz}$ as well as another one near $k_\phi = 2k_{cz}$, as shown in Fig.6(b). The monotonic decay spectrum is broken down at shorter wavelengths due to the radially non-local mode coupling. The width of the spectral prominence mirrors the fact that the unstable ITG mode stands in the range $k_z \leq 1$. Therefore, we conclude that small-scale zonal flows interact with large-scale ITG modes dominantly through the radially non-local mode coupling rather than the usual shearing decorrelation.

Ion transport intermittency: Our nonlinear simulations are designed to explore the role of small scale flows in ion heat transport. For a strong ITG turbulence drive, for example, $\eta_i \geq 4$, simulations show that even for the strongest small-scale flows, the small-scale zonal flows add less effects to the saturation level and the nonlinear evolution of ITG turbulence. As the turbulence drive is reduced, such as to $\eta_i \leq 2.5$, the corresponding levels of ITG fluctuations and self-generated zonal flows decrease much, as shown in Fig.7. Meanwhile, a remarkable intermittent or bursting behavior of ion heat transport appears, accompanied by the intermittent ITG-generated zonal flows with a time lag. It is also observed that the turbulence intensity $\langle \phi^2 \rangle / 2$ and ion heat conductivity $\chi_i$ are in phase during bursts [21]. The bursting period becomes longer, even infinite (it means no linearly unstable ITG modes) as the ETG-driven flows increase, as shown by solid curves.
in Fig. 7. The time-averaged transport becomes decreasing for stronger flows. We next performed simulations by artificially excluding ITG-driven zonal flow components in order to find the related factors for the intermittency occurrence. The transport levels become about one order higher than their counterparts above and no any bursts are observed, as shown by the dashed curves in Fig. 7. It is clear that the ITG-driven zonal flow dynamics still dominates the ITG turbulent transport in the relatively weak turbulence, but the turbulence can be remarkably modulated by small-scale flows. The emergence of ion transport intermittency requires the simultaneous presence of both small-scale flows and ITG-driven zonal flows. The causal relation between the turbulent transport and ITG-generated zonal flows during bursts is plotted in Fig. 8(a) for the case with $A(k_{ex}) = 1.4$ in Fig. 7. The bursting process can practically last long time (we have calculated to $t = 3000$). How these small-scale zonal flows lead to an intermittent behavior in ITG turbulence is a key question. Note that the turbulent fluctuations seem to roughly exponentially go up and down during bursts, as shown in Fig. 7. Performing spectral analyses for ITG turbulence, we found that at bursting peaks, $k_x$ spectra are characterized by a monotonic decay structure with an approximate power law, which is a typical nonlinear Kolmogorov-type scaling. However, the spectral structures near $k_x = k_{ex}$ and $k_x = 2k_{ex}$ are gradually deformed after the bursts, actually degenerated to the linear spectral shape at valleys, as shown in Fig. 8(b), which is dominated by the linear nonlocal mode coupling. The bursts emerge in the recovery phases of nonlinear saturation spectra. The physical mechanism of the intermittency may be understood as follows. At first, the exponentially growing ITG fluctuation is initially slowed down by small scale flows through nonlocal mode coupling, and saturated by self-generated zonal flows as well as the convective nonlinear coupling. Afterward, ITG fluctuation decreases due to the stabilization role of the micro-scale zonal flows, as a result of competition between nonlocal mode coupling and nonlinear inverse cascading. ITG-driven zonal flows follow the turbulent decreasing with a time log behind due to nonlinear drive decreasing. During this phase, the nonlinear turbulence spectrum is alternated to a linear deformed structure. When the effective shearing rate of ITG-generated zonal flows becomes lower than the turbulence decorrelation rate, the ITG fluctuations linearly grow up again and then lead to a burst.
4 Conclusion

In conclusion, our numerical experiments and analyses have shown that a large amplitude zonal flow can be excited in strong ETG turbulence with weak magnetic shears. The electron turbulence can be self-organized to a higher confinement state with reduced heat transport. It suggests a probably controllable method of zonal flows in tokamak discharges by adjusting $q$ profile. Note here that the Kelvin-Helmholtz instability excited in small $k_{\parallel}$ region is one of the candidate to suppress the strong zonal flows in additional to the modulational spectrum change and further investigation is necessary by increasing the resolution in the poloidal direction[5]. Based on the knowledge of the ETG-driven micro-scale zonal flows, we have established a theoretic model on the interaction between different spatio-temporal scale fluctuations through zonal flows. A key physical mechanism of the interaction is found to be a radially non-local mode coupling between unstable and stable or damped components. It can deform the nonlinear monotonic decay spectrum of large-scale ITG turbulence in the inertial range and lead to an intermittent or bursting behavior of turbulent ion transport. These results open up a new paradigm where mutual interactions in the broad dynamic range of fluctuation spectrum including zonal components may provide crucial roles.

The authors thank to Dr. M.Azumi, Profs. M.Yagi, K.Itoh and J.Q.Dong for their fruitful comment and discussions.

References

Interaction among different spatio-temporal scale fluctuations through zonal flows

Y. Kishimoto, J. Qi, L. Y., Y. Idomura, and A.I. Smolyakov

1. Naka Fusion Research Establishment, JAERI
2. University of Saskatchewan, Canada
3. Southwestern Institute of Physics, P.R. China

Acknowledgement:
M. Azumi (JAERI), M. Yagi (Kyushu Univ.), K. Itoh (NIFS)

Contents
- Background and motivation
- Zonal flow characteristics in ETG turbulence
- Ion transport modulation by small scale flows
- Conclusion

Background and motivation

Radial electric fields and flows → Various structure formations

- Neo-classically driven equilibrium flow
- Turbulence driven zonal flow (Hasegawa and Wakatani, PRL '87)
- Self-regulate and suppress transport level
  (I. Lebedev and diamond, PoF '95)
- Interaction among different scale turbulence
  - Direct interaction through eddy viscosity damping
    (Itoh and Itoh, PPC '02; TII/1-4: Yagi, et al.)
  - In-direct interaction through zonal flows
    (Idomura, et al. IAEA '00)

Indirect interaction between ETG and ITG

- ITG zonal flows
  → ETG turbulence

- Suppression by flow shear
  (Kim et al., PoP '96)

- ETG zonal flows
  → ITG turbulence

- No well developed theory
  (cf. Resonant detuning
  Idomura, et al. IAEA '00)

- Zonal flow spectrum: S(q,Ω)
- Response to ion transport

ETG-driven zonal flow spectrum (1)

Modulational instability analysis: 3 and 5 fields H-M model

- Slow or marginal process

- Instability increases in small ks regime
  (Shib ETG model)

- Zonal flow instability
  in weak magnetic shear regime
  → exponential growth
  → higher saturation level
ETG-driven zonal flow spectrum (2)

- Gyro-fluid simulation with strong ETG-drive: \( \eta \approx 6 \)
  (temperature gradient with ITG closure model)
- Weak magnetic shear regime
- Zonal flow instability after saturation and electron transport reduction
- Reduction of high frequency component
  - Quasi-coherent structure

Self-organization of ETG turbulence-zonal flow system

- \( A \times \) ETG-drive increase
- \( z \)-flux: zonal flow increases
  (insufficient to suppress transport)
- Flat-decreasing tendency of heat flux
  (suppression by zonal flows)
- Change of ratio: Turbulent fluctuation
  / Zonal fluctuation
- Self-organized to high-energy state
  with enhanced zonal flows

Coupling of ITG mode with small-scale flows

- Linear Fourier-transformed eigen-mode equation: \( \kappa \)-space
  \( \delta \phi_x(x) = \frac{\partial \delta \phi_y(y)}{\partial x} x = A.q(q) \omega(q) \eta \)
  : coherent micro-scale flow
- Unstable flow
  \( \kappa_0 \) (Two modes)
  \( \lambda(x_0) - \lambda_0.1, \eta_{\phi}^2(q_0) - \eta_{\phi}^2(q_0) \)
  : zonal-phase average
  \( D_{\phi} \), \( D_{\phi} \)
- Non-local mode coupling
- New energy transfer channel
- Reduction of ITG growth rate
- New non-linear dynamics

Ion transport modulation by small-scale flows

- Weak ITG-drive: \( \eta \approx 2.5 \)
- Intermittency in turbulence zonal flow system (out of phase)
  - Time averaged ion transport reduction
- Increase of flow amplitude
  - Increase of bursting period
- Two phases exist
  - Upper state phase
    \( - \): Level \( w \) micro-flow
  - Suspension phase
    \( + \): Laminating/flowing phase
- Intermittency requires both
  - Micro-scale flow
  - ITG-driven zonal flow
Conclusions

- We discussed a possibility of indirect coupling between different scale fluctuations through micro-scale zonal flows as an example of ITG and ETG turbulences.

- ETG-driven zonal flows are enhanced in higher pressure gradient and lower magnetic shear regime, leading to a self-organized high-energy.

- Radial non-local mode coupling and associated energy transfer is one of suppression mechanisms of ITG turbulence by micro-scale zonal flows.

- New energy transfer channel of turbulences by micro-scale zonal flows in wide wave-number space leads to a dynamical behavior, i.e. intermittency, besides suppression role of the transport level.
1.17 Long Time Scale Plasma Dynamics Driven by the Double Tearing Mode in Reversed Shear Plasmas

Y.Ishii,1) M.Azumi,1) Y.Kishimoto1) and J.N.Leboeuf 2)

1) Naka Fusion Research Establishment, JAERI, Naka, Ibaraki 311-0193, Japan
2) Department of Physics and Astronomy, University of California at Los Angeles, Mira Hershey Hall, Los Angeles, CA, USA

E-mail: ishiiy@fusion.naka.jaeri.go.jp

Abstract. The new nonlinear destabilization process is found in the nonlinear phase of the double tearing mode (DTM) by using the reduced MHD equations in a helical symmetry. The nonlinear destabilization causes the abrupt growth of DTM and subsequent collapse after long time scale evolution in the Rutherford-type regime. The nonlinear growth of the DTM is suddenly triggered, when the triangular deformation of magnetic islands with sharp current point at the x-point around the outer rational surface exceeds a certain value. Such structure deformation is accelerated during the nonlinear growth phase. Decreasing the resistivity increases the sharpness of the triangularity and the spontaneous growth rate in the abrupt growth phase is almost independent on the resistivity. Current point formation is also confirmed in the multi-helicity simulation, where the magnetic fields become stochastic between two rational surfaces.

1 Introduction

The formation of the non-monotonic safety factor (q-) profile, or the reversed shear profile, is considered to be one of the attractive methods to attain high performance steady state operation of a tokamak. The MHD stability for this profile is one of important issues to be theoretically clarified for the development of a steady state tokamak. The plasma with this q profile can be linearly unstable against resistive modes even in a low-beta state. In high $\beta$ region, the resistive interchange mode becomes unstable around the inner rational surface [1]. In low $\beta$ region, however, there is a possibility that the double tearing mode (DTM) becomes unstable even when the Mercier criterion for the resistive interchange mode is broken. Figure 1 shows the eigen mode structure, $V_r(m/n = 3/1)$, obtained by the resistive MHD analysis for the reversed shear equilibrium based on the JT-60U reversed shear discharge [2,3]. The eigen mode extends between two q=3 rational surfaces and has the odd parity around each rational one, which means this mode is the double tearing one. In some situations, DTM shows the large growth rate of the resistive internal mode and can drive the plasma to termination almost exponentially in time with the linear growth rate. The nonlinear behaviors of the double tearing mode have been intensively studied by some authors through MHD simulations [4,5]. Recently, we found the new phenomena of the double tearing mode in the nonlinear phase; that is, when two resonance surfaces are apart from each other, the mode gently grows magnetic

FIG. 1. q-profile (marked solid line) and the eigen mode structure of $V_r(3/1)$ (solid line) obtained by the resistive MHD analysis
islands at each resonance surface like in Rutherford regime of the conventional tearing mode [6, 7] but it suddenly shows the rapid growth after both magnetic islands grow enough to interact with each other [8]. The remarkable feature of this new phenomenon is the weak dependence of the mode growth rate on the resistivity \( \eta \) in the explosive growth phase [9]. It must be noted that this process is observed in a plasma with helical symmetry, where all harmonics have the resonance surfaces at the same radius, so that the newly observed phenomena seems to be very different from any theories proposed so far like the nonlinear coupling among different helicities and also the destabilization through the renormalized turbulence transport process, which have been observed in MHD simulations of the major disruption [10, 11]. This process is very important because, even if the plasma safely passes the regime unstable against the conventional double tearing mode with assistance of the magnetic well or the detail current profile control, the slowly growing tearing-like modes can be suddenly destabilized by the nonlinear process and lead to the reconstruction of the current profile to the monotonic profile. The purpose of this paper is to show the details of this nonlinear destabilization of double tearing mode.

2 Model and Simulation Results

We employ the reduced set of resistive MHD equations in cylindrical plasma with helical symmetry and solve them by the finite difference in the radial direction and Fourier expansion in the angular directions [12].

\[
\frac{\partial u}{\partial t} = \frac{1}{r}[u, \phi] + \frac{1}{r}[\psi, j] + \frac{B_0}{R_0} \frac{\partial j}{\partial \phi} + \nu \nabla^2 u
\]

\[
\frac{\partial \psi}{\partial t} = \frac{1}{r} [\psi, \phi] + \frac{B_0}{R_0} \frac{\partial \phi}{\partial \phi} + \nu j - E
\]

\[
j = \frac{\partial^2}{\partial r^2} \psi + \frac{1}{r} \frac{\partial}{\partial r} \psi + \frac{1}{r^2} \frac{\partial^2}{\partial \phi^2} \psi
\]

\[
u = \frac{\partial^2}{\partial r^2} \phi + \frac{1}{r} \frac{\partial}{\partial r} \phi + \frac{1}{r^2} \frac{\partial^2}{\partial \phi^2} \phi
\]

\[(a, b) = \frac{\partial a}{\partial r} \frac{\partial b}{\partial \phi} - \frac{\partial b}{\partial r} \frac{\partial a}{\partial \phi}
\]

The safety factor profile used in the following is

\[
q(r) = q_e \{1 + \left(\frac{r}{r_0}\right)^{2\lambda} \}^{1/2} \{1 + A \exp\{-(\frac{r - r_s}{\delta})^2\},
\]

\[
\psi(r) = -\frac{B_0}{R_0} \int_0^r \frac{r dr}{q(r)}.
\]

Here, \( \psi \) is the poloidal flux function, \( \phi \) is the stream function, \( \eta \) is the resistivity, \( \nu \) is the viscosity, \( j \) is the toroidal current density, \( u \) is the vorticity, \( E \) is the electric field at the wall, \( B_0 \) is the toroidal magnetic field, \( R_0 \) is the major radius and \( \perp \) means the derivative perpendicular to the magnetic field. In these equations, uniform plasma density is assumed and the time is normalized to the poloidal Alfvén transit time \( \tau_{pa} = \sqrt{\rho a / B_0(a)} \) (\( \rho \) is the plasma mass density, \( a \) is the plasma minor radius and \( B_0(a) \) is the poloidal magnetic field at the plasma surface). The resistivity \( \eta \) is normalized such that \( \eta = \tau_{pe}/\tau_\eta \), where \( \tau_\eta \) is the plasma skin time. The rationalized MKS unit is used. The magnetic field and the velocity field are related to the poloidal flux \( \psi \) and the stream function \( \phi \) by \( \vec{B} = B_0 \vec{e}_\phi + \nabla \psi \times \vec{e}_\phi \).
and \( \vec{V} = \nabla \phi \times \hat{e}_\phi \), where \( \hat{e}_\phi \) is the unit vector in the toroidal direction. In the following 3 sections, we consider only the MHD activity with helical symmetry of \( f(r, \theta, \varphi) = f(r, \zeta = \theta - (n/m) \varphi) \), where \( m \) and \( n \) are poloidal and toroidal mode numbers of the MHD mode, respectively, and the helical flux function is defined as \( \psi^\ast(r, \zeta) \equiv \psi(r, \zeta) - (r^2/2)(n/m) \).

We fix the parameters \( \lambda = 1, r_0 = 0.412, \delta = 0.273, r_\delta = 0 \) and \( A = 3 \) through this paper and change only \( q_0 \), which changes the distance of two resonance surfaces, \( \Delta r \). Also, we fix the poloidal/toroidal mode numbers \( (m/n) \) to 3/1, respectively. The maximum number of the Fourier components of the mode is taken to be 100 and the maximum number of equally spaced radial grid is 1600, in order to reproduce fine structures.

The linear stability analysis against the resistive mode in this q-profile shows that, as increasing \( \Delta r \), the exponent factor \( \alpha \) of resistivity \( \eta \) with respect to the growth rate \( \gamma(\propto \eta^\alpha) \) changes from \( \alpha = 1/3 \) of the resistive internal mode in the limit of \( \Delta r = 0 \) to \( \alpha = 3/5 \) of the conventional tearing mode in the limit of \( \Delta r = \infty \) [13]. The dependence of \( \alpha \) on \( \Delta r \) is shown in Fig.(2) for the q profile of Eq.(6). Corresponding to this change of the linear stability, the nonlinear behavior of the mode also changes. That is, in the small \( \Delta r \) region, the mode evolves exponentially with the linear growth rate, and the fundamental mode is essential in this behavior, while, in the large \( \Delta r \) region, the mode enters the Rutherford regime and the magnetic islands saturate at each resonance surface. The nonlinear behavior has been considered smoothly to transit from the exponential growth to the island saturation so far. The new type of the nonlinear instability is found in this midway of these nonlinear behaviors of DTM, as shown in the shaded area \( (0.22 < \Delta < 0.31) \) in Fig.2.

The typical examples of the temporal evolution of magnetic and kinetic energies of this new phenomena for the cases (B), \( \Delta r = 0.285 \), and (C), \( \Delta r = 0.310 \), are shown in Figs.3. After the exponential growth in the linear regime, the mode reduces its spontaneous growth rate and tends to enter the Rutherford-type regime. In this phase, the kinetic energy almost saturates, while the magnetic energy continues to increase with reduced temporal rate and magnetic islands grow in the resistive time scale. Then, after magnetic islands growing around each resonance surface to contact with each other, the mode shows the abrupt growth. In this phase, the inner islands are expelled outside the outer ones and squeeze. The outer islands cover the almost whole region between two rational surfaces and the averaged q-profile becomes flat including the magnetic axis for this case. This may be the plasma collapse or disruption for low beta negative shear plasmas. For the case (C), the simulation results for \( \eta = 5 \times 10^{-6} \) are also plotted in Fig.3(c) and (d). It is clearly shown that the kinetic energy quasi-saturation regime for \( \eta = 5 \times 10^{-6} \) is about two time longer than that for \( \eta = 1 \times 10^{-5} \). This is the same as the Rutherford regime in the conventional tearing mode.
3 Nonlinear Mode Coupling Effects

In order to study the origin of this abrupt growth of DTM during the nonlinear phase, we have performed several simulations. One of the possible candidates for this destabilization is the quasi-linear modification of the q profile and the acceleration of the linear instability. The simulation, where the perturbations set to zero on the way of the abrupt growth and the small one is set on the main (m/n = 3/1) harmonics, shows that the mode returns back to the linear growth phase and again enters the Rutherford-type regime. This means that the modified q-profile does not destabilize the mode in the linear stability sense, including any other higher harmonics. In this way, the quasi-linear modification of the q-profile cannot reproduce the abrupt growth of the mode. This was also confirmed by the comparison of simulations with reducing the maximum number \( l_{\text{max}} \) of Fourier mode. In Figs. 3(a) and (b), time traces for different numbers of \( l_{\text{max}} \) are also plotted. Figures 3(a) and (b) show that the temporal evolution of the mode before the abrupt growth is not sensitive so much on \( l_{\text{max}} \), while the behavior of the abrupt growth strongly depends on \( l_{\text{max}} \); that is, reducing \( l_{\text{max}} \) from some critical number, \( l_c \), the growth becomes more gentle. On the other hand, the simulations with \( l_{\text{max}} \) greater than the critical number \( l_c \) give almost the same result. The critical number \( l_c \) depends on the distance between resonance surfaces, \( \Delta r \), and it is \( l_{\text{max}} = 20 \) for the typical example in Figs. 3(a) and (b). Beyond the upper boundary of the shaded region in Fig. 2, the mode does not show any nonlinear destabilization, even if the number of Fourier harmonics is increased. At the boundary of \( \Delta r = 0.315 \), the outer separatrix of the inner islands and the inner separatrix of the outer ones reach the same radial position. The nonlinear destabilization of DTM, however, does not occur, which means that the interaction of the inner and outer islands is not the sufficient condition of this phenomenon. These simulations clearly show that the abrupt growth of DTM after the Rutherford-type phase is induced by the nonlinear coupling among the higher harmonics, although the harmonics higher than some critical number do not play an essential role in this process. In the case (C), the nonlinear destabilization is triggered for \( l_{\text{max}} = 7 \), but not for \( l_{\text{max}} = 6 \). This corresponds to the fact that the degree of the island deformation, or the formation of the sharp triangular edge, is important for triggering the nonlinear destabilization, as shown in detail later.

Next we move to the details of the nonlinear destabilization. It is interesting to know whether the magnetic perturbations \( \psi_{\geq 1} \) or the kinetic ones \( \phi_{\geq 1} \) are the key factor of the nonlinear destabilization. For this purpose, we reset magnetic or kinetic perturba-
tions to zero on the way of the abrupt growth and investigated the subsequent phenomena. Results are shown in Figs.4. In simulations resetting the kinetic perturbations to zero, the kinetic perturbations recovered to the same level as the original ones in a very short time and shows the abrupt growth (Fig.4(a)), while in the case of resetting the magnetic perturbations to zero, the abrupt growth is not reproduced (Fig.4(b)). For the case retaining fundamental magnetic perturbation (i.e. $\psi_{12} = 0$), the mode resumes the abrupt growth after the higher harmonics of magnetic perturbations grow up to sufficient amplitudes through the mode coupling. This comparison confirms that the nonlinear destabilization originates from the coupling among magnetic perturbations through $J \times B$, not from the driven reconnection type instability. By considering the quasi-linear and the nonlinear mode coupling effects, it was shown that the higher harmonics are important for the abrupt growth of DTM, and they are produced from the magnetic harmonics, $\psi_l$.

4 Current Point Formation

The growth of the mode pushes the inner magnetic islands toward the separatrix of the outer magnetic islands and generates the skin current along the separatrix surface. This skin current prevents the further growth of the mode and leads to the saturation of the mode. This is the nonlinear behavior of the standard DTM with small $\Delta r(< 0.22$ in Fig.2). Contrary to this, in the case of the nonlinearly destabilized DTM, the further growth of the magnetic island increases the triangular deformation of the island shape and forms the skin current highly concentrated to the X-points of the outer magnetic islands. This difference of the nonlinear behaviors between the standard DTM and the nonlinearly destabilized DTM is shown in Figures 5, where the contours of the helical flux surfaces, $\psi^s$, the flow potential, $\phi$, and the toroidal current excluding the fundamental harmonics, $j_{10}$, in the nonlinear phase are plotted.

Figure 5(a) and (b) shows the case of the standard DTM. The mode grows exponentially with the linear growth rate and the convective force to push the magnetic flux is larger than the magnetic reconnection rate. Then, the separatrix of the inner island is uniformly pushed toward the outer islands. This changes the reconnection region from the X-point type to the Y-type layer with skin current flowing along the finite distance as shown in Fig.5(b) [14]. In contrary to this, in the case of the nonlinear destabilization of DTM, there are quadruple vortices and the magnetic flux sustains the X-point structure as shown in Fig.5(c). In this case, the mode enters the Rutherford-type regime growing slowly proportionately to the resistivity, $\eta$. During this phase of weak convection, the mode coupling generates the higher harmonics and deforms the magnetic surface to the sharp triangularity. As the result, the plasma current concentrates in the small region and forms the current point, which is a current sheet with very short width, as shown in Fig.5(d). In this way, the increase of the triangular deformation of magnetic islands and the resultant localization of the skin current to the x-point is the key factor of this new

FIG. 4. Time evolutions of magnetic and kinetic energies of 3/1-mode for the standard and restarted simulations: (a) perturbations of $\phi(l > 0)$ are set to zero, (b) perturbations of $\psi(l > 0)$ are set to zero.
process. From this feature, this can be said the structure driven mode. A remarkable feature of this structure driven mode is the dependency of the spontaneous growth rate, $\gamma_{\text{temp}}$, on the resistivity, $\eta$; that is, the dependence of $\gamma_{\text{temp}}$ on $\eta$ in the explosive growth phase is very weak, $\gamma_{\text{temp}} \sim \eta^\alpha$, $\alpha \approx 0$, as shown in Fig.6. This interesting feature of the nonlinear process was confirmed by simulations showing that the result does not change by the increasing the numbers of radial grid and Fourier harmonics. It is also noted that, by reducing the resistivity, the current peak becomes sharp and high in almost inversely proportional to the resistivity. That is, the local quantities do change sensitive to the resistivity, while the evolution rates of the energies, or the integration quantities, do not change.

![Diagram](image1)

**FIG. 5.** Contours of the helical flux function, $\psi^*$, solid curves in (a) and (c), the flow potential, $\phi$, dotted curves in (a) and (c), and the current, $j_{z>0}$, solid curves in (c) and (d): (a) contours of $\psi^*$ and $\phi$ and, (b) contours of $j_{z>0}$ at $t=130$ for the standard $\text{DTM}(\Delta r = 0.115)$. (c) contours of $\psi^*$ and $\phi$ and, (d) contours of $j_{z>0}$ at $t=330$ for the nonlinearly destabilized $\text{DTM}(\Delta r = 0.285)$

![Diagram](image2)

**FIG. 6.** Time evolutions of the magnetic energies of 3/1-harmonics at the nonlinear destabilization phase of $\text{DTM}$ for the different resistivity.

5 Multi Helicity Simulation

In the above sections, the simulations have been carried out for the helical symmetry assumption with single helicity $m/n=3$, and have shown that the formation of the current point is essential for the destabilization. In the toroidal geometry, however, the different helicity harmonics are coupled with each other through the toroidal coupling. In some case, the mode with different helicity can be unstable simultaneously. These unstable or toroidally coupled modes form the magnetic islands and make the stochastic magnetic field through the island overlapping. This loss of the coherence may affect the formation of the current point and the nonlinear destabilization. In order to investigate this effect, we have done the nonlinear reduced MHD calculations in toroidal geometry, include the pressure effects. The pressure value is set as $\beta_p = 1.0 \times 10^{-6}$ on the magnetic axis, which is low enough so that the most unstable mode is not the pressure driven one but the double tearing one. Figure 7 shows the time evolution of the energies of the fundamental
harmonics, $m/n=3/1$. As shown in Fig.7, the essential feature of the nonlinear destabilization of DTM does not change in the toroidal multi-helicity case. Figures 8(a) and (b) show the poincare plots of the magnetic field $B$ and the plasma current at $t = 320$. As expected, as the magnetic energy increases, the stochastization of the magnetic field line rapidly expands between the inner and outer $q = 3$ rational surfaces and finally covers its whole region. The current point is, however, formed at the reconnection region for the $m/n = 3/1$ islands. This means that even under the stochastic magnetic fields, the current point is formed and causes the nonlinear destabilization of DTM in the toroidal geometry. The more detail analysis shows that the different helicity mode ($m/n = 8/3$) is also linearly unstable for this typical case and the coupling with this mode slightly enhances the destabilization of the fundamental mode and the timing of the abrupt increase of the energies becomes faster.

FIG. 7. Time evolutions of the magnetic and kinetic energies of 3/1-harmonics in the toroidal geometry.

FIG. 8. (a) poincare plot of the magnetic fields and (b) the local maximum and minimum of plasma current and the poloidal flux of $m/n=3/1$-mode (dotted line) in toroidal simulation at $t=320$.

6 Summary and Discussions

In summary, we have shown the new process of the nonlinear destabilization of DTM which can be caused in the reversed shear profile in a tokamak. It was found that the slowly growing DTM can be nonlinearly destabilized and changes to the explosively growing DTM. Moreover, the spontaneous growth rate at this explosive phase is almost independent on the resistivity, due to the efficient reconnection of the magnetic field. In this phenomenon, the formation of the current point during the long time scale evolution phase is the key process. We have also shown that the current point can be formed even under the stochastic magnetic fields in a toroidal geometry. In the recent large tokamak plasmas, the resistivity, $\eta$, becomes about $\eta \approx 10^{-8}$. Hence, after long term evolution of
DTM in the Rutherford type regime, the nonlinear destabilization of DTM occurs in the fast time scale.
In the case of the low $\beta$ disruption in a negative shear plasma, the perturbations growing with a resistive time scale are sometimes observed around each rational surfaces [2]. After the growth in the resistive time scale, the perturbation shows the explosive growth. These feature is roughly consistent with our observation of the nonlinear destabilized DTM, although, at the present stage, the relationship between the precursor with the resistive time scale and the fast time scale phenomenon is not clear in experiments.
Finally, the present paper clarified the new mechanism of the nonlinear destabilization of the MHD mode. The explosive growth of DTM was shown to be originated not from both any type of the quasi-linear destabilization and the turbulence driven instability, where the increase of the transport coefficients driven by the higher harmonics accelerates the growth of the mode. Instead of them, the increase of the triangular deformation of islands plays the key role of these new phenomena, where the skin current concentrates to the X-point, relaxing the excess magnetic energy effectively through dissipation, and the explosion is almost independent on the resistivity.

References

Introduction

1. Resistive time scale; precursors around inner and outer rational surfaces are observed before the plasma collapses in DTM discharges.

2. New type of DTM; nonlinear destabilization process.

3. New reconnection process.

4. Major minor collapses.

5. Time scale splitting in negative shear plasma collapse.
Conclusion and Remarks

1. Nonlinear destabilization phenomenon of DTM is found. Slowly growing DTM is nonlinearly destabilized and enters the explosive growth phase, where the spontaneous growth rate is almost independent on the resistivity.

   This phenomenon is roughly consistent to the time splitting phenomenon observed in the low beta collapse in JT-60U reversed shear plasmas.

2. New reconnection process is found. Triangular deformation of magnetic islands causes the current point, which enhances the reconnection rate during explosive growth phase of DTM.

Current Point Formation under Stochastic Magnetic Fields

- Multi helicity simulation in toroidal geometry (mode number=300)
- Stochastic B by islands overlapping with different helicities
- Current point formation and nonlinear destabilization of DTM

[Graphs showing magnetic energy and current point formation]
1.18 Gyrokinetic Global Analysis of Ion Temperature Gradient Driven Mode in Reversed Shear Tokamaks

Y. Idomura, S. Tokuda, Y. Kishimoto

Department of Fusion Plasma Research, Naka Fusion Research Establishment, Japan Atomic Energy Research Institute, Naka, Ibaraki, 311-0193, Japan

E-mail address of main author: idomuray@fusion.naka.jaeri.go.jp

Abstract: A new toroidal gyrokinetic particle code has been developed to study the ion temperature gradient driven (ITG) turbulence in reactor relevant tokamak parameters. We use a new method based on a canonical Maxwellian distribution \( F_{CM}(P, \epsilon, \mu) \), which is defined by three constants of motion in the axisymmetric toroidal system, the canonical angular momentum \( P \), the energy \( \epsilon \), and the magnetic moment \( \mu \). A quasi-balloonning representation enables linear and nonlinear high-\( m,n \) global calculations with a good numerical convergence. Conservation properties are improved by using the optimized loading method [2]. From comprehensive linear global analyses over a wide range of an unstable toroidal mode number spectrum \( (n=0-100) \) in large tokamak parameters \( (a/R_0=320-460) \), properties of the ITG modes in reversed shear tokamaks are discussed. In the nonlinear simulation, it is found that a new method based on \( F_{CM} \) can simulate a zonal flow damping correctly, and spurious zonal flow oscillations, which are observed in a conventional method based on a local Maxwellian distribution \( F_{LM}(\psi, \epsilon, \mu) \), do not appear in the nonlinear regime.

1. Introduction

For the purpose of studying the ion temperature gradient driven (ITG) turbulence, we have developed a new gyrokinetic toroidal particle code for a 3D nonlinear global simulation (GT3D). This code, which has been developed based on a finite element PIC method [1], has several new features which are essential for studying the ITG turbulence in reactor relevant tokamak parameters. Firstly, we have developed a new method based on a canonical Maxwellian distribution \( F_{CM}(P, \epsilon, \mu) \), which is defined by three constants of motion in the axisymmetric toroidal system, the canonical angular momentum \( P \), the energy \( \epsilon \), and the magnetic moment \( \mu \). In the system with \( F_{CM} \), a free energy related to \( \partial \mu \partial P F_{CM} \), which corresponds to the density and temperature gradients, does not drive any axisymmetric perturbations including zonal flows. However, in the system with a conventional local Maxwellian \( F_{LM}(\psi, \epsilon, \mu) \), which is defined by a flux label \( \psi \), spurious driving effects on axisymmetric perturbations exist. Although this driving effect is a higher order correction compared with the linear driving term for the ITG mode with \( n=0 \), it significantly affects on axisymmetric perturbations or zonal flows, where this ordering does not hold, and spurious zonal flow oscillations grow in simulations using \( F_{LM} \). Therefore, use of \( F_{CM} \) is important especially for studying zonal flows, which are closely related to the suppression of the ITG turbulence. Secondly, the conservation property of GT3D is greatly improved using the optimized loading [2]. An improvement of the conservation property not only demonstrates the validity of the simulation, but suppresses spurious \( E\times B \) flows which are generated also from a breakdown of the particle conservation. Thirdly, we use a quasi-balloonning representation, which enables linear and nonlinear global high-\( m,n \) calculations. This technique is important to study transport properties in recent advanced tokamak configurations, where the conventional kinetic ballooning theory breaks down around a transport barrier region or a weak magnetic shear region. Of course, in case of the nonlinear simulation, we can not use the kinetic ballooning theory. Finally, GT3D has been implemented successfully on the JAERI Origin3800 system. The code is highly scalable and it operates with 40% of processing efficiency up to 512 processors. GT3D has a capability of simulating large tokamak parameters such as \( a/R_0\sim 500 \).
2. Linear Gyrokinetic Global Analysis of ITG Modes in Reversed Shear Tokamaks

From comprehensive global analyses over a wide range of an unstable toroidal mode number spectrum \((n=0-100)\) in large tokamak parameters \((\alpha/\rho_i=320-460)\), it is found that especially in reversed shear tokamaks, properties of the ITG mode are drastically changed through ion heating and density peaking processes. When the ion temperature is sufficiently high, most unstable high-\(n\) modes are excited in the outside of the \(q_{\text{min}}\) region. Residual low-\(n\) global modes in the \(q_{\text{min}}\) region show slab like feature, and their growth rates decrease by a peaked density profile. In the present study, we have assumed a circular concentric tokamak; deuterium plasma, \(R_0=2.6\)m, \(a=0.94\)m, \(B_0=4.6\)T. Peak density and temperature gradients at \(r/a=0.5\) are given as \(L_n/R=0.1\), \(L_{\text{th}}/R=0.5\), and \(L_{\text{th}}/R=0.5\).

2.1 Relation between Radial Eigenmode Structure and \(n\) Spectrum

In Fig. 2, growth rates are plotted for the following three cases of equilibrium configurations; (a) normal shear, \(\alpha/\rho_i=320\) \((T_{\text{ei}}=26\text{keV}, T_{\text{ei}}=8\text{keV})\), (b) reversed shear, \(\alpha/\rho_i=320\), and (c) reversed shear, \(\alpha/\rho_i=460\) \((T_{\text{ei}}=13\text{keV}, T_{\text{ei}}=4\text{keV})\) (see Fig. 1). For the above devise size, the growth rate spectrum spreads over very high-\(n\) \((n=100)\) region. From a detailed study of the eigenmode structure, it is found that the higher-\(n\) modes are excited at the outer magnetic surface. This is because a radial position where the mode is excited is determined from a competition between the following two conditions; local gradient parameters and a destabilization condition in a wave number space, \(k_{\rho_i}=nq(r)/r_{\rho_i}(r)<0.5\). Especially in the latter condition, a geometry effect \(1/r\) and the finite Larmor radius effect \(\rho_i(r)\) are involved. In the reversed shear configuration, where \(q\) is almost constant in the \(q_{\text{min}}\) region, these effects become distinct and the high-\(n\) modes with \(n>20\) are excited in the outside of the \(q_{\text{min}}\) surface. On the other hand, in the normal shear configuration, where \(q\) increases monotonically as a function of \(r\), these effects are cancelled by a change of \(q\). In the lower ion temperature case (case (c)), the unstable region with \(k_{\rho_i}=-0.5\) is shifted into the inner magnetic surface. Accordingly, only in the high temperature reversed shear plasma (case (b)), a low-\(n\) dominant growth rate spectrum is produced in the \(q_{\text{min}}\) region or \(r/a<0.5\).

![Fig. 1: Safety factor profiles of normal and reversed shear configurations.](image1)

![Fig. 2: \(n\) dependence of Growth rates (left) and average positions of the eigenfunctions (right). In the reversed shear configuration with \(\alpha/\rho_i=320\) (case (b)), the high-\(n\) modes with \(n>20\) are excited at the outer magnetic surface.](image2)

![Fig. 3: \(h_n\) dependence of the growth rate for \(n=20\) mode. Here, \(L_n\) is fixed and \(\alpha/\rho_i=320\).](image3)

2.2 Slab like ITG Mode

The ITG mode in the normal shear configuration shows a typical toroidal mode structure (see Fig. 4(a)). On the other hand, low-\(n\) global modes in the \(q_{\text{min}}\) region show a coupled mode structure between the slab and toroidal ITG modes (see Figs. 4(b) and 4(c)). A similar mode structure has been observed also from a global gyrokinetic eigenvalue code [3]. Since the eigenmode structure contains significant double rational surface \((m=nq+1)\) and nonresonant \((m=nq-1)\) components around the \(q_{\min}\) surface, this slab like feature shows a contribution from
a reversed shear slab ITG mode [4,5]. It is noted that a gap mode structure, which was observed in a quasilinear saturation phase of a global PIC simulation [6], is not obtained as a linear eigenfunction. Since the slab ITG mode is sensitive to $\eta_i = L_{ni}/L_i$, we have studied the $\eta_i$ dependence of the growth rate in Fig. 3, where a driving effect on the toroidal ITG mode $L_{ni}$ is fixed. In the normal shear case, where the toroidal ITG mode is dominant, the growth rate is almost constant for $\eta_i=1.5-5$. However, in the reversed shear case, the growth rate is reduced by a peaked density profile or a small $\eta_i$ parameter.

![Fig. 4: Typical eigenfunctions of the low-n and high-n modes in the normal and reversed shear configurations. In the $q_{min}$ region, the low-n global mode shows a coupled mode structure between the slab and toroidal ITG modes. In the case (b), the high-n mode ($n=35$) is excited in the outside of the $q_{min}$ surface.](image)

3 Nonlinear Gyrokinetic Global Simulations using Canonical Maxwellian Distribution

In this section, we show the nonlinear simulation using a new scheme based on a canonical Maxwellian distribution $F_{CM}(P_e, \epsilon, \mu)$. Use of $F_{CM}$ is important because of the following reasons. Firstly, in the canonical coordinates, the linear gyrokinetic equation is given by

$$
\frac{d\delta f}{dt} = \frac{\partial\langle\phi\rangle}{\partial \phi} \frac{\partial F_0}{\partial P_e} - \frac{\partial\langle\phi\rangle}{\partial \epsilon} \frac{\partial F_0}{\partial \epsilon},
$$

where $\langle\phi\rangle$ is the gyroaveraged electrostatic potential, and $\phi$ is the toroidal angle. When a canonical Maxwellian $F_{CM}$ is used as the equilibrium distribution $F_0$, axisymmetric perturbations including zonal flows are not driven by $\partial P_e F_{CM}$, which corresponds to the density and temperature gradients [7]. But, when we use a local Maxwellian $F_{LM}(\psi, \epsilon, \mu)$, zonal flows, which are not subject to the Landau damping, are significantly affected by spurious driving effects. Secondly, in a conventional $\delta f$ method based on $F_{LM}$ which is not an exact equilibrium solution of the gyrokinetic equation, a variation of $F_{LM}$ along the unperturbed characteristics, $dR/d\phi \nabla F_{LM} + dv/\partial \psi F_{LM}$, is artificially assumed to be zero. This treatment violates the conservation properties of the system. However, in a new $\delta f$
method using $F_{CM}$ which is an exact equilibrium solution, the method is constructed without using this kind of artificial assumptions. In the present convergence study using the Cyclone base case [8], we compare results obtained from the new and conventional methods.

3.1 Linier Stability and Conservation Properties

In Fig.5, the growth rate spectrum of the ITG mode is shown. As is understood from the linear gyrokinetic theory, a difference between $F_{LM}$ and $F_{CM}$ provides only a minor correction for the ITG mode. And, both results agree well with the previous linear benchmark calculations [8]. In Fig. 6, the energy conservation property of the new code is compared between the conventional Maxwellian particle loading and the optimized particle loading. In GT3D, a 2D particle distribution function, which is optimized in the $r$ and $\phi$ space, is used for an initial particle loading. The convergence tests have been performed using 40 million marker particles, 32 toroidal modes, 32 poloidal mesh with a quasi-ballooning representation, and 76 nonuniform radial mesh. In the case with the optimized loading, the energy conservation property is improved due to a reduction of a particle noise. As for the early nonlinear stage, similar improvements are observed also by using $F_{LM}$. However, long time behaviour of the conservation property becomes quite bad because of spurious zonal flow oscillations.

![Fig.5: The growth rate spectrum plotted for Cyclone base case; $R_0/a=0.36$, $a_p=152$, $R_0/L_p=6.9$, $\eta_\phi=3.12$, and $q=1.4$ at $r=0.5a$.](image)

![Fig.6: The time history of the field energy, the kinetic energy, and the total energy in the simulation of Cyclone base case using the new code with the optimized loading(left). The energy conservation property is improved by using the optimized loading compared with the Maxwellian loading (right).](image)

3.2 Zonal Flow Damping in Axisymmetric Toroidal System

Figure 7 shows the time history of the fluctuation energy in the new and conventional codes. In the case with $F_{LM}$, spurious zonal flow oscillations grow after the saturation of the ITG mode. Since this oscillation is strongly excited only for $(m,n)=(0,0)$ component, it is not a geodesic acoustic mode, which often appear as a damping mode with $m=1$. On the other hand, in the case with $F_{CM}$, such spurious oscillations are not observed, and the zonal flow energy keeps a quasi-steady state. In order to understand these results, we have performed a zonal flow damping test, which was proposed by Rosenbluth and Hinton [9]. In the test shown in Fig. 8, we have solved only $n=0$ component by preparing an axisymmetric initial perturbation, which produces initial $E\times B$ flows with $v_{DEB}=-0.01v_0$. In the case with $F_{CM}$, zonal flows are damped rapidly with $m=1$ damping oscillations and the residual zonal flow level agrees well with the theoretical prediction. This result is also consistent with the linear gyrokinetic theory, which predicts no driving effect on axisymmetric perturbations including zonal flows. However, in the case with $F_{LM}$, spurious zonal flow oscillations are excited. It is noted that in the large aspect ratio limit, both results agree well with each other, and zonal flow damping is recovered also by using $F_{LM}$ [10]. However, for realistic or small aspect ratio configurations, a difference between $F_{LM}$ and $F_{CM}$ becomes large, and use of $F_{CM}$ is essential to simulate a correct response of a plasma against zonal flows.
4 Summary

From the linear global analysis using GT3D, it is found that most unstable high-$n$ modes are excited in the outside of the $q_{\text{min}}$ region in high temperature reversed shear tokamaks. Since the growth rate of low-$n$ slab like modes in the $q_{\text{min}}$ region is much smaller than that of high-$n$ toroidal modes, the reversed shear configuration has an effective stabilizing effect on the ITG mode in the maximum $R/L_n$ region. In the nonlinear simulation, an improved energy conservation property has been confirmed by using the optimized loading. From the nonlinear simulations using the local ($F_{LM}$) and canonical ($F_{CM}$) Maxwellian distributions, it is found that a choice of an equilibrium distribution function is a critical issue especially for studying zonal flows, since they are easily excited by a spurious driving effect of $F_{LM}$. Therefore, use of $F_{CM}$, which simulate a correct zonal flow damping, is essential for a gyrokinetic simulation.

We would like to thank Dr. R. Hatzky, Dr. S. Wang, Dr. J. Li, Dr. T. Fukuda, Dr. G. Rewoldt, Dr. T. S. Hahm, Dr. L. Villard, and Dr. M. Wakatani for useful discussions. We also thank Dr. M. Kikuchi and Dr. A. Kitsunezaki for their support. The simulations were performed on the JAERI Origin3800 system.

References

Gyrokinetic Global Analysis of ITG Modes in Reversed Shear Tokamaks

Y. Idomura, S. Tokuda, Y. Kishimoto
Naka Fusion Research Establishment, Japan Atomic Energy Research Institute
the 19th IAEA Fusion Energy Conference
14-19 October 2002, Lyon, France

Outline
- New GK toroidal PIC code GT3D
- GK global analysis of ITG modes in RS tokamaks
- Zonal flow damping in nonlinear GK simulations

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New GK toroidal particle code GT3D
- New features of GT3D
  - New method based on Canonical Maxwellian $F_{CM}(p_\perp, \varepsilon, \mu)$
    - Correct n=0 response, zonal flow damping
    - New $\delta f$ scheme which hold the conservation property
  - Optimized particle loading
    - Improved particle and energy conservation
  - Quasi-balloonning field solver
    - Linear and nonlinear global analysis of high-m,n modes
    - High efficiency and scalability on JAERI Origin3500 system
    - Capability of simulating large tokamaks ($a/\rho_i \sim 500$)
- Issues addressed using GT3D are
  - GK global analysis of ITG mode with $n=0 \sim 100$ in reactor relevant reversed shear tokamaks ($a/\rho_i = 320 \sim 450$)
  - Comparison of zonal flow dynamics in the conventional and new GK codes based on the local and canonical Maxwellian

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Basic equations
- Gyrokinetic equation
  \[
  \begin{aligned}
  \frac{df}{dt} + \frac{dR}{dt} + \frac{dV}{dt} F_t = \frac{dV}{dt} \frac{dF_t}{dt} &= 0 \\
  \frac{dR}{dt} &= v_b b + \frac{c}{\Omega_b} b \times \nabla \phi_b (\phi_b) + m_b v_b b \cdot \nabla \phi_b + M \Delta \phi_b \nabla \phi_b \Delta \phi_b \\
  \frac{dV}{dt} &= \frac{B^2}{\Omega_b} \frac{d\Omega_b}{dV} \nabla \phi_b \cdot \nabla b - \frac{m_b v_b}{2 \Omega_b} \nabla \phi_b \cdot \nabla b \\
  \frac{d\Omega_b}{dt} &= \frac{1}{\Omega_b} \left[ F_t \delta \left( R + \rho - x \right) dV \cdot \nabla \phi_b \right]
  \end{aligned}
\]

- Gyrokinetic Poisson equation (k^2 \rho^2 < 1)
  \[
  \begin{aligned}
  \nabla^2 + \nabla \cdot \frac{F_t}{\lambda_{\nu}^2} &= 0 \\
  \phi + \frac{1}{\lambda_{\nu}^2} \left( \phi - \phi_b \right) &= 4 \pi \nu \int F_t \delta \left( R + \rho - x \right) dV
  \end{aligned}
\]

D = q_i m_b c / \rho_i \Omega_b \lambda_{\nu}, \quad \rho_i = v_b / \Omega_b, \quad \lambda_{\nu} = (4 \pi \sigma m_b q_i)^{1/2}

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Calculation model of GT3D
- GK model with finite element method (Fitzaz 1998)
- Coordinate systems
  - Cylindrical coordinates $\{ R, \zeta, Z, \psi, \epsilon, \mu, \xi \}$
  - Particle orbits around the axis
  - Flux coordinates $\{ \psi, \theta, \phi, \epsilon, \mu, \xi \}$
    - Quasi-balloonning field solver
    - B.C. for non-circular tokamaks
  - Canonical coordinates
    $ \{ P_\perp, \theta, \phi, \epsilon, \mu, \xi \} $  
    - New $\delta f$ scheme using $F_{CM}$


**Mode structure of poloidal m harmonics**

- normal shear \((m=15, n=30)\)
- reversed shear \((m=15, n=30)\)

- each mode is excited at mode rational surfaces
- typical toroidal coupling
- Reversed shear slab ITG mode (Idomura 1999)
- Gap structure (Kishimoto 2000) is not obtained as linear eigenfunction

**Zonal flow damping in nonlinear GK simulations**

- Cyclone base case
  - Deuterium plasma
    - \(R_i = 1.3 m\), \(n=0.04\), \(\tau = 150 s\)
    - \(B = 1.9 T\), \(a=0.85\times10^{-2}\) (mV)
    - \(\alpha = 1\), \(s(r_i) = 0.78\)
    - \(P_i = P_L = 6\), \(P_i/L = 2.2\)
- Time evolution of \(\varphi\) structure
  - linear phase
  - saturation phase
  - quasi-steady phase

**\(\eta\) dependence of slab like ITG mode**

- Growth rates vs. \(\eta\) (T profile fixed with \(s=0.11\))

- In the normal shear case, where the toroidal ITG mode is dominant, the growth rate is almost constant for \(\eta \approx 1.5 \sim 5\).
- In the reversed shear case, the growth rate is reduced by a peaked density profile or a small \(\eta\) parameter.

**Spurious zonal flow oscillations caused by \(F_{\text{small}}\)**

- Turbulence saturation levels differ by a factor of \(\sim 2\)
- In the nonlinear regime, high frequency \((\omega \sim \nu R_i)\) spurious zonal flow oscillations grow in the conventional code with \(F_{\text{small}}\)
- In the new code based on \(F_{\text{small}}\), spurious oscillations do not appear and zonal flows are sustained in a quasi-steady state.
Correct zonal flow damping obtained using $F_{CM}$

- In the large $R/a$ limit, flow damping theory (Rosenbluth 1998) is confirmed both in the conventional ($F_{CM}$) and new ($F_{CM}$) codes.
- However, in Cyclone base case ($R/a=2.7$), spurious zonal flow oscillations are excited in the conventional code based on $F_{CM}$.
- Linear GK theory predicts no driving effect on zonal flows.
- Use of $F_{CM}$ is essential to study zonal flows in GK simulations.

Conclusions

- GK global analysis of ITG mode in RS tokamaks:
  - Most unstable high-$n$ modes are excited in the outside of $q_{95}$.
  - $q_{95}$ is characterized by less unstable low-$n$ slab-like modes.
  - Through global effects, change the $q_{95}$-mode structure.
  - RS configuration stabilizes the ITG mode at the $q_{95}$ surface.
- GK nonlinear simulation based on canonical Maxwellian $F_{CM}$:
  - Linear GK theory predicts no driving effect on $n=0$ modes.
  - Spurious zonal flow oscillations are excited in the conventional code based on local Maxwellian $F_{CM}$.
  - New code with $F_{CM}$ simulates zonal flow damping correctly.
  - Use of $F_{CM}$ is essential for toroidal GK turbulence simulations, where zonal flows are important for turbulence suppression.
1.19 Geometrical Improvements of Rotational Stabilization of High-\(n\) Ballooning Modes in Tokamaks

M. Furukawa,1) S. Tokuda,1) and M. Wakatani 2)

1) Naka Fusion Research Establishment, JAERI, Naka, Ibaraki 311-0193, Japan
2) Graduate School of Energy Science, Kyoto University, Gokasho, Uji 611-0011, Japan

E-mail: furukawa@fusion.naka.jaeri.go.jp

Abstract. We have found numerically that damping phases appear in the time evolution of the perturbation energy of high-\(n\) ballooning modes in the presence of toroidal shear flows. The damping dominates exponential growth which occurs in the bad curvature region, resulting in stabilization of ballooning modes. D-shaping of plasma cross-section, reduction of aspect ratio, and arrangement of X-point at inner side of the torus enhance the stabilization effect of the toroidal flow through this mechanism.

1 Introduction

The edge localized modes (ELMs) [1] in the H-mode [2] tokamak plasmas are the magneto-hydrodynamic (MHD) activity. Type-I (giant) ELM is related to ideal MHD ballooning modes or peeling modes [1,3]. At the edge region of the tokamak, the plasma often rotates. The rotation is considered to affect the MHD stability.

The WKB theory for high-\(n\) ideal MHD ballooning modes was developed by Connor, Hastie and Taylor [4]. Introduction of Doppler shift in the eikonal representation for the perturbation enables us study of high-\(n\) ballooning modes for toroidally rotating tokamaks [5–9]. It was shown that the high-\(n\) ballooning equations including toroidal flows have dynamical symmetry, and the solutions can exhibit periodically modulated exponential growth. Numerical solutions for the Shafranov equilibrium were shown in Ref. [10], and the unstable region in the so-called \(S-\sigma\) diagram was shown to shrink by the toroidal flow shear.

However, the mechanism of stabilization for ballooning modes by the toroidal flow shear has not been fully clarified. If the flow shear is very small, then ballooning perturbation is considered to evolve as in a static plasma with a given ballooning angle \(\theta_k\) at each instance. This leads to an expression of the growth rate in a rotating plasma; \(\gamma = \int_\pi^\theta \gamma^\text{st} d\theta_k / 2\pi\), where \(\gamma^\text{st}\) is the growth rate in a static plasma and is a function of \(\theta_k\) [10]. However, if \(\gamma^\text{st} > 0\) for any \(\theta_k\); i.e., pressure gradient exceeds its critical value in a static plasma, \(\gamma > 0\) since \(\gamma^\text{st} > 0\) in the ideal MHD model. Then the system cannot be stabilized. Therefore we have studied the mechanism of stabilization numerically, and found that the perturbation energy damp owing to the flow shear. The damping occurs in the good curvature region. When the damping dominates the exponential growth in the bad curvature region, the ballooning mode is stabilized. Thus, the stabilization of ballooning modes by the toroidal flow shear is expected to be enhanced by the reduction of (i) the instantaneous growth rate and (ii) the duration of the exponentially growing phase. In this paper we control them by changing geometrical parameters such as aspect ratio, ellipticity, triangularity, and position of X-point. We found numerically that D-shaping, reduction of aspect ratio, and arrangement of X-point at inner side of the torus enhance the stabilization effect of the toroidal flow. In Section 2, the physical mechanism of stabilization is clarified. In Section 3, the sensitivity to the geometrical parameters such as aspect ratio, ellipticity, triangularity, and position of X-point are investigated. Conclusions are given in Section 4.
2 Mechanism of Stabilization

We obtain MHD equilibria by solving the Grad-Shafranov equation including toroidal flows [13] numerically under semi-fixed boundary condition. The pressure profile has a large gradient near the plasma edge. The poloidal beta is $\beta_p = 1.2$ for equilibria with the aspect ratio $A = 3$, and the ratio $\beta_p/A = 0.4$ is fixed when $A$ is varied to keep the Shafranov shift. The current density profile is slightly modified from a parabola to adjust the safety factor $q = 5$ at the 95% flux surface. We have investigated the stability on the 95% flux surface, which is 3–5cm inner from the separatrix when the minor radius is 1m. The total plasma current is adjusted so that $q = 1$ at the magnetic axis. As for the toroidal flow, the flow shear can be given arbitrarily on a magnetic surface since the flow shear does not contribute to the force balance as long as the magnitude of the flow itself is zero. Thus the toroidal rotation frequency $\Omega$ is zero in the equilibrium calculations. Finally, when a magnetic-shear parameter $s_m$ and a pressure-gradient parameter $\alpha_p$ are varied on a magnetic surface, we have used the local equilibrium of Greene and Chance [14].

First we show in Fig. 1 the time evolution of $\|\xi^2_1\|$ and $\|\xi^2_1\|$ for the aspect ratio $A = 3$, the ellipticity $\kappa = 1.4$, the triangularity $\delta = 0.4$, the magnetic shear parameter $s_m = 3$, the pressure gradient parameter $\alpha_p = 3.4$, and toroidal flow shear $\Omega' \tau_A = -0.03$, where $|a| = \int ad\phi$, $\phi$ is the poloidal angle in the covering space, and $\tau_A$ is the Alfvén time (connection length/Alfvén velocity). The prime denotes the derivative with respect to the normalized poloidal flux. The value of $\Omega' \tau_A = -0.03$ is achieved in conventional tokamak experiments [15]. In National Spherical Torus Experiment (NSTX), $\Omega' \tau_A \approx -0.3$ is obtained [16]. The horizontal axis denotes time normalized by the period $\tau_d \equiv 2\pi/(d\Omega/dq)$. The vertical line at $\tau_d = 1$ and 2 indicates the timings when the phases of each twisted slice mode are the same at $\theta = 0$ (bad curvature side) [9].

![FIG. 1. Time evolution of $\|\xi^2_1\|$ and $\|\xi^2_1\|$](image1)

When the phases of each twisted slice mode are the same at $\theta = 0$, $\|\xi^2_1\|$ grows exponentially.

![FIG. 2. Time evolution of $\|\xi^2_1\|$ for $\alpha_p = 3.0$ (stable) and 3.4 (unstable). Stability is determined by the competition between the exponential growth and the damping of $\|\xi^2_1\|$.](image2)

We found from Fig. 1 that damping phases appear in the time evolution of $\|\xi^2_1\|$. The damping is the crucial mechanism for the stabilization due to the flow shear. We also found that $\|\xi^2_1\|$ grows around $\tau_d = 1, 2, \cdots$, and $\|\xi^2_1\|$ begins to grow after $\|\xi^2_1\|$ becomes sufficiently large. If $\|\xi^2_1\|$ does not increase on the average over the time period, then $\|\xi^2_1\|$ oscillates rather than grows or
damps. The instantaneous growth rate is nearly equal to the growth rate in the static plasma. As for the time duration of the exponentially growing phase, it is very short in Fig. 1, since the unstable region in the $\theta_s$ space is narrow in the static plasma. Thus $||\xi_1^2||$ is closely related to the driving mechanism of the instability, and therefore we focus on it in the following.

In Fig. 2, the time evolution of $||\xi_1^2||$ is shown for $\alpha_p = 3.0$ and 3.4. The flow shear is $\Omega' \tau_A = -0.03$. We found that the damping of $||\xi_1^2||$ dominates the exponential growth at $t/\tau_d = 1, 2, \cdots$ for $\alpha_p = 3.0$, and the ballooning mode is stabilized. For $\alpha_p = 3.4$, on the other hand, the damping is not strong enough to dominate the exponential growth. The competition between the damping and the growth determines the stability of the ballooning mode in the presence of the toroidal shear flow.

Therefore, we expect that the stabilization of ballooning modes by a flow shear could be further enhanced by reduction of (i) the instantaneous growth rate and (ii) the duration of the exponentially growing phase. In the next Section, we verify it by changing geometrical parameters such as aspect ratio $A$, ellipticity $\kappa$, triangularity $\delta$, and position of X-point.

3 Improved Stability by Geometrical Effects

3.1 D-shaping and Aspect Ratio

First, we change the ellipticity and triangularity while keeping the magnetic curvature at the outer side of the torus unchanged. This means the driving force of ballooning modes; i.e., the product of the pressure gradient and the magnetic curvature, is held constant. Figure 3 shows the time evolution of $||\xi_1^2||$ for circular cross-section and D-shaped tokamaks. The aspect ratio is $A = 3$ and the flow shear is $\Omega' \tau_A = -0.03$. The instantaneous growth rate at the exponentially growing phase is almost the same for the two equilibria. Here, $\alpha_p = 1.8$ for the circular cross-section, and $\alpha_p = 3.8$ for the D-shape. These equilibria are located near the first stability boundary in the $S-\alpha$ diagram. Thus if $\alpha_p$ is the same, the instantaneous growth rate is smaller in the D-shaped tokamak plasma. The reason is known as the increase of the good curvature region. As for the duration of the exponentially growing phase, we found that it is shorter for the D-shape, since the good curvature region is wider than that for the circular cross-section. This leads to enhancement of the stabilization effect of the flow shear.

Figure 4 shows the critical pressure gradient $\alpha_p^{\text{crit}}$ as a function of triangularity $\delta$. To keep the magnetic curvature at the outer side of the torus unchanged, $\kappa$ and $\delta$ are changed simultaneously; $(\kappa, \delta) = (1.0), (1.2, 0.2), (1.4, 0.4)$, and $(1.6, 0.6)$. We found, from Fig. 4, that the increment of $\alpha_p^{\text{crit}}$ due to the flow shear increases as $\delta$. Therefore, the D-shaping not only raises the critical pressure gradient of ballooning modes in a static plasma, but also enhances the stabilization effect of toroidal flow shear. This is favorable for tokamaks aiming at high beta.

Next, we change the aspect ratio. As in the case of D-shaping, the reduction of the aspect ratio increases good curvature region on a magnetic field line. In Fig. 5, the critical pressure gradient $\alpha_p^{\text{crit}}$ is shown as a function of $A$. As $A$ is reduced, $\alpha_p^{\text{crit}}$ increases in both static and rotating plasmas. The increment of $\alpha_p^{\text{crit}}$ due to the flow shear also increases as $A$ is reduced. Thus the reduction of $A$ is favorable to achieve high beta plasmas.
3.2 Position of X-point

It is known that a magnetic field line stays for much of its length in the vicinity of the X-point, and the local shear is divergent. Therefore ballooning/interchange stability was studied for model [17] and JT-60U [18] equilibria with an X-point, and it was shown that an X-point at the outer side of the torus does not change first stability boundary significantly [18].

In the present paper, we include a toroidal shear flow, since the mechanism of stabilization of ballooning modes by the flow shear is closely related to the magnetic configuration. We have examined ballooning stability of two extremely different equilibria; one has an X-point at the inner side of the torus, and the other has at the outer side. The aspect ratio of both equilibria is \( A = 10 \) and the beta is very low, thus the cross-sectional shape is nearly circular except for the region close the separatrix. These are not realistic, however, the difference of magnetic configuration due to the X-point is magnified.

We show the growth rate \( \gamma T_A \) as a function of flow shear \( \Omega' T_A \) in Fig. 6. The growth rate without a flow is 0.2 for both equilibria. In the equilibrium with inside X-point, the ballooning mode is stabilized by a smaller \( \Omega' T_A \) than in the equilibrium with outside X-point. We have also calculated \( \alpha_p^{\text{crit}} \) and its increase by a flow shear \( \Omega' T_A = -0.03 \) for the separatrix equilibria, however, the first stability boundary is not affected largely. The reason is conjectured as follows. In a static equilibrium, growth rate of ballooning mode becomes significantly large at \( \alpha_p \) slightly larger than its marginal value around the first stability boundary. Thus a small flow shear cannot affect its growth significantly.

4 Conclusions

We have clarified the mechanism of stabilization of high-\( n \) ballooning modes by toroidal flow shear numerically. Damping phases appear in the time evolution of the perturbation. The damping dominates the exponential growth in the bad curvature region, which leads to the stabilization of the ballooning modes. D-shaping and reduction of aspect ratio enhance the stabilization
FIG. 5. Reduction of aspect ratio increases not only $\alpha_p^{\text{crit}}$ in a static plasma, but also the increment of $\alpha_p^{\text{crit}}$ due to the toroidal flow shear.

FIG. 6. Growth rate $\gamma_{TA}$ is plotted as a function of flow shear $\Omega \tau_A$. The flow shear required to stabilize the ballooning mode is smaller for the equilibria with an inside X-point than that for the outside one.

Effect of toroidal shear flow through this mechanism, as well as raise the critical pressure gradient in a static plasma. In the equilibrium with inside X-point, the flow shear required to stabilize the ballooning mode is smaller than in the equilibrium with outside X-point. However, the critical pressure gradient is not largely changed by the modest flow shear.

Acknowledgements: I would like to thank Dr. Y. Kishimoto, Dr. M. Azumi, Dr. H. Shirai, Dr. T. Ozeki, Dr. M. Kikuchi, and Dr. A. Kitsunezaki for fruitful discussion and comments.

References

Geometrical improvements of rotational stabilization of high-\(n\) ballooning modes in tokamaks

M. Furuikawa, S. Tokuda
Japanese Atomic Energy Research Institute,
Naka Fusion Energy Establishment, Japan
M. Wakatani
Graduate School of Energy Science,
Kyoto University, Japan
19th IAEA Fusion Energy Conference
Lyon, 14-19 Oct., 2002

E-mail: M. Furuikawa: furukawa@fusion.naka.jaeri.go.jp

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THP2-03

Highlight

\(\lambda\) Toroidal flow shear damps perturbation energy of ballooning mode when the mode passes good curvature region

\(\lambda\) When the damping dominates exponential growth in bad curvature region the ballooning mode is stabilized

\[\gamma = \frac{1}{2\pi} \int d\theta \frac{\partial \gamma''}{\partial \theta}(\theta_0)\]

\(\gamma\): growth rate
\(\theta_0\): ballooning angle
\(\gamma''\): growth rate without a flow

However, in ideal MHD model, \(\gamma'' \geq 0\) then \(\gamma > 0\) (unstable even if a flow exists)

(l) Clarify how ballooning modes are stabilized by toroidal flow shear
Appearance of damping phase in the time evolution of perturbation energy

Damping phases appear

At $t/k_d = 1.2, ..., t/k_d$, the position of identical phase of each twisted slice mode comes to $\theta = 0$ and $\|\phi_{\perp}\|$ is closely related to driving mechanism of instability.

Background and Motivation - 2

Competition between damping (good curvature) and exponential growth (bad curvature) determines the stability.

Magnetic configuration is changed through geometry, which leads to reduction of
(i) instantaneous growth rate
(ii) duration of the exponential growth

Equilibrium

- Equilibrium is obtained by solving the Grad-Shafranov equation under semi-fixed boundary condition
- Flow shear can be arbitrary given on a flux surface since it does not contribute to the force balance as long as the magnitude of the flow is zero
- Pressure gradient and magnetic shear
- Local equilibrium model by Greene and Chance is used

Stability is calculated on 95% flux surface in the following.
**Stability improvement by D-shaping**

- **Time evolution**
  - Instantaneous growth rate at the exponentially growing phase is almost the same, although \( \alpha_s = 1.8 \) for \( \delta = 0 \) and \( \alpha_s = 3.8 \) for \( \delta = 0.6 \).
  - Duration of the exponential growth phase for \( \delta = 0.6 \) is shorter than that for \( \delta = 0 \).

- **Critical pressure gradient**
  - Increment of \( \alpha \) increases as \( \delta \) (and \( \delta \)) is increased, as well as \( \alpha \) without a flow increases.

**Background and Motivation - 3**

Ballooning/Interchange instability was studied for models and JT-60 equilibria in the absence of a plasma flow.

- Inside X-point Source of ballooning instability
  - Magnetic field line stays for much of its length
  - Local shear is divergent

(III) How different is the stabilization effect of toroidal shear flow

2. Ibid 26, 1063 (1986).
3. Azumi (private communication).

**Stability improvement by reduction of aspect ratio**

- **Time evolution**
  - Instantaneous growth rate at the exponentially growing phase is almost the same, however, \( \alpha_s = 3.0 \) for \( A = 4 \) and \( \alpha_s = 4.2 \) for \( A = 2 \).
  - Duration of the exponential growth phase for \( A = 2 \) is shorter than that for \( A = 4 \).

- **Critical pressure gradient**
  - Increment of \( \alpha \) increases as \( A \) is reduced, as well as \( \alpha \) without a flow increases.

**Stabilization by shear flow for in/outside X-point equilibrium**

- **Time evolution**
  - Growth rate v.s. flow shear
  - Duration of the exponential growth phase for inside X-point is shorter than that for outside X-point

The value of the flow shear required for stabilization is smaller for inside X-point than that for outside X-point.
Most dangerous perturbation in high-$n$ limit

Heuristic dispersion relation — Eq. (1) of Weirbrook and Chen (1991)
for ballooning modes
in a rotating plasma

\[ (\omega - k \cdot v)^2 = v_A^2(k_0^2 - \beta \mu_0^2) \]

- $\omega$: frequency
- $k$: wave vector
- $v$: equilibrium flow velocity
- $v_A$: Alfvén velocity
- $k_0$: parallel wave number
- $\beta$: effective beta

Kinetic energy bending force

$\frac{1}{2} \rho \frac{\partial^2 \frac{\partial \phi}{\partial t}}{\partial t^2} + \nu \frac{\partial \phi}{\partial t} = \frac{1}{\mu_0} \left[ (\nabla \times \mathbf{Q}) \times \mathbf{B} + (\nabla \cdot \mathbf{B}) \cdot \mathbf{Q} \right]
+ \nabla \cdot (p \nabla \phi + \nabla \phi) + \nabla \cdot (\rho \phi \cdot \nabla v - \rho v \cdot \nabla \phi)

\mathbf{Q} = \nabla \times (\mathbf{B} \times \mathbf{B})$: perturbed magnetic field

$\mathbf{v}$: equilibrium velocity,
$\gamma$: specific heat ratio


High-$n$ ballooning equation in toroidally rotating tokamak

Wave equations for $\zeta_0$ and $\zeta_1$ along a magnetic field line

\[ \rho \frac{\partial^2 \frac{\partial \zeta}{\partial t}}{\partial t^2} + 2 \rho \Omega \hat{\mathbf{z}} \cdot \frac{\partial \hat{\mathbf{z}}}{\partial t} \frac{\partial \zeta}{\partial t} + 2 \rho \Omega \cdot \hat{\mathbf{z}} \frac{\partial \zeta}{\partial t} = \frac{\partial}{\partial t} \left[ \rho \left( \frac{\partial \zeta}{\partial g} + \frac{\partial \zeta}{\partial \phi} \right) \right]
+ C \left( \mathbf{B} \cdot \nabla \phi \right) \left( \frac{\partial \zeta}{\partial g} + \frac{\partial \zeta}{\partial \phi} \right)
\]

We have solved these equations numerically
as an initial value problem

Precession of identical-phase position of twisted slice modes

This picture leads, in small flow shear limit,

\[ \gamma = \frac{1}{2\pi} \int d\theta \gamma^*(\theta) \]

\( \gamma^* \): growth rate

Identical-\( \phi \) position

Toroidal-flow velocities on neighboring magnetic surfaces

As time evolves, constant-\( \phi \) lines move with different velocities.

\[ \phi = 0 \]

Identical-\( \phi \) position

Constant-\( \phi \) lines on neighboring magnetic surfaces

Qualitative argument on separatrix equilibria

(1) Magnetic field line stays for much of its length
(2) Local shear is divergent

Source of ballooning instability

Inside X-point favorable unfavorable
(1) Magnetic well favorable
(2) Local shear at bad curvature region for same global shear

Outside X-point unfavorable favorable

Global shear

\(-\alpha\) to \(\alpha\)
1.20 Objectives and Design of the JT-60 Superconducting Tokamak


1) Japan Atomic Energy Research Institute, 2) Tohoku University, 3) University of Tsukuba, 4) Osaka University, 5) Kyushu University, 6) Keio University, 7) Hokkaido University, 8) Tokyo Institute of Technology, 9) the University of Tokyo, 10) Kyoto University, 11) Mie University, 12) Toshiba Corporation Power Systems and Services Company, 13) Hiroshima University, 14) Central Research Institute of Electric Power Industry, 15) Ibaraki University, 16) ITER JCT, 17) Nagoya University, 18) National Institute for Fusion Science, 19) National Institute of Advanced Industrial Science and Technology, Japan

e-mail contact of main author: ishida@naka.jaeri.go.jp

Abstract. A fully superconducting tokamak named as JT-60SC is designed for the modification program of JT-60 to enhance economical and environmental attractiveness in tokamak fusion reactors. JT-60SC aims at realizing high-beta steady-state operation in the use of low radio-activation ferritic steel in low V* and \( \rho^* \) regime relevant to the reactor plasmas. Objectives, research issues, plasma control schemes and a conceptual design for JT-60SC are presented.

1. Objectives

The modification program of JT-60 viewing the next decade is oriented in the following directions and a conceptual design of the modification is presented in nation-wide collaboration with universities, institutes and industries. In order to improve economic and environmental suitability of tokamak fusion reactors, the accomplishment of low circulating power operation in accord with a high pressure plasma (i.e., high-beta steady-state operation) [1] and the establishment of utilization technology of low radio-activation materials to minimize the influence of radioactive waste to the environment [2] are crucially important. It is the demonstration of the use of low activation ferritic steel for reactor-relevant plasmas that are necessary to expedite the practical use of the material as a most promising candidate for the first wall material in DEMO reactors. The attainment of these objectives by modifying the JT-60 would effectively contribute to earlier realization of tokamak fusion reactors in cooperation with the ITER program and material developments.

So far, the JT-60 program has consistently pursued the subjects for the basis of steady-state tokamak operation and pioneered advanced tokamak operation regimes consistent with a high bootstrap current fraction such as high-beta, reversed shear discharges [3]. Thus, the modified JT-60 is oriented to lead the way to demonstrate the high-beta steady-state operation in the use of the ferritic steel characterized by ferromagnetic properties to the plasma confinement device for the reactor-relevant plasmas.

2. Research Issues and Main Parameters

Scale of the plasma to be modified is discussed as follows. It is important for close extrapolation to perform demonstration experiments in steady state bringing non-dimensional parameters of normalized Larmor radius \( \rho^* \), normalized collisionality \( v^* \) and normalized beta \( \beta_n \) closely to those of reactor plasmas. Considering these non-dimensional parameters, the concept of the modification is determined to be a large superconducting tokamak, here called JT-60SC, which allows to create high performance plasmas in a break-even class (DT-equivalent fusion energy multiplication factor \( Q_{DT} \sim 1 \)) and to sustain the plasmas for a long
duration (~ 100 s) sufficiently exceeding the current diffusion time or skin time.

Figure 1 shows the target area of JT-60SC above the experimental data on $\beta_N$ achieved in various tokamaks in comparison with steady state reactor designs of SSTR [4], CREST [5] and ARIES [6], a non-inductive operation scenario of ITER [7], where the $\beta_N$ values required from reactor designs are ranged from $\beta_N$$=3.5 - 5.5$. In response to the above requirements, it is necessary for the modification to incorporate superconducting toroidal and poloidal coils including renewal of main components such as vacuum vessel. However, the maximum utilization of existing facilities and equipments in JT-60 such as buildings, power supplies, heating equipments and diagnostics would significantly increase the advantage in cost and period for the modification.

Research issues for JT-60SC are summarized with target values based on conceptual reactor designs of SSTR and CREST emphasizing reactor economics as follows.

(i) Realization of high-beta steady-state operation:
- high beta plasma control ($\beta_N = 3.5 - 5.5$),
- steady state plasma control ($f_{\text{p}} = 50 - 90\%$),
- divertor heat and particle control ($t_{\text{red}} \sim 95\%$, $t_{\text{p}} + t_{\text{e}} \sim 5$), and
- disruption control (avoidance and mitigation).

(ii) Demonstration of the use of low-activation ferritic steel:
- ideal MHD mode control with resistive and ferromagnetic wall,
- locked mode and neoclassical tearing mode control with additional error fields, and
- heat and particle control with plasma-material interaction.

<table>
<thead>
<tr>
<th>Parameter</th>
<th>JT-60U</th>
<th>JT-60SC</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pulse length</td>
<td>15 s</td>
<td>100 s (flat top)</td>
</tr>
<tr>
<td>Max. input power</td>
<td>40 MW (10 s)</td>
<td>44 MW (10 s)</td>
</tr>
<tr>
<td></td>
<td>15 MW (100 s)</td>
<td></td>
</tr>
<tr>
<td>Plasma current</td>
<td>3 MA</td>
<td>4 MA</td>
</tr>
<tr>
<td>Toroidal field $B_t$</td>
<td>4 T</td>
<td>3.8 T ($B_{\text{pol}}$=2.8 m)</td>
</tr>
<tr>
<td>Major radius $R_p$</td>
<td>3.4 m</td>
<td>2.8 - 3 m (2.8 m$^*$)</td>
</tr>
<tr>
<td>Minor radius $a_p$</td>
<td>0.9 m</td>
<td>0.7 - 0.9 m (0.86 m$^*$)</td>
</tr>
<tr>
<td>Elongation $\kappa_p$</td>
<td>1.8 (at $\kappa_p^* = 0.05$) $\leq 2 (1.8^*)$</td>
<td></td>
</tr>
<tr>
<td>Triangularity $\delta_p$</td>
<td>0.4 ($\delta_p^* = 0.33$) $\leq 0.5 (0.35^*)$</td>
<td></td>
</tr>
</tbody>
</table>

Fig.1. Normalized beta as a function of toroidal field showing the target area of JT-60SC along with reactor designs and the present experimental data region.

Fig.2. Comparison of cross-sectional views between JT-60U and JT-60SC.
where \( f_{bs} \) and \( f_{rad} \) denote the bootstrap current fraction, the radiation fraction and the effective helium confinement time.

In response to the above requirements, main parameters of JT-60SC are defined nominally with the plasma current of 4 MA, the toroidal field of 3.8 T, the major radius of 2.8 m and the minor radius of 0.85 m as shown in Table 1 and a conceptual design is completed as shown in Fig. 2. Main heating equipments for JT-60SC are neutral beams capable of 40 MW/10 s or 13 M\( \text{W}/100 \) s (including negative ion beams: 10 MW/10 s or 3 M\( \text{W}/100 \) s) and electron cyclotron waves of 4 MW/10 s or 1.7 MW/100 s. Using the existing heating facilities, the JT-60SC is able to produce and sustain reactor-relevant plasmas for 100 s beyond the skin time \( \tau_{\text{skin}} \sim 30 \) s in the range of \( \rho^* \sim 0.01 \) and \( \nu^* \sim 0.01 \); \( Q_{DF}=1.0 \) is projected for an inductive discharge with 4 MA, 3.8 T, \( P_{NB}=13 \) MW and \( HH_{PB(2,2)}=1.18 \).

Figure 3 shows \( \beta_N \) as a function of the bootstrap current fraction illustrating the target area of JT-60SC, where full current drive data from high-\( \beta_N \) and reversed shear discharges in JT-60U [8,9,10,11], projections from SSTR, CREST and ITER are plotted. Typical results from the time-dependent transport simulations using the TOPICS code for JT-60SC are also plotted within the JT-60SC target area [12]. This shows that the attainment of high-\( \beta_N \) operation with a high bootstrap fraction under the full current drive condition is an important issue. In Fig. 4, \( \beta_N \) is shown as a function of the ratio of the sustain duration of the high beta plasma to the skin time corresponding to the plots shown in Fig. 3; where the skin time is simply defined as \( \tau_{\text{skin}}=\phi_{\text{c}}/\gamma \) around a half of minor radius. The necessity of long pulse experiments exceeding the skin time for the full current-drive high performance plasmas are clearly shown in Fig. 4 in comparison with the JT-60U data. The target area of JT-60SC leading to reactor improvements has never been explored in any other tokamaks and is set beyond the scope of the ITER design for steady state operation.

3. Plasma Control Schemes

The eventual goal in JT-60SC is to achieve a comprehensive solution of the high-beta steady-state tokamak operation which is characterized by simultaneous attainments of sufficiently high confinement, normalized beta and bootstrap current fraction, efficient heat & particle control and almost disruption-free operation even in the use of the ferritic steel. On the way to the goal, the following physics issues are addressed with feasible plasma control schemes.

3.1 High Beta Plasma Control

Figure 5 shows the extended performance capability by the modification of JT-60 to JT-60SC
to demonstrate the sustainment of high $\beta_N$ plasmas in comparison with the present data from various tokamaks. CREST and SSSTR are designed with $\beta_N=5.5$ and 3.5, respectively, well above no wall limit of ideal MHD instabilities. A non-inductive operation is aimed at $\beta_N=3.1$ in ITER.

As the target $\beta_N$ range in JT-60SC is substantially above no wall MHD limit shown in Fig.6, the passive stabilizer plate made of the ferritic steel (F82H) is placed on the location similar to the blanket surface of a DEMO reactor; the JT-60SC plasma is approximately at $r_{wall}/a\sim 1.3$. To stabilize resistive wall modes with multiple toroidal mode numbers of $n=1$ and 2 which could be unstable due to magnetic field penetration into the resistive wall, 18 sector coils are placed inside the vacuum vessel and behind the stabilizer plate, for active feedback control with fast response.

According to scaling laws for onset condition of neo-classical tearing modes (NTMs) [13], NTMs could be unstable in the required high beta plasmas and terminate the sustainment of the high $\beta_N$ operation. Active and local electron cyclotron current drive (ECCD) technique becomes feasible for suppression of the NTMs by steering the launcher in combination with a high-resolution measurement (~1 cm) of the magnetic island since JT-60U has demonstrated the feedback stabilization of NTMs using the same techniques [14]. Time dependent NTM stability analysis combined with TOPICS and ECCD codes has been carried out for the JT-60SC plasma at 3 MA, showing that the NTMs can be suppressed by the present ECCD power of ~2 MW for 100 s and its decay time is reduced with decreasing the island width at which the ECCD is applied as show in Fig.7 [15].

3.2 Steady-State Plasma Control

Real time stabilization of the high beta plasma is planned with effective shaping, current/pressure and plasma rotation profile control capabilities using the poloidal field coils.

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Fig.5. Attainable $\beta_N$ as a function of time for various heating scenarios (44 MW/10 s, 41 MW/20 s, 30 MW/30 s, 27 MW/50 s, 15 MW/100 s) in JT-60SC along with the data on high beta achieved in tokamaks and the reference $\beta_N$ values of SSSTR, CREST and ITER.

Fig.6. $\beta_N$ as a function of normalized wall radius showing the beta limits for kink-ballooning modes with $n=1$ to 4 and possible resistive wall mode region along with the used pressure and current profiles.

Fig.7. Temporal change of the normalized island width for three timings of ECCE onset showing different decay times of the island suppression.
a variety of beam arrangements in combination with central and off-axis, tangential and perpendicular injections with 90 and 500 keV, and steerable injection of 110 GHz EC wave from five gyrotrons.

Analyses based on high-$\beta_p$ and reversed shear plasmas in JT-60U using TOPICS and ACCCOME codes show the potential feasibility of high-beta steady-state operation for JT-60SC. As shown in Fig.8, the parameter scan for 3 MA discharges with 30 MW using the ACCOME code shows the wide range of full current drive capability including $H_{H|P(B<2)}=2.0$, $\beta_B=3.5$ and $f_{BS}=65\%$ with $n_0\tau_E T_0=1\times10^{11}$ keVsm$^3$. The time dependent transport analysis using the TOPICS code shows the potential of high-beta steady-state operation at $\beta_n\sim5$ with $f_{BS}=86\%$ for 100 s as shown in Fig. 9 where the parameters are $I_p=1.5$ MA, $B_t=2$ T, $H_{H|P(B<2)}=2.2$, $P_{NB}=11$ MW including N-NBI. The ERATO-J code analysis shows that the profiles with internal transport barrier could be stable for kink-ballooning modes.

3.3 Divertor Heat and Particle Control

Effective heat and particle control by the use of divertor is required as leading to simultaneous achievement of full current drive and partially detached divertor at high beta and reduction of impurity influx. The impurity control can be accomplished by a strong SOL flow produced by intense divertor pumping in combination with gas puff or high-field-side pellet injection. To ensure the heat and particle control including pulsed heat loads due to type 1 ELMs, the vertical target divertor is designed to be compatible with the elongated D-shaped equilibrium maintaining a sufficient SOL width of $\sim3$ cm with a semi-closed divertor as shown in Fig.10 [16]. With the semi-closed divertor, the shaping control enables heat and particle control for the single null configurations with a large shift of the X-point and a wide expansion of the magnetic flux near the divertor region due to increase in the beta value. Independent pump-out scheme from inner and outer divertor slots using separated cryo-pumps is adopted for continuous partial detachment control: where the cryo-pump is capable of pumping out $-50$ m$^3$/s at each slot based on numerical divertor simulation using the SOLDOR/NEUT2D code.

Fig.8. Non-inductive, bootstrap and beam-driven current fractions, HH-factor, normalized density and $\beta_0$ and fusion tripple product as a function of the central density for full current drive condition from ACCOME code calculations at $I_p=3$ MA, $B_t=3.8$ T and $P_{NB}=30$ MW.

Fig.9. High-beta steady-state operation scenario sustained for 100 s from the TOPICS code analysis at $I_p=1.5$ MA, $B_t=2.0$ T and $P_{NB}=11$ MW, showing the temporal change of the plasma current, bootstrap current, beam current and OH current, $B_n$ and HH-factor and the profiles of $T_e$, $T_i$, $n_e$, $n_i$ at 100 s and $q$ at 20 s to 100 s.
Since the divertor heat load could reach ~20 MW/m² without any radiation loss, effective forced water cooling of the target plates is required as well as the radiative divertor formation. The material of divertor facing component will start from low-Z materials such as CFC including a local test of high-Z materials such as W-alloy towards the demonstration of divertor materials for DEMO reactors.

3.4 Effects of the Use of Ferritic Steel

The promising results from JFT-2M experiments using ferritic steel plates fully placed in the vacuum vessel [17] strongly support the use of ferritic steel in JT-60SC. In JT-60SC, ferritic steel is used for stabilizing baffle plates and first wall consisting of armors and pedestals so that the plasma is closely surrounded by the ferritic steel as shown in Fig.11. In JT-60SC, 18 TF coils generate magnetic fields with ripple rate up to 0.59% in the plasma region. To reduce the ripple rate in the ferritic steel configuration due to port hole arrangements, additional ferritic steel plates are appropriately installed inside the vacuum vessel behind the TFC. Consequently, the TF ripple rate of the 18th mode is reduced to down to 0.29% for plasma region over a wide range of $B_t=2.0$-3.8 T. The OFMC (Orbit Following Monte Carlo) code analysis to calculate fast ion losses indicates the effectiveness of ripple reduction where the fast ion loss for nearly perpendicular beams is evaluated to be 2.0% as shown in Fig.12.

Error fields induced by the ferritic steel are calculated on the plasma surface precisely including asymmetry and possible installation errors of the ferritic steel components; the Fourier spectrum of $m=7$ $2/n=1$ are evaluated to be ~0.4 G. It is negligibly small in comparison with an onset level of locked mode, 7.6 G, at 3.8 T deduced from the result from DIII-D [18]. It is theoretically suggested that, with the presence of ferromagnetism in the wall material, the critical beta is reduced by ~10% with a wall thickness of 0.07a for $\mu/\mu_0=2$ at which the ferritic steel is sufficiently saturated [19]. Effect of flow velocity on the ideal MHD with resistive and ferromagnetic wall has been also investigated as shown in Fig.12, where the growth rates for both resistive and ideal wall branches are increased with permeability even with substantial flow velocity. These predictions should be elucidated in JT-60SC.

4. Superconducting Coils and Vacuum Vessel

Major components of the JT-60SC device are the superconducting coil system and the vacuum vessel containing a single-null elongated plasma, stabilizing baffle plates, first wall, a
vertical pumped divertor, fast position control coils and sector coils [20]. A cryostat encloses all the superconducting coils, the vacuum vessel and the support structures with thermal shields as shown in Fig.14.

The superconducting coil system consists of 18 toroidal field coils (TFC) and 10 poloidal field coils (PFCs) external to the TFC [21]. The PFC consists of 4 segments of the central solenoid (CS) and 6 equilibrium coils (EFCs). The TFC system produces 3.8 T at \( R_p=2.8 \) m with a maximum field of 7.4 T at winding and a magnetic stored energy of 1.7 GJ. A squired Nb3Al cable-in-conduit conductor (CIICC) using a stainless steel is a primary candidate for the TFC in JT-60SC, having the advantage of low strain sensitivity on superconducting performance allowing a react-and-wind technique and a high critical current density leading to a significant reduction of the amount of superconducting material [22]. Inboard side of the TFC cases is wedged against the centering force of the TFC. Inter-coil structures between the unit coils are provided to resist the toroidal forces as well as the centering force.

The PFC system provides a flux swing of 40 volt-seconds to inductively initiate the discharge and ramp up the plasma current up to 4 MA while leaving ~15 volt-seconds capable of sustaining the current flat top for 100 seconds at 4 MA without non-inductive current drive. The plasma is initiated by applying a toroidal electric field of 0.3 V/m with assistance of ECH. The CS is composed of 4 segment unit coils stacked in a support structure. A squired Nb3Sn CIICC using a stainless steel is used for the CS. The EF coil system consists of 6 coils, which are independently operated. The EFC for the bottom divertor coil uses the same conductor as the CS. Other EF coils are designed to use a squired Nb3Ti CIICC with a circular central channel for cooling, fabricated by a roll forming method [23].

The vacuum vessel is a double-walled structure with a polygonal shape having a low toroidal resistance of ~30 \( \mu \Omega \). The double walls made of low Co-contamination SS (SS316L) are filled with pure water for neutron shielding. Additionally, the SS316L boards are installed outside the vacuum vessel for \( \gamma \)-ray shielding. So that suppression of nuclear heating in the superconducting coils is established below 2.5 mW/cm\(^2\) at coil winding. A neutron budget of 2 \( \times 10^{20} \) neutrons/year is also established for JT-60SC based on 100 seconds of operation within 4 \( \times 10^{-17} \) n/s in deuterium. Vertical stability and kink stability for high beta plasmas can be achieved with passive structure of stabilizing baffle plates located inboard and outboard of the plasma. For slower growth of these instabilities, in-vessel copper coils to control horizontal and vertical fields and 18 sector coils to control helical magnetic field components

![Fig. 12. 3D analysis fast ion losses (W/m\(^3\)) on the first wall taking into account port arrangements without (top) and with (bottom) ripple compensation by ferritic steel where arrows show the direction of nearly perpendicular beam injection.](image)

![Fig. 13. Normalized growth rate as a function of normalized wall radius for permeability \( \mu/\mu_0=1, 2 \) (for JT-60SC), 4 of the resistive wall under substantial flow velocity in comparison with no flow case.](image)
are installed as shown in Fig.11.

5. Summary

Objectives and design for the modification of JT-60 to a fully superconducting tokamak are defined and implemented, respectively, to realize high-beta steady-state operation in the use of low radio-activation ferritic steel towards improvements in economical and environmental attractiveness in tokamak reactors in national-wide collaboration with universities, institutes and industries in Japan. This planning is under discussion at governmental committees.

Acknowledgements

The authors appreciate contributions from the JT-60 Team, Superconducting Magnet Laboratory and NBI Heating Laboratory at JAERI.

References

[17] Tsuzuki et al., IAEA-CN-94/EX/C1-1; K. Kamiya et al., IAEA-CN-94/EX/P2-05
Objectives and Design of the JT-60 Superconducting Tokamak


Japan Atomic Energy Research Institute, Tokai University, University of Toyama, Osaka University, Kyushu University, Koto University, Hokkaido University, Tokyo Institute of Technology, the University of Tokyo, Kyoto University, Mie University, Toshiba Corporation Power Systems and Services Company, Hiroshima University, Central Research Institute of Electric Power Industry, Sendai University, JICR, Nagaoka University, National Institute for Fusion Science, National Institute of Advanced Industrial Science and Technology

19th IAEA Fusion Energy Conference
October 14-18, 2002
Lyon, France

Objectives and Issues of JT-60 Modification

- Objectives:
  - to realize high-beta steady-state operation in the use of low radio-activation ferritic steel in low \( v^* \) and \( \rho^* \) regime relevant to reactor plasmas

- Research Issues:
  - Realization of high-beta steady-state operation
    - High beta plasma control \((\beta_n = 3.5 - 5.5)\)
    - Steady state plasma control \((f_{\rho} = 50 - 90\%)\)
    - Divertor heat & particle control \((f_{\rho} = 95\% \text{, } f_{\rho} = 5\%)\)
    - Disruption control (avoidance, mitigation)

2) Demonstration of the use of ferritic steel widely surrounding the plasma
  - Ideal MHD mode control with resistive and ferromagnetic wall
  - Locked mode and NTM control with additional error fields
  - Heat & particle control with plasma-material interaction

Important Issues towards Fusion Plant

- Further improvement in economical and environmental attractiveness is required for commercial fusion plants.

<table>
<thead>
<tr>
<th>Economical attractiveness</th>
<th>Environmental attractiveness</th>
</tr>
</thead>
<tbody>
<tr>
<td>( R_{max} = 13T )</td>
<td>Stainless steel</td>
</tr>
<tr>
<td>Thermal efficiency 45%</td>
<td>Low-activation ferritic steel</td>
</tr>
<tr>
<td>( R_{max} = 2.5 )</td>
<td>Acceptable level for safety level burial</td>
</tr>
<tr>
<td>( R_{max} = 4.5 )</td>
<td>To 200 times</td>
</tr>
<tr>
<td>( R_{max} = 5.5 )</td>
<td>High fly weight ratio</td>
</tr>
<tr>
<td>SST design</td>
<td>3,000 times</td>
</tr>
</tbody>
</table>

Realization of high-beta steady-state operation
Demonstration of the use of low radio-activation ferritic steel

Modification To a Fully Superconducting Tokamak

Requirements for plasma scale and duration:
- \( R_0 = 3 \text{ m} \rightarrow \text{low (} \rho^*, v^* = 0.01 \text{) plasmas close to reactor plasmas} \)
- \( \tau_P = 100 \text{ s} \rightarrow \text{sufficiently longer duration than current diffusion time} \)

Plasma performance with the existing heating facilities:
- Production of \( Q_{PP} = 1 \text{ plasmas is made possible for } 15 \text{ MW/100 s.} \)

Neutral beam injector
Superconducting coils
Neutral beam injector

<table>
<thead>
<tr>
<th>Parameter</th>
<th>JT-60SC</th>
</tr>
</thead>
<tbody>
<tr>
<td>Duration</td>
<td>100 s</td>
</tr>
<tr>
<td>( P_{in} )</td>
<td>64 MW (10 s)</td>
</tr>
<tr>
<td></td>
<td>15 MW (100 s)</td>
</tr>
<tr>
<td>( I_p )</td>
<td>4 MA</td>
</tr>
<tr>
<td>( B_0 )</td>
<td>3.0 T (R0=2.8 m)</td>
</tr>
<tr>
<td>( R_0 )</td>
<td>2.8 - 3.2 (2.8 m²)</td>
</tr>
<tr>
<td>( e )</td>
<td>0.7-0.9 (0.85 m²)</td>
</tr>
<tr>
<td>( S_p )</td>
<td>5.2 (1.8)</td>
</tr>
<tr>
<td>( S_{gg} )</td>
<td>5.6 (0.35)</td>
</tr>
</tbody>
</table>

* Nominal
Potential for Extension of Performance and Duration to Steady State Regime

- The JT-60SC target area is compared with reactor designs and the data on full current drive advanced discharges in JT-60U.

- Full current drive high-performance plasmas were achieved in JT-60U with $\beta_n \approx 1.3-2.3$ for high-$\beta_n$ and reversed shear discharges.

- Operation scenarios in JT-60SC can be projected to the target area.

High Beta Plasma Control

- ERATO-J code analysis shows that ideal kink–ballooning modes can be stabilized up to $\beta_n = 5.5$ with ideal wall at $r_w / a \approx 1.3$.

- Resistive wall modes can appear above $\beta_n = 3$, where 18 sector coils placed in the vacuum vessel are used for suppression.

- Attainable $\beta_n$ for long-pulse and high power heating capabilities for 4 MA and 1.5 MA in JT-60SC showing the capability to achieve $\beta_n = 3.5-5.5$.

NTM Regime and Suppression

- NTM onset in JT-60SC is predicted to be at $\tau_{ext} \approx 1.0-2.2$.

- Feedback stabilization of NTM in JT-60U succeeded. [Hayama et al., EX/2-2]

- NTM stability analysis using ECCD and TOPICS codes for JT-60SC shows the possibility of faster stabilization and power saving with early EC wave injection.

High Performance Full Current Drive

- High performance full CD capabilities are investigated for 3 MA, 3.8 T and $P_W = 30$ MW (30 s) using ACCOME code.

- This shows high-performance full current drive capability with $I_0 / I_e = 65\%$, $\beta_n = 3.5$ and $n_e T_e = 1 \times 10^{17}$ keV/cm$^3$ for HH-2 at $n/\tau_{ext} = 0.6$. 
**Steady State Plasma Control**

- Discharge scenarios are investigated by using the TOPOCS code on the basis of the JT-60U profile data for reversed shear plasmas.
- The time dependent transport analysis shows the potential of steady state operation at $\beta_n=5$ for 100 s stable for kink-ballooning modes.

**Effects of the Use of Ferritic Steel**

- **Fast ion loss**
  - Fast ion loss on the first wall is calculated with ferritic steel.
  - Fast ion loss is reduced from 25% to 2% for perpendicular beams.
- **Ferromagnetic effect on MHD**
  - Simple modeling calculation suggests:
    - Critical beta is reduced by ~10%.
    - Growth rate rises even with substantial flow velocity.
  - These effects would require larger amplifier gain of vector coils or plasma flow for stabilization.

**Plasma Facing Components**

- Divertor heat & particle control:
  - Semi-closed vertical target divertor compatible with high beta configuration.
  - Strong (~50 m/s) and separate pumping from inner and outer divertor for detachment control.
  - Forced cooling CFC target to remove ~10 kW/m² for 30 kW with $f_{tor}=400$.

- Arrangements of ferritic steel:
  - Ferritic steel arrangements could produce error fields less than 1 Gauss with low-n mode components, but lower than onset condition for locked modes at $\beta_n=3.5$.

**Bird’s Eye View of JT-60SC**

- JT-60SC is enclosed in a cryostat with a diameter of 12 m and a total weight of ~1700 tons.

**Superconducting coils**

- **Toroidal field coils**
  - Number: 18
  - $B_{max}$: 7.4 T
  - Conductor: Nb$_3$Al

- **Center solenoid**
  - Number: 4
  - $B_{max}$: 7.4 T
  - Conductor: Nb$_3$Sn

- **Equilibrium field coils**
  - Number: 6 (div. coil)
  - $B_{max}$: 5.1 (7.4 T)
  - Conductor: NbTi (Nb$_3$Sn)

[Sakayasu et al., JT/P2-09]
Summary

- The modification program of JT-60 to a fully superconducting tokamak is being planned under nation-wide collaboration in Japan.
- The program aims at realizing high-beta steady-state operation in the use of ferritic steel for reactor-relevant plasmas.
- Plasma performance addressing the research issues is shown to be feasible under the maximum utilization of the existing heating facilities.
- Basic design for the modification based on R&D for superconducting coils is presented.
- This planning is now under discussion at governmental committees.
1.21 Ferromagnetic and Resistive Wall Effects on Beta Limit in a Tokamak

G. Kurita, T. Tuda, S. Ishida, S. Takeji, A. Sakasai, M. Matsukawa, T. Ozeki, M. Kikuchi, M. Azumi

Naka Fusion Research Establishment, Japan Atomic Energy Research Institute, Mukoyama, Naka-machi, Naka-gun, Ibaraki-ken, Japan 311-0193

e-mail contact of main author: kurita@naka.jaeri.go.jp

Abstract. Ferromagnetic and resistive wall effect on beta limit in a tokamak is investigated. It is shown that the beta limit is reduced to 90% of that without ferromagnetic effect for high aspect ratio tokamak, if the ferromagnetic wall of relative permeability of 2 is used. The effect of toroidal plasma flow is also investigated, and the flow velocity of 0.03νₐ, νₐ is toroidal Alfvén velocity, is sufficient for the resistive wall to have stability effect of ideal wall. Both the resistive wall and ideal kink modes are destabilized by the ferromagnetic wall effects.

1. Introduction

In order to improve economic and environmental suitability of tokamak fusion reactors, both the accomplishment of high beta plasmas and the practical use of low activation materials to reduce the amount of radioactive waste are crucially important [1]. Although low radioactivation ferritic steel is considered as a most promising candidate for structural material in DEMO reactors, the influence of a ferromagnetic property in the ferritic steel on MHD stability and beta limits has been poorly investigated so far [2]. The effect of ferritic steel on MHD stability can be regarded as an additional factor to deteriorate the stability in a close relationship with stability for resistive wall mode (RWM) [3]. This paper finds substantial influences of residual magnetism in passively stabilizing wall on ideal MHD stability, i.e., "ferromagnetic wall mode", even though the ferromagnetism is sufficiently saturated at a high toroidal field (typically, μ/μ₀~2) and shows evaluations of deterioration of the beta limit due to the ferromagnetic property for the first time: where μ and μ₀ denote the permeability of ferromagnetic wall and vacuum, respectively. The toroidal flow effect on ferromagnetic and resistive wall mode is also investigated.

2. Basic analysis of ferromagnetic wall effect on kink mode

The roles of the ferromagnetic wall on the MHD stability is the twofold; the attraction of the perturbed magnetic field and the enhancement of the local skin time. The first one effectively moves the wall far from the plasma, even further than the infinity, and then widens the unstable regime of safety factor. The second reduces the growth rate and is

![Fig.1 Growth rate of n=1 free boundary kink mode versus safety factor of uniform current cylindrical tokamak with permeability effect. The permeability increases the growth rate of kink mode, especially for low qₐ region, and the stability window is reduced, amount of which is shown by distance between two arrows.](image-url)
stabilizing. However, the analysis (see Appendix) shows that the enhancement of the local skin time is canceled out by the magnetic field compression and that the effective skin time on the MHD stability is expressed by the vacuum permeability; \( \tau_\eta = \mu_0 \eta_\alpha d/(7m) \eta_\omega \) where \( \eta_\omega \) is the wall resistivity, \( \eta_\alpha \) and \( d \) are the plasma and wall minor radii and the wall thickness, \( m \) is the poloidal mode number. These features were confirmed by the numerical simulation of the free boundary kink mode in a cylindrical plasma with ferromagnetic and resistive wall (see FIG.1) and it was shown that, even for the almost saturated state of the permeability, the ferromagnetic wall has the considerable effect on the MHD stability. The increment of the unstable \( q \) regime for the uniform current is in good agreement with the analytic evaluation of \( \delta \eta_\omega = (\alpha/\eta_\omega)^2 (\mu/\mu_0 - 1) m d/(2r_w) \).

3. Critical beta analysis with and without ferromagnetism

In the present paper, the stability for ferromagnetic and resistive wall modes are analyzed using the linear MHD code, AEOLUS-FT, based on the original resistive MHD equations developed at JAERI. The linearized resistive MHD equations with plasma flow and permeability effect are shown below,

\[
\begin{align*}
\rho_0 \frac{\partial \vec{v}}{\partial t} &= -\rho_0 (\vec{v}_0 \cdot \vec{V}) \vec{v} - \vec{V} p + (\vec{j}_0 \times \vec{b}) + (\vec{V} \times (\vec{b}/\mu) \times \vec{b}) \\
\frac{\partial \vec{b}}{\partial t} &= \vec{V} \times (\vec{v}_0 \times \vec{b} + \vec{v} \times \vec{b}_0 - \eta \vec{V} \times (\vec{b}/\mu)) \\
\frac{\partial p}{\partial t} &= -((\vec{v}_0 \cdot \vec{V}) p - (\vec{v} \cdot \vec{V}) p_0 - \Gamma p_0 \vec{V} \cdot \vec{v})
\end{align*}
\]

Here, subscript 0 denotes the equilibrium quantity, \( \mu (= \mu/\mu_0) \) is relative permeability and \( \Gamma \) is specific heat ratio. Applicability and accuracy of this code for fixed boundary problem were confirmed by a benchmark test with the FAR code developed at ORNL[4]. For free boundary problem, the "pseudo-vacuum" model [5, 6] is used instead of "real vacuum" where the vacuum is replaced by highly resistive plasma, in the AEOLUS-FT code. In the following numerical calculations, the time is normalized to the poloidal Alfvén transit time \( \tau_\text{AE} = \sqrt{\rho_c R} B_\text{t} \) where \( R \) is major radius and \( B_\text{t} \) is toroidal magnetic field, and the ferromagnetic and resistive wall is assumed to surround the plasma uniformly and the distance between the wall and the plasma is also uniform.

3.1 Resistive wall mode without ferromagnetism

In order to look at MHD stability of the resistive wall mode without ferromagnetism in the wall, we investigate the plasma surface safety factor dependence on the growth rate for the plasma with a uniform current profile and parabolic pressure profile, a circular plasma cross section and a high aspect ratio without ferromagnetism. The AEOLUS-FT code analysis shows that, by changing the resistivity of the wall from \( \eta_\omega = 1 \) (representing "pseudo-vacuum") to \( \eta_\omega = 10^4 \) and \( 10^6 \), the growth rates of \( n=1 \) modes are reduced from the growth rate for free-boundary kink mode to that of resistive wall mode, where the obtained growth rate is consistently shown to be of the order of the inverse time constant of resistive wall. From these calculations, the dependence of the growth rate on \( \eta_\omega \) for low beta and high aspect ratio plasma, is found to be almost the same as cylindrical analysis [7], indicating the validity of the AEOLUS-FT code calculation. Here used are a circular plasma cross-section and a high aspect ratio, \( q_\alpha = 2.5 \) and the minor radius of resistive wall of 1.14a and a resistivity of the wall fixed at \( \eta_\omega = 10^4 \). The poloidal mode numbers are taken into account from \( m=1 \) to \( m=10 \). The number of non-uniform grid points in the minor radius direction is typically 2000.
3.2 Dependence of critical beta on permeability

Under the above conditions with parabolic current and pressure profiles, the dependence of the $n=1$ mode growth rate on the poloidal beta for the plasma is obtained from the AEOLUS-FT code analysis, where the thickness of the wall is fixed at $d=0.07a$ and the permeability in the ferromagnetic and resistive wall is changed from $\mu/\mu_0=1$ to 8. As shown in FIG.2, the growth rates clearly increase and the critical poloidal beta values are substantially reduced down to 90% at $\mu/\mu_0=2$, 78% at $\mu/\mu_0=4$ in comparison with the critical beta value at $\mu/\mu_0=1$. Figure 3 shows the comparison of the mode structures of $m=3/n=1$ with and without ferromagnetism corresponding to the cases shown in FIG.2. This figure clearly represents a feature of magnetic field attraction due to ferromagnetism in the wall in comparison with the resistive wall without ferromagnetism.

3.3 Dependence of critical beta on thickness of ferritic wall

Effects of the thickness of the ferromagnetic and resistive wall on the critical beta can appear as a competition between stabilizing and destabilizing effects caused by skin time and ferromagnetism, respectively. When the wall thickness is increased with the inner wall radius fixed from $d=0.07a$ to 0.11a and 0.14a, the critical beta increases as 1.49, 1.67 and 1.70, respectively, due to increasing the skin time if the permeability effect is not taken into account ($\mu/\mu_0=1$). However, with the permeability effect, the critical beta saturates or even decreases with the thickness of the wall above a threshold value of the thickness since the destabilizing effect due to ferromagnetism becomes larger. Indeed, for the case of $\mu/\mu_0=2$, the critical beta value is increased from 1.34 at $d=0.07a$ to 1.41 at $d=0.11a$, but is decreased to 1.39 at $d=0.14a$.

4. Effect of toroidal plasma flow

The effect of toroidal or poloidal plasma flow has been considered to play an important role in stabilizing resistive wall mode [8, 9] with the effect of viscous damping [3, 10]. We investigate the effect of toroidal plasma flow on resistive wall and ideal kink modes using AEOLUS-FT code, which solves the complex eigen-value problem. We use the same analytical equilibrium as that in section 3 (large aspect ratio with parabolic profiles for both plasma current and pressure) with poloidal beta of 1.8. The equilibrium is sufficiently unstable for $b/a=1.43$. In these calculations, we use the rigid plasma rotation and change the position of resistive wall for different flow velocity. The calculated growth rates versus the position of resistive wall, $r_w/a$, are shown in FIG.4. For no rotation case, the growth rate shows the monotous decreasing function of resistive wall position. For high rotation case, on the other hand, they appears steep decreasing function at large values of $r_w/a$ as ideal wall.
branch, and appears again at small value region as resistive wall branch. The growth rates of ideal wall branch tends to that of ideal wall case for sufficient large flow speed, $v_{\phi 0} = 0.03v_a$ ($v_a$ is toroidal Alfvén velocity). The growth rates in the interm region are expected to be stabilized, if we incorporate the effect of viscous damping for equilibria of small aspect ratio tokamak [3,10]. Figure 5 shows the growth rate for $v_{\phi 0} = 0.06v_a$ flow velocity case with the ferromagnetic effect. Growth rates are more increased for larger values of relative permeability for all calculation region.

![Graph 4: Growth rate versus wall position for 3 toroidal flow velocity values. Relative permeability value is 1.]

![Graph 5: Growth rate versus wall position for 3 relative permeability values. Toroidal flow velocity is 0.06va](image)

5. Conclusion

In conclusion, the presence of ferromagnetic wall mode is identified as the critical beta is reduced to 90% of that without ferromagnetism with a wall thickness of 0.07a for $\mu/\mu_0 = 2$ at which the ferritic steel is sufficiently saturated. Even though the skin time in the wall is increased with the wall thickness, the ferromagnetism can suppress the improvement in the critical beta or decrease it if the wall thickness becomes larger than a threshold value. The effect of toroidal plasma flow is also investigated, and the flow velocity of 0.03v_a, $v_a$ is toroidal Alfvén velocity, is sufficient for the resistive wall to have stability effect of ideal wall. Both the resistive wall and ideal kink modes are destabilized by the ferromagnetic wall effects. These results would have an impact on reactor designs utilizing ferritic steel material with ferromagnetism. Finally, we note the above roles of ferromagnetic wall, that is, the attraction of the magnetic perturbation and the resultant reduction of the MHD stability, is basically independent on the plasma model, like the inclusion of the viscous damping term, and in this sense, the ferromagnetic wall mode is the generic one. However, the critical beta value is affected by details of the plasma model and the configuration, and the quantitative evaluation of it is now under way, including the effect on the feedback control.

Acknowledgements

The authors would like to express their sincere thanks to Drs. A. Kitsunezaki and H. Ninomiya for their fruitful discussions and continuing encouragement.

References


Appendix

In order to understand the basic features concerning the ferromagnetic wall effects on the MHD stability, we study MHD mode with poloidal number \( m \) in a cylindrical plasmas, under the assumption of long wavelength limit. We locate the ferromagnetic wall with resistivity \( \eta_w \), permeability \( \mu \) and width \( d \), at \( r=r_w \), and the perfect conductor wall at \( r=b \), while the plasma boundary at \( r=a \). The perturbed helical magnetic flux \( \psi \) in the vacuum regions \((a<r<r_w \) and \( r_w+d<r<b)\) and inside the ferromagnetic wall \((r_w<r<r_w+d)\) are analytically solved and these are connected with each other by using the boundary conditions: \( [\psi]=0 \) and \( [\psi/\mu]=0 \) at both sides of the ferromagnetic wall \((r=r_w \) and \( r=r_w+d)\), where \( \psi' \) is the radial derivative of \( \psi \) and \( [f]=f(x+0)-f(x-0) \). Then the resultant vacuum solution takes the following form:

\[
\frac{\Delta^*_a + 1}{\Delta^*_a - 1} = \left( \frac{l'_{m+} + \hat{\sigma}l_{m+}}{(K'_{m+} + \sigma I_{m+})} - \frac{(l'_{m+} + \sigma K_{m+})(l'_{m+} + \sigma I_{m+})}{(K'_{m+} + \sigma K_{m+})(I_{m+} - \sigma I_{m+})} \right) \left( \frac{a}{r_w} \right)^{2m}
\]

where \( I_{m+} \) and \( K_{m+} \) are the 1\textsuperscript{st} and 2\textsuperscript{nd} modified Bessel function of \( m \)-th order with argument \( \kappa r_w \), and \( l_{m+} \), and \( K_{m+} \) are those with argument \( \kappa (r_w+d) \),

\[
\kappa^2 = \gamma \mu/\eta_w, \quad \sigma = \mu/m \kappa r_w, \quad \hat{\sigma} = \sigma r_w (r_w + d), \quad \hat{\mu} = \mu/\mu_0, \quad \Delta^*_a = a \psi'/m \psi |_{r=a+c}.
\]

\( \gamma \) is the growth rate of the mode. By connecting \( \Delta^*_a \) with the solution of \( \psi \) in the plasma, the dispersion relation of the mode is obtained. In the case of the thin ferromagnetic wall, taking the first order of \( d/r_w \), the above equation is expressed by the following simplified form:

\[
\frac{\Delta^*_a + 1}{\Delta^*_a - 1} = \frac{\gamma \tau_w - (md/2r_w)(\hat{\mu} - \hat{\mu}^{-1})}{1 + \gamma \tau_w + (md/2r_w)(\hat{\mu} + \hat{\mu}^{-1} - 2)} \left( \frac{a}{r_w} \right)^{2m}
\]

where \( \tau_w = \mu_0 \partial \psi/2 \eta_w \) is the skin time of the ferromagnetic wall. Note that this skin time is independent on the permeability of the wall, which means the enhancement of the local skin time \( \tau_w = \mu_0 \partial \psi/2 \eta_w \) is compensated by the flux compression through the boundary condition. This equation also shows that the ferromagnetic wall makes the mode unstable even for the high conductivity. The growth rate of the MHD mode can be expressed as,

\[
\gamma \tau_w = \Gamma + \Gamma_\mu
\]

where

\[
\Gamma_w = \frac{\alpha}{(a/r_w)^{2m} - \alpha}, \quad \Gamma_\mu = \left( \frac{md}{2r_w} \right) \frac{\alpha (\hat{\mu} + \hat{\mu}^{-1} - 2) + (\hat{\mu} - \hat{\mu}^{-1}) (a/r_w)^{2m}}{(a/r_w)^{2m} - \alpha}
\]

and \( \alpha = (\Delta^*_a + 1)/(\Delta^*_a - 1) \) is generally the function of growth rate determined by the plasma dynamics. For the range where the ideal mode is stable and the plasma inertia is neglected, the parameter \( \alpha \) is independent on \( \gamma \) (for uniform current case, \( \alpha = n \eta_w - m + 1 \)). Then the first term of the right hand side gives the resistive wall mode and the second term shows that the high permeability enhances the mode growth rate and widens the unstable region.
Objectives of JT-60 Super-Conducting Modification

- Extension of the JT-60 program is oriented in support of ITER under maximum utilization of the present facilities.
- The scene for the extension is being set at JAERI in national wide collaboration with universities, institutes and industries.

1) ESTABLISHMENT OF HIGH PERFORMANCE STEADY STATE OPERATION
   - HIGH BETA PLASMA CONTROL \( (\beta_B = 3.5 - 5.5) \)
   - STEADY STATE Plasma CONTROL \( (\beta_B = 50 - 90\%) \)
   - DIVERTOR HEATAPARTICLE CONTROL \( (\langle n_e \rangle \approx 95\% \quad \langle n_e \rangle / q = 5 \) \)
   - DISRUPTION CONTROL (avoidance, mitigation)

2) PLASMA APPLICABILITY TEST OF ADVANCED MATERIALS
   - Demonstrate the potential of practical use for advanced materials such as low activation ferritic steel with reactor-relevant plasma

Parameters for JT-60 Super-Conducting Modification

- JT-60SC pursues plasma parameters deduced from DEMO concepts with high \( \beta \) using low activation ferritic steel in the vacuum vessel.
- For achievement of the high \( \beta \) plasma, RWM study with ferromagnetic wall effect becomes crucially important.

<table>
<thead>
<tr>
<th>Parameter</th>
<th>JT-60U</th>
<th>JT-60SC</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pulse length</td>
<td>15 s</td>
<td>160 s</td>
</tr>
<tr>
<td>Max. power</td>
<td>40 MW (10 s)</td>
<td>44 MW (10 s)</td>
</tr>
<tr>
<td></td>
<td>15 MW (100 s)</td>
<td></td>
</tr>
<tr>
<td>Plasma current ( I_p )</td>
<td>3.5 MA</td>
<td>4 MA</td>
</tr>
<tr>
<td>Toroidal field ( B_t )</td>
<td>4 T</td>
<td>3.81 ( (B_p = 2.8 \text{ m}) )</td>
</tr>
<tr>
<td>Major radius ( R_p )</td>
<td>3.4 m</td>
<td>2.5 - 3 m ( (2.8 \text{ m})^2 )</td>
</tr>
<tr>
<td>Minor radius ( a_p )</td>
<td>0.9 m</td>
<td>0.7 - 0.9 m ( (0.85 %) )</td>
</tr>
<tr>
<td>Elongation ( \varepsilon )</td>
<td>1.8 ( (\varepsilon_p = 1.38) )</td>
<td>1.2 ( (1.7^\circ) )</td>
</tr>
<tr>
<td>Triangularity ( \delta_p )</td>
<td>0.4 ( (\delta_p = 0.06) )</td>
<td>0.5 ( (0.35^\circ) )</td>
</tr>
</tbody>
</table>
Full Set of Resistive MHD Equations

- Linearized resistive MHD equations with plasma flow
  \[ \rho \frac{D}{Dz} \theta = -\rho v \nabla \theta - D_p + \left( \frac{\beta_p}{\beta} \right) \left( \nabla \times \eta \nabla \theta / \mu \right) \]
  \[ \frac{\partial \theta}{\partial t} = 0 \]
  \[ \frac{\partial \psi}{\partial t} = -v \nabla \theta \]
  \[ \nu : \text{plasma flow} \]
  \[ \mu : \text{relative permeability} \]
  \[ \Gamma : \text{specific heat ratio} \]

- Pseudo-vacuum model
  - parabolic current profile
  - resistivity profile
  \[ \eta(\rho) = 10^{-8} : \text{in plasma} \]
  \[ \eta(\rho) = 10^{5} = \text{in pseudo-vacuum} \]
  \[ \eta(\rho) = 10^{-4} : \text{in resistive wall} \]

Analysis of RWM in JT-60 Super-Conducting Modification

- Eddy current pattern at the resistive wall surface in toroidal-polaroid real plane induced by RWM (r/a = 1.16)

- Fourier components of eddy current in toroidal direction induced at the surface of resistive wall due to RWM.

- Two red points shown in the middle figure correspond to two red points shown in the right figure.

- As beta increases, higher number poloidal modes are excited in phase as shown in the left figure, and the mode pattern localizes in low field side.

RWM Analyses of JT-60 Super-Conducting Modification

- Vacuum is represented by pseudo-vacuum occupied by high resistive plasma.

Control of RWM in JT-60SC

To obtain high \( \beta \) plasma in JT-60SC, sector coils of 6 x 3 (18 coils) are planned to be installed for RWM control.

Sector Coil Configuration

Cross Section of JT-60SC Device
**Effect of Ferritic Wall on MHD Stability**

To achieve higher $\beta$ beyond no wall $\beta$-limit, stabilization by the conducting wall is necessary.

Low activation ferritic steel, the first candidate of the wall material, is known as strong ferromagnetic material, and its effect on MHD stability has been considered to be important.

To investigate the ferritic wall effect on MHD stability of RWM, we have developed the linear MHD code, AEOLUS-FT.

As an initial stage to include the effect of ferritic wall on the full MHD calculation, fundamental characteristics for cylindrical and high aspect ratio tokamak plasma are presented here.

**Basic Characteristic of Ferritic Wall Effect on Kink Mode**

Cylindrical uniform current model:

- ferritic wall radius : $r_w$
- ferritic wall thickness : $d$

For $\gamma r >> 1$, we have Shafranov's growth rate of kink mode,

$$\gamma = \frac{2m}{\rho_d} - \frac{1}{1 - a/c_s}$$

where

$$\rho_d = \frac{m - q_n - 1}{2\pi}$$

Marginal point is moved by

$$a = \frac{\mu_0}{1 - \beta_c}$$

* Effective skin time, $\tau_s$, is not changed by $\mu$

* Unstable q region expands beyond $m - 1$ due to destabilization by spatial change of relative permeability in no ideal wall case, $\rightarrow$ disappearance of stability window.
Check the results of AEOLUS-FT Code with pseudo-vacuum I

- Analytical equilibria with uniform current profile are used.
  \[ p(r) = p_0 \left( 1 - \frac{r}{a} \right)^{2/1.1} \]
  \[ b/a = 1.43 \cdot \frac{r_{nf}}{a} = 1.143 \]
  \[ A = 14.3, \quad \Gamma = 5/3 \]
  \[ \eta_{\text{shear}} = 10^{-4}, \quad \frac{\eta_{\text{visc}}}{\eta_{\text{visc}}^{\text{bulk}}} = 10^{-3} \]
  \[ \eta_{\text{magn}} = 10^{-4}, \quad \frac{\eta_{\text{visc}}}{\eta_{\text{visc}}^{\text{bulk}}} = 10^{-3} \]

- Growth rates of toroidal kink mode are in good agreement with analytical cylindrical growth rates of kink mode except for \( q_n \) around integer values.
- Resistive wall mainly reduces the growth rate at low \( q \) side of kink mode, where RWIs appear and their growth rates increase with beta value.

Check the results of AEOLUS-FT Code with pseudo-vacuum II

- Analytical equilibrium with parabolic current profile
  \[ p(r) = p_0 \left( 1 - \frac{r}{a} \right)^{2/1.1} \]
  \[ b/a = 1.43 \]
  \[ A = 14.3, \quad \Gamma = 5/3 \]
- Numerical equilibrium with uniform current profile
  \[ \frac{r_{nf}}{a} = 1.143 \]
  \[ \eta_{\text{shear}} = 10^{-4} \]
- Growth rate of low \( \beta \) kink mode are in good agreement with cylindrical growth rates also for parabolic current profile. (Cylindrical growth rates are obtained by reduced MHD code.)
- The growth rate calculated for numerical equilibria with uniform current with and without resistive wall shows almost the same as that for analytical one.
- Stability calculations with numerical equilibria are also confirmed.

Reduction of Critical Beta Value by Ferritic Wall Effect

- Analytical equilibria with parabolic current profile are used.
  \[ p(r) = p_0 \left( 1 - \frac{r}{a} \right)^{2/1.1} \]
  \[ b = 1.43 a, \quad r_{nf} = 1.143 a, \quad \eta_{\text{magn}} = 10^{-4} \]
  \[ A = 14.3, \quad q_n = 2.5 \]

- Critical beta value is reduced to 90% of that without ferritic wall effect for relative permeability of \( \mu_p = 2 \), and is reduced to even about 50% for \( \mu_p = 4 \).
- The eigen-functions of poloidal flux, \( \psi(r) \), becomes steeper in the ferritic wall for larger values of relative permeability, which indicates large flux absorption.
- The critical beta becomes a decreasing function of the thickness of the wall, \( d_w \), if ferritic wall effect is considered.

Effect of Toroidal Plasma Flow on RWM

- Rigid toroidal plasma flow
  \[ \psi(r) = \psi_0 \left( 1 - \frac{r}{a} \right) \]
  \[ b = 1.43 a, \quad \eta_{\text{magn}} = 10^{-4} \]
  \[ A = 14.3, \quad q_n = 2.5 \]

- For higher toroidal plasma flow, the reduction of growth rate occurs.
- The toroidal plasma flow of \( \psi_{\text{bulk}} \) is poloidal Alfvén velocity, seems to be sufficient for resistive wall to have stabilizing effect of ideal wall.
- Velocity difference between background rotation and mode rotation, shown by broken lines, becomes large as resistive wall radius reduces in RWM region.
Effect of Ferritic Wall on RWM with Toroidal Plasma Flow

- Rigid toroidal plasma flow
- $l/r = \log \{1 - (r/a)^2\}^{1.02}$
- $p(r) = p_0 \{1 - (r/a)^2\}^{1.1}$
- $b = 1.43 a$, $\eta_w = 10^{-4}$
- $A = 14.3$, $\eta_w = 2.5$
- $n_1 = 10$, $n_2 = 1$
- $N_r = 2001$

with mesh accumulation

- Increase of relative permeability of the wall increases the critical toroidal plasma flow for resistive wall to have stabilizing effect of ideal wall.

Summary & Future Works

Summary

- Linear MHD code using full set of resistive MHD equations is developed for analyses of RWM.
- Critical beta reduces to about 90% of that without ferromagnetic effect in the case of relative permeability of 2.

Future Works

- To include the ion sound wave damping effect in realistic, low aspect ratio, configuration.
- Feedback calculation using magnetic field made by sector coils
1.22 Design and Technology Development of Solid Breeder Blanket Cooled by Supercritical Water in Japan

M. Enoeda, Y. Kosaku, T. Hatano, T. Kuroda, N. Miki, T. Honma and M. Akiba

Japan Atomic Energy Research Institute, Naka-machi, Naka-gun, Ibaraki-ken, Japan

e-mail contact of main author: enoedam@fusion.naka.jaeri.go.jp

Abstract. This paper presents results of conceptual design activities and supporting R&D's of a solid breeder blanket system for the demonstration of power generation fusion reactors (DEMO blanket), which is cooled by supercritical water. The Fusion Council of Japan developed the long-term research and development program of the blanket in 1999. Among the program, Japan Atomic Energy Research Institute has been assigned as a hub institute for developing a solid breeder blanket system in Japan. To make the fusion DEMO reactor more attractive, higher thermal efficiency of more than 40 % has strongly been envisaged. The design work has shown the feasibility of the first wall thermo-mechanical performance and tritium breeding performance of the blanket. In parallel with the design activities, engineering R&D's have extensively been conducted, which cover all necessary issues, such as, material development for structural materials, tritium breeding materials and neutron multiplier materials, neutronics experiments and analyses, and development of the fabrication technology of the blanket module.

1. Introduction

The Fusion Council of Japan has established the long-term research and development program of the blanket in 1999. In the program, Japan Atomic Energy Research Institute has been designated as a leading institute for developing a solid breeder blanket system in Japan. To make the DEMO reactor more attractive, higher thermal efficiency of more than 40 % has been strongly envisaged. From this viewpoint, the conceptual design of the DEMO reactor has been performed by JAERI recently, aiming at the achievement of similar plasma performance, such as fusion power, Q value, and neutron wall load with more economical attractiveness [1]. In line with the reactor design proposed, the DEMO blanket design has been intensively conducted. Major design parameters of the DEMO blanket are summarized in Table I. Load conditions and applied materials are similar to those of SST-2[2]. Therefore, past R&D results are available. Not only the design development, but also recent achievement of technology development was reported in this paper.

2. Design Development

One of most critical issues of the DEMO blanket design is the removal of a high heat flux of 1 MW/m² onto the first wall, while keeping the temperature of the first wall structure lower than 450 °C with exit coolant temperature, 510 °C. To solve this problem, a unique coolant flow pattern has been developed in this design. As can be seen in FIG. 1, the coolant with the inlet temperature, 280 °C first flows through the first wall area of the blanket modules, which are connected in series, and the coolant temperature is raised up to around 200 °C. Then, the coolant is discharged into the steam generator. Table 1 shows major design parameters of supercritical water cooled blanket.

<table>
<thead>
<tr>
<th>Item</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Surface heat flux</td>
<td>0.5 (peak 1) MW/m²</td>
</tr>
<tr>
<td>Neutron wall load</td>
<td>3.5 (peak 5) MW/m²</td>
</tr>
<tr>
<td>Neutron Fluence</td>
<td>&gt;10 MWea/m²</td>
</tr>
<tr>
<td>Coolant Material</td>
<td>Supercritical water</td>
</tr>
<tr>
<td>Coolant Pressure</td>
<td>25 MPa</td>
</tr>
<tr>
<td>Inlet / Exit Temperature</td>
<td>280/510 °C</td>
</tr>
<tr>
<td>Tritium Breeding Ratio</td>
<td>&gt;1.05</td>
</tr>
<tr>
<td>Structural Material</td>
<td>RAFS® (F82H)</td>
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<tr>
<td>Tritium Breeder</td>
<td>Li₂O or Li₂TiO₃</td>
</tr>
<tr>
<td>Neutron Multiplier</td>
<td>Be or Be₁₂Ti</td>
</tr>
</tbody>
</table>

* Reduced activation ferritic steel
380 °C at the exit of the first walls of the same series of modules. Then, the coolant flows into the breeding area of the blankets, which are also connected in series, and at the exit the coolant temperature of 510 °C can be obtained (FWs-to-Breeders Series Cooling Pattern). The thermal efficiency analysis of the cooling system showed that the thermal efficiency of more than 41 % is expected with this flow pattern by the heat balance calculation of the process flow diagram of the cooling system.

Detailed structure of the blanket module is shown in FIG. 2. Dimension of the blanket module is smaller than 2 m high, 2 m wide, and 0.6 m thick. Reduced activation ferritic steel, F82H, which is currently under development by JAERI, was selected as the structural material. Ceramic breeder and beryllium neutron multiplier are packed in a form of a small pebble in a layer structure as shown in the figure. Lithium ceramics, such as Li₂TiO₃ or Li₂O, was selected as the primary candidate tritium breeder material. Beryllium or inter-metallic compound, such as Be₁₂Ti, was selected as the neutron multiplier.

In the thermal and neutronics analyses of the breeding blanket, it is the most important point that the temperatures of the breeder and multiplier materials are required to be kept in the appropriate range without reducing the net tritium breeding ratio (TBR) less than 1.05, from the viewpoints of fuel self sufficiency and preparation of startup fuel for the next fusion plant. In this study, neutron and γ-ray spectrum analyses have been performed by using one dimensional S₉ code, ANISN with the group constant set, FUSION-40.

FIG. 1. Coolant Temperature Design by FWs-to-Breeders Series Cooling Pattern.

FIG. 2. Schematic structure of the supercritical water cooled blanket.

FIG. 3. Profiles of temperature and tritium breeding ratio along the thickness of the blanket.
TABLE II: RESULTS OF TBR CALCULATION WITH CANDIDATE OPTIONS OF MATERIALS AND STRUCTURE.

<table>
<thead>
<tr>
<th>Materials</th>
<th>Li$_2$O / Be</th>
<th>Li$_2$TiO$_3$/ Be</th>
<th>Li$_2$TiO$<em>3$/ Be$</em>{12}$Ti</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^6$Li Enrichment</td>
<td>30% 90%</td>
<td>30% 90%</td>
<td>30% 90%</td>
</tr>
<tr>
<td>Packing Structure</td>
<td>Breeder / Multiplier Separate</td>
<td>Breeder + Multiplier Mix</td>
<td></td>
</tr>
<tr>
<td>Temperature Limits</td>
<td>Breeder 900°C</td>
<td>900°C</td>
<td>600°C</td>
</tr>
<tr>
<td></td>
<td>Multiplier 600°C</td>
<td>900°C</td>
<td></td>
</tr>
<tr>
<td>Local TBR</td>
<td>1.53 1.56</td>
<td>1.41 1.52</td>
<td>1.37 1.24 1.35 1.35 1.43</td>
</tr>
<tr>
<td>Coverage Requirement</td>
<td>69% 67%</td>
<td>74% 69%</td>
<td>77% 85% 78% 78% 73%</td>
</tr>
</tbody>
</table>

* Required coverage fraction of the plasma facing surface of the breeding region of the blanket in the total area of the plasma facing surface, to achieve net TBR, 1.05.

Nuclear heating rate and TBR has been estimated by using APPLE-3 code. By using obtained values of nuclear heating rate, one dimensional thermal analysis has been performed to obtain temperature distribution [3]. FIGURE 3 shows the distribution of local TBR in radial thickness direction, together with the temperature distribution in case where 30% $^6$Li enriched Li$_2$TiO$_3$ and Be were applied as the breeder and multiplier materials. As can be seen from this figure, temperature of the breeder can be kept below the temperature limit of 900 °C. In this case, the local TBR reached 1.41, which satisfies the net TBR of 1.05 with 74% of coverage ratio of the blanket in the total plasma facing surface area in the vacuum vessel. Table II summarizes estimated values of TBR with major candidate options of materials, $^6$Li enrichment and structure. Li$_2$TiO$_3$ and Be$_{12}$Ti is expected to have better compatibility with water in high temperature than Li$_2$O and Be. Even in case Li$_2$TiO$_3$ or Be$_{12}$Ti are applied, net TBR satisfied more than 1.05.

In the design process, thermo-mechanical design is also another critical issue for the feasibility of the design. FIGURE 4 shows the results of temperature and stress analyses of the first wall structure by using ABAQUS code based on the heating and cooling conditions specified by the design requirement and coolant temperature design.

The highest temperature appears at the rear side of the first wall due to the apparent heat flux (about 0.4 MW/m$^2$) by the volumetric heating in the breeder zone. Peak stress appeared at the
corner of the cooling channel, however, it satisfies the 3Sm value of the reduced activation ferritic steel, F82H, at 500 °C (430 MPa). FIGURE 5 shows the estimated values of the temperature and stress of structural material in this design. As can be seen from FIG. 5, the stress by the internal coolant pressure in the cooling channels and pipes satisfied Sm or 1.5Sm value. Temperature estimation showed the temperature ranges from 400 °C to 570 °C. This result indicates the necessity of the incorporation of creep effect in the thermo-mechanical design.

The design work included the analyses of tritium inventory and permeation, diverter design, power plant design, tritium systems for purge gas tritium recovery and water detritiation, the design of the remote maintenance system and blanket replacement procedure. Preliminary integration of the design of a solid breeder blanket cooled by supercritical water was achieved in this study.

3. Supporting Technology R&D's

In parallel with the design activities, supporting technology R&D's have been extensively conducted, which cover development for structural materials [4], tritium breeding materials and neutron multiplier materials [5], neutronics performance experiments and analyses [6], evaluations of thermal characteristics of packed pebble bed [7] and development of the fabrication technology of the blanket module. Since the first wall with embedded cooling channels consists of rectangular cooling channels and flat panels, a hot isostatic pressing (HIP) technique has been applied for the first wall. A HIP-bonded F82H first wall mock-up with built-in rectangular cooling channels has been successfully fabricated, and tested to demonstrate the structural soundness under the accelerated heat flux condition of 2.7 MW/m². As can be seen from FIG. 6, it is confirmed that no degradation of the fatigue lifetime performance can be found in F82H after the HIP-bonding process [8].

For the thermal design of the breeder layer, effective thermal conductivity of a pebble bed is the most important characteristics. By using the representative pebble of Li₂TiO₃ fabricated by Sol-Gel method [5], the measurement has been performed by single packing pebble bed and binary packing pebble bed. FIGURE 7 shows the schematic structure of the measurement apparatus by hot wire method [7]. The test section simply consists of the sample pebble bed and the hot wire (PtRh heater) at the center in the axial direction. Effective thermal conductivity is calculated by observed transient temperature change of the hot wire. FIGURE 8 shows the measured values of effective thermal conductivities of Li₂TiO₃ pebble beds. By applying
binary packing of 1.91 mm diameter pebbles and 0.28 mm diameter pebbles, the effective thermal conductivity became 10 to 20% larger than single packing pebble bed, which met with the correlation estimation. By this result, the uncertainty of the thermal design of the breeder pebble bed was decreased. Also, the hot wire method was extended to measure the effect of the stress in the pebble bed to the effective thermal conductivity, which is the most important unknown data of the thermo-mechanical design of the pebble bed blankets[9,10].

4. Conclusions

(1) A supercritical water cooled solid breeder blanket was proposed as the advanced concept of the water cooled solid breeder blanket.
(2) Design development covered major critical issues. Preliminary integration of the design was achieved.
(3) Technology R&D’s progressed in the area of the first wall, box and pebble bed structure fabrication. The results of R&D’s were reflected to the blanket module design.

References

Design and Technology Development of Solid Breeder Blanket Cooled by Supercritical Water in Japan

M. Enoda, Y. Koseki, T. Hatano, T. Kuroda, N. Miki, T. Honma and M. Akiba
Japan Atomic Energy Research Institute

Introduction
(1) Japan Atomic Energy Research Institute has been designated as a leading institute for developing a solid breeder blanket system as the first candidate DEMO blanket in the R&D program of the blanket established by the Fusion Council of Japan.

(2) To make the DEMO reactor more attractive, higher thermal efficiency of more than 40 % has been expected.

(3) For attractive DEMO blanket, a new concept of the solid breeder blanket cooled by supercritical water has been proposed.

Design Development

New concept of the solid breeder blanket cooled by supercritical water is proposed. The design covered (1) the design of coolant flow path arrangement for blanket modules, (2) module structure design, (3) thermo-mechanical analysis of blanket module, (4) neutronics analysis of blanket module, (5) the analyses of tritium inventory and permeation, (6) diverter design, (7) tritium systems for purge gas tritium recovery and water detoxification, (8) the design of the remote maintenance system and blanket replacement procedure. Integrated design was achieved.

Structure Design of Solid Breeder Blanket Cooled by Supercritical Water for DEMO

Structural features and key materials
- Modular type, front access replacement on sight
- Box wall with embedded coolant channels
- Pebble bed type breeder and multiplier layers separate with cooling tubes and partition walls
- Supercritical water for coolant
- Coolant flow path arrangement to cool first walls first and, then, breeder and multiplier layers of multiple blanket modules
- Reduced activation ferritic steel (RAFS) for structural material
- L_2 TiO_3 or Li_2O for tritium breeding material
- Be or Be intermetallic compound for neutron multiplier
- Safety enhancement by selecting water resistive blanket materials
- Optional W coating for first wall protection

Structure of Typical Blanket Module
Height 1 m X Width 2 m X Thickness 0.6 m (Weight < 1 ton)
Total number of modules = 266 (16 modules x 16 sectors)

MAJOR DESIGN PARAMETERS OF THE SUPERCritical WATER COOLED BLANKET

<table>
<thead>
<tr>
<th>Item</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Surface heat flux</td>
<td>0.5 (peak) 1 MW/m²</td>
</tr>
<tr>
<td>Neutron wall load</td>
<td>3.5 (peak 5) MW/m²</td>
</tr>
<tr>
<td>Coolant Material</td>
<td>Supercritical water</td>
</tr>
<tr>
<td>Coolant Pressure</td>
<td>25 MPa</td>
</tr>
<tr>
<td>Coolant Temperature</td>
<td>280 °C</td>
</tr>
</tbody>
</table>

Coolant Temperature Estimation showed the reasonable set of flow rate and temperature distribution for each cooling loop. Thermal efficiency was estimated to be more than 41 % by the direct cycle which uses steam from blankets directly for the turbine.

PARAMETERS OF THE DEMO

<table>
<thead>
<tr>
<th>Item</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>R</td>
<td>5.6m</td>
</tr>
<tr>
<td>L_1</td>
<td>1.45 m</td>
</tr>
<tr>
<td>V</td>
<td>386 m³</td>
</tr>
<tr>
<td>S_1</td>
<td>12.5^2</td>
</tr>
<tr>
<td>S_2</td>
<td>20.9</td>
</tr>
<tr>
<td>P</td>
<td>2.3 GW</td>
</tr>
<tr>
<td>Q</td>
<td>40</td>
</tr>
<tr>
<td>t</td>
<td>4.3</td>
</tr>
</tbody>
</table>

The DEMO Fusion Reactor (Kondoh et al., ISFNT6)

FWs-Breeder Series Cooling Pattern and Thermal Efficiency of Power Plant

Coolant Temperature Estimation

Energetic Flow from FWs to Breeder Region

Coolant Temperature Estimation

Module No.

FWS-Breeder Series Coolant Flow Path.
Fabrication Technology Development

The fabrication technology development for generic water cooled solid blankets have shown progress on:

1. The fabrication technology development of first wall and box structure with embedded cooling channels.
2. Clarification of thermal characteristics of breeder and multiplier pebble beds.

FW and Box Structure

Comparison of fatigue data by high heat flux test of the mockup and reference data by IEA Round Robin showed preliminary soundness of the mockup fabrication by HIP. Improvement of HIP joining technique is continued.

Breeder and Multiplier Pebble Beds

Effective thermal conductivity was measured by Hot Wire Method. Hot wire method has such merit as, small amount of pebble specimen, uniform bed temperature and less than 10 °C of hot wire, short observation time. Data of Li2TiO4 pebble beds are measured and clarified the correlation parameter of effective thermal conductivity.

Related R&D Achievements in Japan

- Structural Material Development: FT1-1Ra
- Breeder and Multiplier Materials Development: FT/P1-09
- Neutronics Experiments for DEMO Blanket: FT/P1-10

Conclusions

1. A new design of a solid breeder blanket cooled by supercritical water was proposed. The blanket design has been performed in conjunction with DEMO fusion reactor design.
2. Integrated design was achieved, covering major critical issues.
3. Technology R&D's progressed in the area of the first wall, box and pebble bed structure fabrication. In addition to the past R&D achievement, critical issues of element technology for fabrication of a solid breeder blanket module has been clarified.
1.23 Development of Advanced Blanket Materials for Solid Breeder Blanket of Fusion Reactor


1) Oarai Research Establishment, Japan Atomic Energy Research Institute, 3607, Narita-cho, Oarai-machi, Higashiibaraki-gun, Ibaraki-ken, 311-1394 Japan
2) Naka Fusion Research Establishment, Japan Atomic Energy Research Institute, 801-1, Naka-machi, Naka-gun, Ibaraki-ken, 311-0193 Japan
3) Tokai Research Establishment, Japan Atomic Energy Research Institute, 2-4, Tokai-mura, Naka-gun, Ibaraki-ken, 319-1195 Japan
4) Center for Advanced Research of Energy Technology, Hokkaido University, 13-8, Kita-ku, Sapporo-shi, Hokkaido, 060-8628 Japan
5) Department of Quantum Engineering and systems Science, University of Tokyo, 7-3-1, Hongo, Bunkyo-ku, Tokyo, 113-8656 Japan
6) Research Institute for Applied Mechanics, Kyushu University, 6-1, Kasugakoen, Kasuga-shi, Fukuoka-ken, 816-8580 Japan
7) Advanced Reactor Fuel Division, Nuclear Fuel Industries, Ltd., 3135-41, Tokai-mura, Naka-gun, Ibaraki-ken, 319-1196 Japan
8) Engineering Department, New Metal Division, NGK INSULATORS, LTD., 1, Maegata, Handa-shi, Aichi-ken, 475-0825 Japan

e-mail contact of main author: kawamura@oarai.jaeri.go.jp

Abstract. The design of advanced solid breeding blanket in the DEMO reactor requires the tritium breeder and neutron multiplier that can withstand the high temperature and high neutron fluence, and the development of such as advanced blanket materials has been carried out by the cooperation activities among JAERI, universities and industries in Japan. The Li$_2$TiO$_3$ pebble fabricated by wet process is a reference material as a tritium breeder, but the stability on high temperature has to be improved for application to DEMO blanket. As one of such the improved materials, TiO$_2$-doped Li$_2$TiO$_3$ pebbles were successfully fabricated and TiO$_2$-doped Li$_2$TiO$_3$ has been studied. For the advanced neutron multiplier, the beryllides that have high melting point and good chemical stability have been studied. Some characterization of Be$_2$Ti was conducted, and it became clear that Be$_2$Ti had lower swelling and tritium inventory than that of beryllium metal. The pebble fabrication study for Be$_2$Ti was also performed and Be$_2$Ti pebbles were successfully fabricated. From these activities, the bright prospect was obtained to realize the DEMO blanket by the application of TiO$_2$-doped Li$_2$TiO$_3$ and beryllides.

1. Introduction

The design of advanced fusion blanket has been studied to realize DEMO reactors in Japan. In the design under development, coolant temperature is more than 500°C, and it is required for the tritium breeder and neutron multiplier in the blanket to accommodate the high temperature and high neutron fluence. Therefore, the development of advanced blanket materials has been pursued to realize higher performances required. In this paper, the collaborative activities among JAERI, universities and industries in Japan are reported on the development of these advanced materials.

For the tritium breeder, lithium titanate (Li$_2$TiO$_3$) pebbles with a diameter of 0.3-2mm were chosen as a tentative reference material from viewpoints of tritium recovery, chemical stability and so on. Concerning a pebble fabrication method, a wet process was chosen from points of mass productivity, $^6$Li recycle, etc., and was tested. From the results, it became obvious that a specification target (density: 80-85%T.D., grain size: <5μm) of tritium breeder was reached by wet process. In order to estimate the tritium release behavior at the lower temperature
which has large effects on fusion blanket design, in-situ tritium recovery experiments with \( \text{Li}_2\text{TiO}_3 \) pebbles fabricated by wet process were carried out at the Japan Materials Testing Reactor (JMTR) [1] and it became obvious that \( \text{Li}_2\text{TiO}_3 \) was a good material. However, it was clear that the stability at high temperature was bad. Therefore, the development of \( \text{TiO}_2 \)-doped \( \text{Li}_2\text{TiO}_3 \) is started as improved tritium breeder. In this study, the characteristics of this material are compared to that of \( \text{Li}_2\text{TiO}_3 \) and the application on the condition of DEMO blanket is also estimated.

For the neutron multiplier, beryllium metal (Be) is a reference material in the blanket design. However, it may not be applicable to the DEMO blanket that requires high temperature (\(-900^\circ \text{C}\)) and neutron dose (\(-20,000\) appmHe, \(-50\text{dpa}\)), because of high reactivity and large swelling. Therefore, it is necessary to develop the advanced material for a neutron multiplier that has high temperature resistance and high radiation resistance. Beryllides such as \( \text{Be}_{12}\text{Ti} \) and \( \text{Be}_{12}\text{V} \) have been expected as promising candidates for advanced neutron multipliers from the viewpoints of high melting point, high beryllium content, low radio activation, good chemical stability, etc. Several characterizations have been studied for \( \text{Be}_{12}\text{Ti} \) to evaluate the advantage. And the pebble fabrication study of \( \text{Be}_{12}\text{Ti} \) and \( \text{Be}_{12}\text{V} \) was carried out.

2. Development of improved tritium breeder
2.1 Characterization of \( \text{TiO}_2 \)-doped \( \text{Li}_2\text{TiO}_3 \)

The \( \text{TiO}_2 \)-doped \( \text{Li}_2\text{TiO}_3 \) in which \( \text{TiO}_2 \) contents are different, was examined in order to make clear \( \text{TiO}_2 \) doping effect. As the one of un-irradiated material properties, the effect of \( \text{TiO}_2 \) content on thermal properties of \( \text{Li}_2\text{TiO}_3 \) was measured and optimum \( \text{TiO}_2 \) content in \( \text{Li}_2\text{TiO}_3 \) was evaluated in this study. First, specific heat and thermal diffusivity of \( \text{TiO}_2 \)-doped \( \text{Li}_2\text{TiO}_3 \) were measured by differential scanning calorimeter method and laser flush method, respectively.

The results of specific heat measurement showed that specific heat of \( \text{TiO}_2 \)-doped \( \text{Li}_2\text{TiO}_3 \) (\( \text{TiO}_2 \) content: 2.5-20mol%) coincided with that of undoped \( \text{Li}_2\text{TiO}_3 \) within the range of 10%. Thermal diffusivity of \( \text{TiO}_2 \)-doped \( \text{Li}_2\text{TiO}_3 \) is shown in FIG. 1. The difference of thermal diffusivity between \( \text{TiO}_2 \)-doped \( \text{Li}_2\text{TiO}_3 \) and undoped \( \text{Li}_2\text{TiO}_3 \) was up to 15%. From these results, it was estimated that the effective thermal conductivity of pebble bed using \( \text{TiO}_2 \)-doped \( \text{Li}_2\text{TiO}_3 \) pebbles was almost the same as that using undoped \( \text{Li}_2\text{TiO}_3 \) pebbles. And, the results of specific heat measurement showed that the performance in phase transformation of \( \text{TiO}_2 \)-doped \( \text{Li}_2\text{TiO}_3 \) (\( \text{TiO}_2 \) content: 14mol%) on around 960°C at which the phase of \( \text{Li}_2\text{TiO}_3 \) changed to unstable state phase (\( \beta + \gamma \) phases) was almost the same as that of \( \text{Li}_2\text{TiO}_3 \). Additionally, the preliminary test for the estimation of irradiation damage of \( \text{TiO}_2 \)-doped \( \text{Li}_2\text{TiO}_3 \) was performed by means of the triple ion beams (0.25MeV H\(^+\), 0.6MeV He\(^+\) and 2.4MeV O\(^{2+}\)) [2] and it was also confirmed that the estimation on transformation mechanism of \( \text{TiO}_2 \)-doped \( \text{Li}_2\text{TiO}_3 \) was possible. From these results, it was considered that the optimum \( \text{TiO}_2 \) content in \( \text{Li}_2\text{TiO}_3 \) was less than 14mol%.

2.2 Fabrication technology of \( \text{TiO}_2 \)-doped \( \text{Li}_2\text{TiO}_3 \) pebbles

The fabrication tests of undoped and \( \text{TiO}_2 \)-doped \( \text{Li}_2\text{TiO}_3 \) pebbles (\( \text{TiO}_2 \) content: 2.5, 5 and 10mol%) were examined by two kinds of wet processes [3] (wet process with dehydration reaction and wet process with substitution reaction) and characteristics of pebbles were
evaluated. Sintered temperature decreased about 100°C when TiO₂ content in the TiO₂-doped Li₂TiO₃ pebbles was 5mol%. Relationship between grain size and sintered density of undoped and TiO₂-doped Li₂TiO₃ pebbles fabricated by the wet process is shown in FIG. 2. Especially, it was obvious that the grain size of the pebbles with 85% T.D. was less than 5μm on 5mol% TiO₂ doped in Li₂TiO₃ pebbles. No grain growth occurred in 5mol% TiO₂-doped Li₂TiO₃ at the annealing temperature of 900°C and 1000°C for 20 min. by the annealing test. The collapse strength of 5mol% TiO₂-doped Li₂TiO₃ pebbles was 1.5 times as large as that of undoped Li₂TiO₃ pebbles. From these results, bright prospect was obtained concerning the fabrication of the TiO₂-doped Li₂TiO₃ pebbles of diameter 0.2-2mm by the wet process.

3. Development of advanced neutron multiplier
3.1 Characterization of Be₁₂Ti
3.1.1 Compatibility
The compatibility test of Be₁₂Ti was carried out with structural material (SS316LN) and tritium breeder (Li₂TiO₃) at 600°C, 700°C and 800°C up to 1000h by annealing [4,5]. The results of the compatibility for SS316LN are shown in FIG. 3. It was obvious that the compatibility between Be₁₂Ti and SS316LN was much better than that of between Be and SS316LN. The thickness of reaction layer between Be₁₂Ti and SS316LN at 800°C was one tenth of that for Be. As to the compatibility between Be₁₂Ti and Li₂TiO₃, the reaction products on the Be₁₂Ti and Be in contact with Li₂TiO₃ were not found at any temperatures up to 1000h. On the other hand, the diffused Li into Be was identified at 800°C for 300h and 1000h. The results of these compatibility evaluations showed that Be₁₂Ti had the advantages for high temperature.

3.1.2 Swelling property
Be₁₂Ti and Be specimen were irradiated up to ~4x10²³ n/m² (E>1MeV) at 500°C in JMTR [6]. Helium production rate and dpa for Be were about 70appmHe and 0.5dpa, respectively. Swelling was calculated from the dimension and weight measurement results for the neutron irradiated Be₁₂Ti disk heated at 1100°C for 1h after irradiation. The swelling value of Be₁₂Ti was less than 3%. On the other hand, the swelling value of Be was ~60%. From these results, swelling of Be₁₂Ti under the high temperature neutron irradiation can be expected smaller than that of Be.

FIG. 2. Relationship between grain size and sintered density of Li₂TiO₃ and TiO₂-doped Li₂TiO₃ pebbles.

FIG. 3. Results of compatibility test.

FIG. 4. Desorption rate of deuterium.
3.1.3 Tritium inventory
The desorption property of deuterium was evaluated by the heating test after deuterium implantation [7]. A part of results is shown in FIG. 4. Deuterium was implanted up to $1 \times 10^{21}$ ions/m$^2$ at room temperature. The profile of desorption rate for Be$_{12}$Ti has a peak at about 100°C. On the other hand, the peak temperature of desorption rate for Be is higher (350°C-700°C) than that for Be$_{12}$Ti. The amount of 20% in implanted deuterium is retained in Be around 700°C. These results made clear that the deuterium desorption property of Be$_{12}$Ti was more superior than that of Be. It is obvious that the tritium inventory from Be$_{12}$Ti is much smaller than that for Be.

3.1.4 Evaluation of TBR
The evaluation of Tritium Breeding Ratio (TBR) using beryllide as a neutron multiplier was carried out using two models that were mono material packing and mixed material packing (tritium breeder and neutron multiplier) [8]. The tritium breeder was Li$_2$TiO$_3$ of 85%T.D. and 50at% $^6$Li enrichment. The packing fraction of pebble beds was 80%P.F. DOT3.5 code and FUSION-40 (based on JENDL3.2) were used for the calculation. The neutron wall load was 5MW/m$^2$. Assumed temperature in the blanket was as same as current blanket design. The result of the TBR evaluation is shown in FIG. 5. TBR of blanket with Be$_{12}$Ti pebbles was only 10% smaller than that with Be pebbles. It is considered that this value is within design window and the improvement by raising temperature is expected. It is also made clear that mixed pebble bed of tritium breeder and neutron multiplier would improve TBR 10% better.

3.1.5 Ion implantation
High He irradiation effects were preliminary evaluated by in-situ experiment using Multi Beam High Voltage Electron Microscope (MBHIVEM) [9]. Helium ion and electron were irradiated at the same time. The fluxes of He ion and electron were $3.87 \times 10^{16}$ ions/m$^2$ and $2.86 \times 10^{15}$ electrons/m$^2$, respectively. After the irradiation at room temperature, tiny bubbles and black dots were observed in pure Be specimens, however it was hard to observe the such irradiation defects in Be$_{12}$Ti as shown in FIG. 6. Be$_{12}$Ti had less irradiation defects formation than that of Be.

3.2 Fabrication technology of beryllides pebbles
After several studies, it became clear that the fabrication process and the chemical composition of the beryllides were critical for the brittleness and some beryllium contents gave better ductility [10]. Electrodes with some Be contents were sufficiently

FIG. 5. The results of TBR evaluation

FIG. 6. Photographs of Be and Be$_{12}$Ti after irradiation

FIG. 7. Photographs of electrode and pebbles
ductile for the rotating electrode method and some pebbles were obtained (see FIG. 7).

4. Conclusion
The results of advanced blanket material development are as follows:
As to tritium breeder,
- Phase transformation at around 960°C was not observed up to 10mol% TiO₂ doping.
- The effective thermal conductivity of TiO₂-doped Li₂TiO₃ pebble bed was within the design window.
- 5mol%TiO₂-doped Li₂TiO₃ pebbles with the target values were successfully fabricated by indirect wet process.
As to neutron multiplier,
- Compatibility of beryllide is smaller than that of Be.
- Swelling of beryllide is smaller than that of Be.
- Tritium inventory of beryllide is lower than that of Be.
- TBR using beryllide as a neutron multiplier is ensured the enough value.
- Irradiation defects of beryllide at high fluence irradiation are fewer than that of Be.
- Some prospects for ductility improvement of beryllide electrode were obtained.
From these activities, the advanced blanket materials with high temperature resistance have been successfully developed, and bright prospect was obtained to realize the DEMO blanket by the application of TiO₂-doped Li₂TiO₃ and beryllides.

Reference
Introduction (Development of Blanket Materials)

The development of improved tritium breeder

TiO₂-doped Li₂TiO₃ pebble has been developed as improved tritium breeder in order to satisfy the target values.

(Target values: density: 80–85 %T.D., grain size: -5μm)

The development of advanced neutron multiplier

Beryllides have been studied as alternative of beryllium metal for DEMO blanket.

(DEMO blanket condition: -900 °C, -20000 appmHe and -50 dpa)

Characterization and pebble fabrication of tritium breeder and neutron multiplier were conducted.

Characterization of TiO₂-doped Li₂TiO₃ (1)

Specific heat

Performance of phase transformation

The following reaction occurred in TiO₂-doped Li₂TiO₃ by sintering process in the fabrication:

Specific heat of TiO₂-doped Li₂TiO₃ (25–250°C) associated with that of un-doped Li₂TiO₃ within the range of 10%.

Characterization of TiO₂-doped Li₂TiO₃ (2)

Thermal diffusivity

Temperature effect of irradiation damage

The thermal diffusivity was decreased with increasing TiO₂ doping.

The content of Li₂TiO₃ was increased with increasing TiO₂ doping. (The thermal diffusivity of Li₂TiO₃ was smaller than that of Li₂TiO₃.)

The difference of thermal diffusivity between TiO₂-doped Li₂TiO₃ and un-doped Li₂TiO₃ was up to 10%.

The change in the temperature effect on the formation of anatase on the surface of Li₂TiO₃; the higher is temperature, the more efficient is the formation of anatase phase.

The preliminary test for the estimation of irradiation damage using un-doped Li₂TiO₃ confirmed that the estimation on damage mechanism of TiO₂-doped Li₂TiO₃ was possible.
Fabrication Technology of TiO₂-doped Li₂TiO₃ Pebbles

Wet process flow

- Mixing
- Drying
- Calcination/sintering
- Pelletizing

Effect on sintering of TiO₂ doping

The grain size of pellets with 85%T.D. was greater than 5μm on 5mol% TiO₂-doped in Li₂TiO₃ pellets by wet process.

Photographs of TiO₂-doped Li₂TiO₃ pellets

Compatibility Properties & Tritium Inventory Evaluation

- Compatibility with SS916LN
- Deuterium desorption property
  - Implantation condition: D⁺ ion fluence (50keV): 1x10¹²ions/cm²
  - Irr. Temp: R.T.
  - Holding time after Irr.: 2h
- Heating rate: 1 K/s

- Compatibility with Li₂TiO₃
  - Heating line (860°C)

Swelling Property & Behaviour of Irradiation Defects

- Swelling property
  - Irradiation condition in JMTR: 4x10⁶n/cm²
  - Irradiation temperature: 550°C

- Ion implantation
  - Irr. temp: R.T.
  - Acc. Vol: 50keV
  - Dose: 1x10¹⁰/cm²

The swelling value of Be-Ti was less than 3%. On the other hand, the swelling value of Be was ~60%.

Small bubbles and black dots were observed in pure Be specimen, however it was hard to observe the such irradiation defects in Be-Ti.

Development of Pebble Fabrication (in the case of Be-Ti)

- Survey test of electrode fabrication
  - Small scale fabrication test by REM (REM: Rotating Electrode Method)
  - Sample: 5 at.%, 10 at.%, 15 at.%(BeTie)
  - Heating rate: 1 K/s

It becomes clear that the fabrication process and the chemical composition of the beytitises were critical for the brittleness and some beytium contents gave better ductility.
Conclusion

**Tritium breeder**
- Phase transformation at around 960°C was not observed up to 10mol% TiO₂ doping.
- The effective thermal conductivity of TiO₂-doped Li₂TiO₃ pebble bed was within the design window.
- 5mol% TiO₂-doped Li₂TiO₃ pebbles with the target values were successfully fabricated by in-direct wet process.

**Neutron multiplier**
- Compatibility of beryllide is smaller than that of Be.
- Swelling of beryllide is smaller than that of Be.
- Tritium inventory of beryllide is lower than that of Be.
- TBR using beryllide as a neutron multiplier is ensured the enough value.
- Irradiation defects of beryllide at high fluence irradiation are fewer than that of Be.
- Some prospects for ductility improvement of beryllide electrode were obtained.

Bright prospect was obtained to realize the DEMO blanket by the application of TiO₂-doped Li₂TiO₃ and beryllides.
1.24 Neutronics Experiments for DEMO Blanket at JAERI/FNS


Japan Atomic Energy Research Institute, Tokai-mura, Naka-gun, Ibaraki-ken, 311-1195 Japan

e-mail: sato@naka.jaeri.go.jp

Abstract. In order to verify the accuracy of the tritium production rate (TPR), neutron irradiation experiments have been performed with a mockup relevant to the fusion DEMO blanket consisting of F82H blocks, Li2TiO3 blocks with a 6Li enrichment of 40 and 95 %, and beryllium blocks. Sample pellets of Li2TiO3 were irradiated and the TPR was measured by a liquid scintillation counter. The TPR was also calculated using the Monte Carlo code MCNP-4B with the nuclear data library JENDL-3.2 and ENDF-B/VI. The results agreed with experimental values within the statistical error (10 %) of the experiment. Accordingly, it was clarified that the TPR could be evaluated within 10 % uncertainty by the calculation code and the nuclear data. In order to estimate the induced activity caused by sequential reactions in cooling water pipes in the DEMO blanket, neutron irradiation experiments have been performed using test specimens simulating the pipes. Sample metals of Fe, W, Ti, Pb, Cu, V and reduced activation ferritic steel F82H were irradiated as typical fusion materials. The effective cross-sections for incident neutron flux to calculate the radioactive nuclei ($^{56}$Co, $^{186}$Re, $^{48}$V, $^{206}$Bi, $^{65}$Zn and $^{51}$Cr) due to sequential reactions were measured. From the experimental results, it was found that the effective cross-sections remarkably increases with coming closer to polyethylene board that was a substitute of water. As a result of the present study, it has become clear that the sequential reaction rates are important factors to accurately evaluate the induced activity in fusion reactors design.

1. Introduction

In the nuclear fusion DEMO (demonstration) reactor, the blanket is required to provide a tritium breeding ratio larger than unity by neutron-induced reactions in lithium in the blanket. Solid breeder blankets being developed by JAERI for tokamak-type DEMO reactors utilize Li2O or Li2TiO3 as tritium breeder material, beryllium as neutron multiplier, reduced activation ferritic steel F82H (2% W, 8% Cr and 90% Fe) as structural material and water as coolant [1,2]. Neutrons are captured by the structural material, water and the divertor, and escape to the outside through ducts. To ensure tritium breeding ratio larger than unity is a critical issue in the development of the blanket and the fusion reactor design. Also important is to develop a blanket with a low activation level. It was pointed out recently that the activation processes via not only primary neutron reactions but also sequential reactions should be considered in activation calculations [3,4]. In order to experimentally evaluate these issues, neutronics experiments have been performed by using DT neutrons at Fusion Neutron Source (FNS) [5] facility of JAERI.

2. Tritium production experiments

The $^6$Li-enriched Li2O or Li2TiO3 is proposed as prospective candidate of breeding materials for the DEMO reactor blanket. The tritium production rate for the blanket design, however, has never been evaluated experimentally with the D-T neutron source, and tritium production experiments with blanket assemblies using $^6$Li-enriched (40 and 95 %) Li2TiO3, Be and F82H, were carried out at JAERI-FNS to evaluate the tritium production rate (TPR).

Figure 1 shows a schematic view of the experimental assembly. As no reflector was set around the D-T source, incident neutrons on the left surface are monochromatic at 14 MeV. The tritium-titanium (TiT) target generates $1.7 \times 10^{11}$ D-T neutrons/s on average via $^3$H(d,n)$^4$He reaction. In order to measure the D-T neutron generation rates, the associated
α particles were monitored with Si surface barrier detectors. The test assembly was mounted on thin aluminum support frames and a steel deck. The distance between the surface of the assembly and the Ti target was 200 mm. The thickness of the first F82H, Li$_2$TiO$_3$, second F82H and beryllium layer were 16, 12, 3, and 200 mm, respectively. The lateral side of Li$_2$TiO$_3$ region was covered with 12 mm thick B$_4$C. The size of the assembly was about 500 x 500 mm$^2$ in area with a total thickness of 231 mm. The assembly was surrounded by 100-150 mm thick nat-Li$_2$CO$_3$ to shield the neutrons reflected by the experimental room walls. Square holes 50 x 50 mm$^2$ in area were provided for the fluence measurement through the centerline of the first and second F82H layers. Laminated F82H sample sheets 1.6 and 1 mm in thickness were filled in the center holes of the first and second F82H layers, respectively, to measure the activation distribution in the F82H. Pellets of Li$_2$TiO$_3$, 12 mm in diameter and 2 mm in thickness, were located in a bore in the $^6$Li$_2$TiO$_3$ layer as detectors to measure the TPR distribution. A liquid scintillation counter system (Aloka-5500) was employed to measure the β-radiation from the yielded tritium [6], and a high purity Ge detector was used to measure the gamma ray emitted from the activated foils [7].

Monte Carlo transport codes MCNP-4B and -4C were used to calculate the TPR and foil activations with JENDL-3.2 and ENDF-BVI nuclear data libraries. Figure 2 shows the TPR for Li$_2$TiO$_3$ pellets obtained by the experiment and calculations. The pellet nearest to the beryllium zone shows the highest TPR. For all the other pellets, TPRs are nearly constant except for a little larger value of the pellet adjacent to the first F82H layer. Although plenty of thermal neutrons are produced by the beryllium zone, a fairly large ratio of them must be captured in the Li$_2$TiO$_3$ pellet nearest to the beryllium zone because of the large thermal cross section of $^6$Li(n,α) reaction. On the other hand, TPRs in the medium locations are expected to be created mainly by $^6$Li(n,α) reactions of giant resonance cross section at 240 keV. Thus, the important matter for the reliable calculation seems to be how excellently the reaction rate is reproduced around the resonance. In Fig. 2 the calculated TPRs agreed well with the experimental values within the experimental error of about 10 %, which means TPR can be calculated by the Monte Carlo method within the uncertainty of 10 %. As for the nuclear data, significant difference was not observed between JENDL-3.2 and ENDF/B-VI.
3. Sequential Reaction Experiments

Recently it was pointed out that the activations via sequential charged particle reactions (SCPRs) defined as the reactions induced by secondary charged particles should be considered in activation calculations for safety designs of future D-T fusion reactors [3,4]. As for the SCPRs induced by charged particles emitted from primary neutron reactions in homogeneous materials, some experimental results have been reported by Ikeda et al. [8] and Maekawa et al. [9]. However, a special consideration of the SCPRs will be required in the boundary region between different materials. Around the surface of a cooling water pipe, it is expected that the radioactivity production via SCPRs would be enhanced by recoiled proton from hydrogen in the water. The enhancement of the radioactivity production makes corrosion products more activated. It gives rise to critical issues because the corrosion products may be transported along coolant loops into regions outside the biological shield. Thus, the experimental studies have been performed with test specimens simulating the cooling water pipe.

A typical arrangement of the test specimens is shown in Fig. 3. Each irradiation foil was laminated on a polyethylene board 3 mm in thickness that simulates water flowing inside a cooling pipe. The size of the foil was 10 x 10 x (0.05-0.25) mm$^3$. Natural Fe, W, Ti, Pb, Cu, V and low activation ferritic steel F82H were employed as samples. The radioactive nuclei generated from these materials by the irradiation of secondary protons have sufficiently long half-lives, and the energies of the emitted decay gamma rays are suitable for measurement. Neutrons bombarded the sample materials at the magnitude of $10^{10}$ order n/cm$^2$/s flux for 23-40 hours. In order to determine the neutron flux incident on each laminated sample, niobium foils of 10 x 10 x 0.1 mm$^3$ were attached on both sides of it. A $^{235}$Th fission chamber located at the ceiling of the target room was used as a monitor for generated neutrons. After a suitable cooling time, the gamma rays emitted from the irradiated foils were measured by a high purity Ge detector, and the peaks corresponding to the radioactive nuclei ($^{56}$Co, $^{184}$Re, $^{48}$V, $^{206}$Bi, $^{65}$Zn and $^{51}$Cr) produced by the SCPRs in Fe, W, Ti, Pb, Cu and V were identified in the measured spectra, respectively.
FIG 3  Typical arrangement of the irradiated sample.

FIG 4  The effective cross-sections for the $^{56}\text{Co}$ production in Fe foils with respect to the distances from the surface of the polyethylene board.

FIG 5  The gamma-ray spectrum of the F82H foil attached close to the polyethylene board measured after 2 months of cooling time.

The effective cross section was defined as the sequential reaction rate per target atom of natural abundance per neutron flux [10]. For tungsten and lead, the effective cross-sections associated with SCPRs have been measured for the first time. The sequential reaction rate averaged over the thickness of the foil was experimentally derived from the full-energy peak counts by using the full-energy peak efficiencies, well-known decay constant, and gamma ray emission probabilities. Figure 4 shows the derived effective cross-sections for the $^{56}\text{Co}$ production in Fe foils with respect to the distances from the surface of the polyethylene board. With coming closer to the board, an exponential increase of the effective cross-sections is clearly observed in the region between positions #1 and #6. The values around the distance range from 300 to 450 μm are rapidly dropping. This is because the flight range of 14-MeV protons is about 450 μm. During the flight, protons lose the energy down to 5.35 MeV, that is the Q value of the $^{56}\text{Fe}(p,n)^{56}\text{Co}$ reaction [11]. Figure 5 shows the gamma ray spectrum of the F82H foil attached close to the polyethylene board measured 2 months after the irradiation. The gamma rays corresponding to $^{56}\text{Co}$ due to the $^{56}\text{Fe}(p,n)^{56}\text{Co}$ sequential reaction were clearly observed. It is noteworthy that the intensity of the 847-keV gamma ray from $^{56}\text{Co}$ is about
10% as large as that of the 835-keV gamma ray from $^{54}$Mn due to $^{54}$Fe(n,p) reaction by primary neutrons. In the equilibrium, the activity of $^{56}$Co whose half-life is 77 days was calculated at about 3% of the activity of $^{54}$Mn (312 days, likewise) on the basis of the effective cross section obtained in this experiment. Because the contact dose rate factor per unit activity for $^{56}$Co is about 4 times larger than that for $^{54}$Mn, the contribution of $^{56}$Co to the spatial radiation dose rate must be about 10% of that of $^{54}$Mn. During the operation before the equilibrium is attained, the contribution must be larger than the value, so that it cannot be neglected in the safety design.

4. Summary

In order to verify the accuracy of the tritium production rate (TPR), neutron irradiation experiments have been performed with an assembly simulating the fusion DEMO blanket. The TPR was also calculated using the Monte Carlo code MCNP-4B and -4C with the nuclear data library JENDL-3.2 and ENDF-B/VI. It was clarified in the experiment that the TPR was drastically enhanced in the breeder regions adjacent to the beryllium blocks due to an increase of thermal neutrons in the beryllium blocks. The TPRs derived from the calculation results agreed with those from the experiment within 10%, which corresponds to the statistical error in the experiment. Further experiments are planned with Li$_2$TiO$_3$ in a form of small pebbles as proposed in the DEMO blanket, and the evaluation accuracy for the pebble configuration will be examined.

Undesirably long-lived radioactive nuclei are expected to be generated by sequential reactions induced by secondary charged particles. In the present study, neutron irradiation experiments have been performed for test specimens simulating cooling water pipes in the DEMO blanket. Sample metals of Fe, W, Ti, Pb, Cu, V and F82H were irradiated. The effective cross-sections for producing the radioactive nuclei ($^{56}$Co, $^{184}$Re, $^{48}$V, $^{206}$Bi, $^{65}$Zn and $^{51}$Cr) generated by the sequential reactions were measured. In the present study, it has been clarified that the sequential reaction rates are enhanced, by taking into account the recoiled protons from water, by more than 10 times compared with those induced by the charged particles emitted from the primary neutron reactions in the metals themselves. From the present study, it has been clarified that the sequential reaction rates are of great importance to evaluate the shut-down dose rates and the activated corrosion products.

References

Neutronics Experiments for DEMO Blanket at JAERI/FNS


Background and Objective

1) To provide a TBR of more than unity is a critical issue in the DEMO blanket. Tritium production experiments have been performed to validate the design calculation uncertainty for the multi-layered blanket assembly proposed by JAERI.

2) Evaluation of the induced activity with high accuracy is required. Sequential reactions experiments due to proton have been performed for the water pipe.

Tritium Production Experiment

Appearance of the Experiment Assembly

Neutron irradiation experiments have been performed with a mockup relevant to the Fusion DEMO blanket.

Results and Discussions

Irradiated L2TiO3 pellets were dissolved by concentrated HCl.

β-rays emitted from tritium in the solution were measured by liquid scintillation counter.

TPR was drastically enhanced in the breeder regions adjacent to the beryllium blocks due to an increase of thermal neutrons. Most of TPRs derived from the calculation results agree with those from the experiment within 10%, statistical error in the experiment (10%).
Sequential Reactions Experiment

Undesirable nuclei would be produced via the sequential reactions!

Charged particles (CPs) are generated by two ways: A: target material, B: another material

- A(n,CPs) → A(CPs,x) : previous study
- B(n,CPs) → B(CTp,x) : present study

Many recoiled protons are generated by irradiating water, and the induced activity due to the sequential reactions is enhanced. Neutron irradiation experiments were performed for test specimens simulating water pipes.

Results and Discussions

Measured Effective Cross Section as a Function of Distance from Polyethylene Board

The effective cross-sections due to the sequential reactions are increased in a form close to an exponential curve in all samples with reducing the distance to the polyethylene board. The sequential reaction rates are enhanced, by taking into account the recoiled protons from water, by more than 10 times.

Gamma ray spectrum from irradiated F82H

Gamma ray from \(^{56}\text{Co}\) due to \(^{56}\text{Fe}(p, n)\) is about 10% of that from \(^{56}\text{Mn}\) due to \(^{56}\text{Fe}(n, p)\).

Gamma ray dose rates from \(^{56}\text{Co}\) are 10 – 45% those from \(^{56}\text{Mn}\).

Conclusion

1) Tritium Production Experiment
   Calculation results agree with experimental results within experimental error (about 10%).

2) Sequential Reaction Experiment
   Activities due to water are one order of magnitude larger than those in metal themselves. Activities in F82H are experimentally measured (e.g. \(^{56}\text{Co}\) due to \(^{56}\text{Fe}(p, n)\)).
1.25 Tight Aspect Ratio Tokamak Power Reactor with Superconducting TF Coils

S. NISHIO\(^1\), K. TOBITA\(^1\), S. KONISHI\(^1\), T. ANDO\(^1\), S. HIROKI\(^1\),
T. KURODA\(^1\), M. YAMAUCHI\(^1\), M. AZUMI\(^1\), M. NAGATA\(^2\)

1) Naka Fusion Research Establishment, Japan Atomic Energy Research Institute, Naka-machi,
Naka-gun, Ibaraki-ken, 311-0193 Japan

2) Himeji Institute of Technology

e-mail contact of main author: nishio@naka.jaeri.go.jp

Abstract. Tight aspect ratio tokamak power reactor with superconducting toroidal field (TF) coils has been proposed. A center solenoid coil system and an inboard blanket were discarded. The key point was how to find the engineering design solution of the TF coil system with the high field and high current density. The coil system with the center post radius of less than 1 m can generate the maximum field of 20 T. This coil system causes a compact reactor concept, where the plasma major and minor radii of 3.75 m and 1.9 m, respectively and the fusion power of 1.8 GW.

1. INTRODUCTION

For the fusion power plant, a realization of a competitive cost of electricity (COE) is the first priority to be adopted by the utility companies. The COE must be low decided, but not less important the construction cost (so called capital cost) must be low. The COE’s lowering by a scale merit approach is not necessarily received favorably. Tight aspect ratio tokamaks with the aspect ratio A in the range 1.2 \(\sim\) 2.0 can offer the possibility of compact fusion reactors in the low capital cost way. This scheme is often called the spherical torus (ST) approach. In the usual ST approach, all non-essential components such as inboard blanket or shield, inboard poloidal coil (PF) systems like a center solenoid (CS) coil system are discarded from the inner side of the plasma. The only customary tokamak component that remains on this side is a single turn copper TF coil center post (CP). In spite of the excellent plasma performances granted by very low aspect ratio less than 1.5, it could not sufficiently compensate the Joule losses in the normal-conducting (NC) TF coil as illustrated in the ARIES-ST power reactor design study [1]. If a super-conducting (SC) TF coil system is used instead, approximately 1 meter of shielding is required between the SC TF coils and the plasma on the inboard side to protect the superconductors from neutron damage and nuclear heating. Consequently, it has been widely recognized that a super-conducting compact tokamak reactor with such the tight aspect ratio would not be feasible. Our recent study, however, opened up the possibility of realizing such a very compact reactor being compatible with the use of SC TF coil system. The key issue is to find the SC TF coil design solution. How high field and how high current density in the coil windings are required for compensating the handicap of the thick shielding?

In section 2, the design guidelines for the tight aspect ratio tokamak reactor are described. The design features of the SC TF coil system are briefly described in section 3. Possible plasma performances within the reliable engineering constraints are described in section 4. The recommendable reactor concept and the discussion for design deepening are described in section 5. The last section consists of design summary.

2. Design Guidelines

The following three guidelines are listed up for compatibility between the tight aspect ratio plasma and the SC TF Coil system.

i) Discard the center solenoid (CS) coil. Non-inductive current generation and sustaining methods such as RF wave or NBI are adopted. In the A-SSTR2 design study, the plasma break-down and current ramp-up scenarios were successfully established without CS coil system by 1.5D simulation code [2]. Furthermore the experimental demonstration by using JT-60 tokamak device was carried out [3].

ii) Discard the inboard blanket. Although the tritium breeding function is limited in the outboard blanket, the tritium breeding ratio requirement of more than 1.05 can be
attainable. Approximately 1 meter of the inboard shielding structure is taken into account.

iii) Increment the TF coil average current density. Not only high magnetic field strength but also high current density are required for the TF coil. This is an essential issue to compensate the "burden" of 1 m shield thickness.

3. TF Coil System

Because of discarding the CS coil system, the TF coil inner legs naturally become a solid-like integrated center post (CP) structure. For compensating ~1 m shield thickness, a slender i.e. high current density CP is indispensable. Fortunately, this TF coil configuration has the structural rigidity and the stored energy is relatively low for its field strength. The SC material is Bi2212/Ag/AgMgSb multi-filament with an operation temperature of 20 K, which is as same as the A-SSTR2 SC material [4]. A calculation method for the material composition optimization among the SC filament, stabilizer, structure (load-carrying) material and cooling channel area has been established. Namely, it is possible to know the maximum achievable field strength under the given CP radius. The design condition is same as the A-SSTR2 [2]. For instance, the SC filament operation current density is 500 A/mm² (critical density is 1,000 A/mm²), the maximum allowable field strength of the SC filament is 23 T, the coil terminal voltage is less than 20 kV, the temperature margin is 3~5 K and the design stress of structure material JJ1 [5] is $S_m = 800$ Mpa. The maximum field ($B_{max}$) dependence on the CP radius was evaluated by 3D FEM stress calculation. The result is shown in Fig.1.

4. Plasma Performance

An acceptable nuclear heating rate of 0.5 mW/cm² in the TF coil conductors corresponds to the distance of 95cm between plasma surface and the TFC as to the torus inboard region as shown in Fig.2. An SiC/SiC composite for the power core structure material and VH₁ for the bulk shield material are introduced. A coolant material is liquid lithium. The outboard radial build shown in Fig.2 satisfies the local TBR (tritium breeding ratio) of 1.3, where the liquid lithium is not only the coolant but also the tritium breeding material.

A plasma ellipticity is directly and deeply connected with the reactor concept. While a highly elliptic plasma is preferable for attaining a high beta plasma and for receiving a high plasma current, it becomes difficult to obtain plasma MHD equilibrium solutions and to control plasma vertical position. From the plasma equilibrium point of view, the high elliptic plasma requires the PF coil position near the equatorial plane of the tokamak machine. The well maintainability is a high priority matter in our reactor design. The power core components are designed for horizontal insertion and withdrawal of entire sectors. Therefore we have no PF coil near the equatorial plane. And the use of low activation material is another high priority matter. A vanadium alloy is adopted to the passive stabilizing shell structure. A electric resistivity of the vanadium alloy is 30 times

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Fig.1 Maximum field dependence on center post radius. In our study, the design stress of 800 Mpa is adopted.

Fig.2 Radial build based on neutronics (shield and TBR) calculation.
higher than a copper material. For the replaceable torus sector unit, the possible shell structure must be the so called "saddle loop" structure shown in Fig.3. The plasma ellipticity in our design has been decided to be 2.1. The normalized beta in our design is decided with consideration of certain design margin based on the maximum achievable values of Ref. [6]. The plasma performance calculation has been carried out under the physics constraints which are the energy confinement of $\frac{HH}{\eta}=1.8$ and the plasma density of $n/n_{GW}=1.0$. A self-consistent plasma parameter set is calculated for the each given CP radius value. Figure 4 shows the dependences of $R_p$ (plasma major radius), $a_p$ (plasma minor radius), $A$ (plasma aspect ratio), $P_f$ (fusion power), $P_n$ (average neutron wall load), $I_p$ (plasma current), $f_{BS}$ (bootstrap current fraction), $P_{NB}$ (neutral beam power required for plasma current drive) and $Q$ (energy multiplication factor) on the $R_{CP}$ (TF coil center post radius). The plasma average temperature is assumed to be 20 keV.

When the CP radius is in slender region thinner than 60 cm, the field strength on plasma axis is considerably low. This leads the plasma sizes (major and minor radii), the plasma current, and the required current drive power very large. It is not acceptable performance for the power reactor. On the other hand, in the thicker region than 1 m, despite the reduction tendencies of the plasma sizes and current are saturated, the TF coil construction cost must be high. And the high field leads the neutron wall load high. Maximum allowable wall load is limited less than 5 MW/m². The maximum wall load of 5 MW/m² corresponds to the average value of 3.5 MW/m². Therefore, the CP radius of 90 cm is chosen for our new reactor concept named VECTOR (VEry Compact TOKamak Reactor). Major specifications of VECTOR are listed in Table 1.

The plasma equilibria during the ramp-up phase are found within the reasonable ampere-turns less than 100 MA. In a non-inductive plasma initiation (breakdown) and current ramp-up scheme, the NBI (neutral beam injection) should be replaced by RFW (radio frequency wave) heating and current drive to avoid high shine through and orbital losses in the low density, low current (high

\begin{table}[h]
\centering
\begin{tabular}{|l|l|}
\hline
\textbf{Table 1 VECTOR Major Parameters} & \\
\hline
\textbf{Plasma Major Radius} & $R_p = 3.75$ m \\
\textbf{Plasma Minor Radius} & $a_p = 1.9$ m \\
\textbf{Plasma Ellipticity} & $\kappa = 3.75$ m \\
\textbf{Plasma Current} & $I_p = 18.3$ MA \\
\textbf{Normalized Beta} & $\beta_N = 3.75$ \\
\textbf{Fusion Power} & $P_F = 1.8$ GW \\
\textbf{Neutron Wall Load} & $P_n = 3.5$ MW/m² \\
\textbf{Maximum Field in TFC} & $B_{MAX} = 19.6$ T \\
\hline
\end{tabular}
\end{table}
q) plasma. Here, an EC (Electron Cyclotron) system is considered for heating and current drive.

5. VECTOR Mechanical Configuration

The replacement scheme of the power core component and the vacuum boundary layout are very simple in comparison with the conventional tokamak device. The power core components are designed for horizontal insertion and withdrawal of entire sectors. Since the structure material of the power core components is not a conducting material, the eddy currents are not induced on the components by the plasma disruption event. Therefore, the firm connection (it is an obstacle to the quick replacement) between the adjacent power core sectors and/or the assembling components is not needed. The torus segmentation and replacement scheme are strongly related to the vacuum boundary layout.

A high degree of vacuum integrity is required for both the plasma chamber and the cryostat for the SC coil system. In the VECTOR configuration, a single-walled containment of the plasma chamber constitutes also the continuous toroidal vacuum-tight containment of the cryostat as schematically shown in Fig.5. The bell-jar envelopes all the TF and PF coils. Its outer cylindrical part, provided with certain number of windows for the withdrawal of torus sectors. The bird’s-eye view of VECTOR is shown in Fig.6.

Fig.5 Schematic drawing of vacuum boundary and torus segmentation for VECTOR. The vacuum seal between adjacent torus sectors is needless.

6. Summary

i) Not only low COE (cost of electricity) but also low capital cost (construction cost) are required for the fusion power plant.

ii) A tight aspect ratio tokamak reactor is promising concept answering for the above requirements. But the concept with normal conducting TF coil does not answer for the COE requirement [1].
iii) A key point of the attractive tokamak with a tight aspect ratio is to find the engineering design solution for the super-conducting TF coil with a high field and high current density.

iv) The TF coil system with the center post radius of less than 1 m can generate the maximum field strength of ~20 T.

v) Such the TF coil system mentioned above causes a compact reactor concept, where the plasma major and minor radii of 3.75 m and 1.9 m, respectively and the fusion power of 1.8 GW.

Acknowledgments

The authors are indebted to T. Takizuka, Y. Takase, and Y. Ono for information, suggestion, and discussion. The authors would like to acknowledge the beautiful CAD drawings provided by J. Maeno and the many graphs prepared by T. Nishino.

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**Tight Aspect Ratio Tokamak Power Reactor with Superconducting TF Cells**

**S. HISHIO**

for VESTOR Design Team

JAERI: Naka Fusion Research Establishment

Kaga-machi, Naka-gun Ibaraki-ken 311-0193, JAPAN

Hishio@naoh.jaeri.go.jp

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**Lower Capital (Construction) Cost**

**Contents**

- ST Plasma, Its Higher Performance
- Compatible Reactor Structure with ST Plasma
- Toroidal Field Coil System
- Compact & High Performance Power Reactor

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**ST Plasma, Its excellent performance.**

*Beta, Ellipticity, Bootstrap current*

Even for BS current fraction, ST plasma is **superior**!
Design Flow of SC-ST TF Cell

Optimization of TFC Material Composition

- Design Current
- Current Density of Filament
- Superconducting Filament Area
- TFC Aspects
- TFC Shape
- Shield Energy
- Terminal Voltage of TFC System
- Conductor Current
- Allowable Current Decay Rate
- Joule Heating
- Structural Material
- Structural Material Area
- Cooling Channel Area
- Insulator Area
- Plasma Design
- Neutronics Design

Performance of SC-ST TF Cell System

Performance of SC-ST TF Cell System

- Maximum Field, Bmax (T)
- Center Post Radius, Rcp (m)
- Average Current Density (A/m²)

Thin or thick, which is more attractive?

Shield and TBR

- Criterion
  - Nuclear Heating (<5mW/cc)
  - Local TBR (>1.3)

- Material Choice
  - No Electromagnetic Force
  - Low Activation

- Tritium breeding is only in outboard.
- Tungsten located inboard is for shielding and neutron reflection.

Plasma Ellipticity Guideline (1)

Plasma Ellipticity & PFC Position

- High K (2.0)
- Low K (2.0)

A priority matter: Maintainability
Plasma Ellipticity, K = 2.1
Plasma Ellipticity Guideline (2)

Shell Material: V-alloy
(\(\eta = 4.8 \times 10^{-3}\) cm, 30 times higher than copper)

A priority matter: Low Activation

Feedback Control ??

Feedback Plasma Position Control

Plasma positional balance is broken by some kind of small disturbance. When plasma displacement reach to 1 cm, the feedback control starts.

Control Voltage of 15 V is Acceptable!
Plasma Equilibrium

<table>
<thead>
<tr>
<th>Coil No.</th>
<th>Current</th>
</tr>
</thead>
<tbody>
<tr>
<td>Plasma</td>
<td>10.3 MA</td>
</tr>
<tr>
<td>1</td>
<td>3.3 MAT</td>
</tr>
<tr>
<td>2</td>
<td>1.7 MAT</td>
</tr>
<tr>
<td>3</td>
<td>5.0 MAT</td>
</tr>
<tr>
<td>4</td>
<td>-14.2 MAT</td>
</tr>
<tr>
<td>5</td>
<td>-21.7 MAT</td>
</tr>
<tr>
<td>6</td>
<td>7.6 MAT</td>
</tr>
<tr>
<td>7</td>
<td>15.0 MAT</td>
</tr>
<tr>
<td>8</td>
<td>10.0 MAT</td>
</tr>
</tbody>
</table>

Ψ_{Supply} = 45 Vs

VECTOR Mechanical Configuration

Plasma Major Radius: \( r_m = 3.75 \) m  
Plasma Minor Radius: \( a = 1.9 \) m  
Plasma Ellipticity: \( \beta = 2.1 \)  
Plasma Current: \( I = 18.3 \) MA  
Normalized Beta: \( \beta_n = 5.6 \)  
Fusion Power: \( P_F = 1.8 \) GW  
Neutron Wall Load: \( P_B = 3.2 \) MW/m²  
Maximum Field: \( B_{max} = 19.6 \) T  
Field on axis: \( B_0 = 4.7 \) T  
Toroidal Beta: \( \beta_t = 11.5 \% \)

Vacuum Boundary

Reactor Comparison

Both Lower capital cost & COE can be achieved by VECTOR-type (SC & ST) reactor.

C/E Surmise

Scaling Law of COE (by K. Okano)

\[
\text{COE}_{\text{C}} = \frac{11.17}{P_B^{0.5} \rho_0^{0.53} f_{\text{norm}}^{0.5}}
\]

for standard tokamaks (\( A = 3-4 \))

Lower COE also can be achieved by VECTOR-type (SC & ST) reactor.
(i) Not only low COE but also low Capital Cost are required for fusion power plant.
(ii) ST tokamak reactor is promising concept. But NC TFC doesn't (didn't) answer for.
(iii) Key of SC TFC concept is high field & slender. (Thick Shield is unavoidable for SC.)
(iv) The ST type SC TFC design method has been established.
(v) Tight aspect ratio tokamak power reactor with SC TFC, VECTOR has been proposed.
(vi) $A=2$, $R_p = 3.75 \text{ m}$, $a_p = 1.9 \text{ m}$ and $P_T = 1.8 \text{ GW}$. 
1.26 Advanced Fusion Technologies Developed for JT-60 Superconducting Tokamak


Japan Atomic Energy Research Institute, Naka Fusion Research Establishment
Naka-machi, Naka-gun, Ibaraki-ken, 311-0193 JAPAN

E-mail: sakasai@naka.jaeri.go.jp

Abstract. The modification of JT-60 is planned as a full superconducting tokamak (JT-60SC). The objectives of the JT-60SC program are to establish scientific and technological bases for the steady-state operation of high performance plasmas and utilization of reduced-activation materials in economically and environmentally attractive DEMO reactor. Advanced fusion technologies relevant to DEMO reactor have been developed in the superconducting magnet technology and plasma facing components for the design of JT-60SC. To achieve a high current density in a superconducting strand, Nb3Al strands with a high copper ratio of 4 have been newly developed for the toroidal field coils (TFC) of JT-60SC. The R&D to demonstrate applicability of Nb3Al conductor to the TFC by a react-and-wind technique have been carried out using a full-size Nb3Al conductor. A full-size NbTi conductor with low AC loss using Ni-coated strands has been successfully developed. A forced cooling divertor component with high heat transfer using screw tubes has been developed for the first time. The heat removal performance of the CFC target was successfully demonstrated on the electron beam irradiation stand.

1. Introduction

The modification of JT-60 is planned as a full superconducting tokamak (JT-60SC) [1, 2]. The objectives of the JT-60SC program are to establish scientific and technological bases for the steady-state operation of high performance plasmas and utilization of reduced-activation materials in economically and environmentally attractive DEMO reactor [3]. The plasma current of JT-60SC is $I_p = 4$ MA, the toroidal field is $B_t = 3.8$ T, the major radius is $R_p = 2.8$ m and the minor radius is $a_p = 0.85$ m (the elongation $\kappa_5 \sim 1.8$, the triangularity $\delta_5 \sim 0.35$). Comparison of the cross-sectional views between present JT-60U and designed JT-60SC is shown in Fig. 1 along with the present perpendicular NBI. In JT-60SC, primary heating and current drive are conducted using NBI system consisting of 2 N-NBI units, 8 perpendicular and 4 tangential P-NBI units and ECRF system. The superconducting coil system is composed of toroidal field (TF) coils, central solenoid (CS) and equilibrium field (EF) coils. A magnet stored energy of the TF coils for JT-60SC is estimated to be 1.7 GJ, which is the largest value in comparison with main coils constructed so far and in construction now.

Advanced fusion technologies relevant to DEMO reactor have been developed in the superconducting magnet technology and plasma facing components for the design of JT-

Fig. 1. Cross-sectional views of the present JT-60U and the designed JT-60SC.
60SC. A Nb$_3$Al strand of an 11 km length with a high copper ratio of 4.1 has been newly manufactured for the TFC of JT-60SC. The R&D to demonstrate applicability of Nb$_3$Al conductor to the TFC by a react-and-wind technique have been carried. A full-size NbTi conductor with low AC loss and reduced cost required for the EF coils has been successfully developed. A forced cooling divertor component with high heat transfer using screw tubes has been developed and tested by the electron beam irradiation.

2. Development of Nb$_3$Al Superconducting Magnet

2.1 Design of Toroidal Field Coils for JT-60SC

The TF coil system of the JT-60SC has been designed to consist of 18 "D" shape coils, which have a height of 6.0 m and a width of 3.9 m. The maximum magnetic field ($B_{\text{max}}$) in the windings is 7.4 T at an operational current ($I_{\text{op}}$) of 19.4 kA. Table I shows the main parameters of the TF coils [4].

The Nb$_3$Al conductor is a promising superconductor for the magnets of JT-60SC, because of its low strain sensitivity on superconducting performances. A stainless steel (SS316LN) conduit is adopted for JT-60SC, because Incoloy 908 requires careful control of the oxygen concentration during heat treatment. The superconducting strand of Nb$_3$Al or Nb$_3$Sn with relatively high critical current density ($J_c$) is required for JT-60SC. The current density of the cable-in-conduit (CIC) conductor for the TFC of JT-60SC ($B_{\text{max}} = 7.4$ T) is relatively higher than that for the TFC of ITER ($B_{\text{max}} = 11.8$ T). Therefore, a superconducting strand with high Cu/non-Cu ratio is required to attain a highly stable coil. The allowable Cu/non-Cu ratio of the strand for the TFC of JT-60SC was estimated to be around 4. The Cu/non-Cu ratio for a Nb$_3$Al Insert Coil under the framework ITER-EDA was 1.5. Nb$_3$Al strands with a copper ratio of > 2 have not been developed yet. The development of the Nb$_3$Al strand with a high $J_c$ gives a compact magnet design and substantial cost saving because of the reduced amount of the superconducting materials.

In addition, the Nb$_3$Al conductor allows us to fabricate the TFC by a react-and-wind technique. The TFC of JT-60SC was designed to make it possible to be fabricated by the react-and-wind technique for superconducting magnet technology of the large-scale magnet fabrication such as the TFC of SSTR [5].

The maximum bending strain of the TFC conductor for JT-60SC becomes 0.4%, which does not make large decrease of $J_c$. The bending strain $\varepsilon$ is defined as follows,

$$\varepsilon = \frac{D}{2R},$$

where $D$ is the diameter of the cable, $R$ is the curvature radius of bending. The TF coil has consists of 7 double pancakes with 154 turns. The conductor is designed to be a squired CIC conductor installed a cable, which consists of 216 Nb$_3$Al strands and 108 pure copper wires into a squired stainless steel conduit [6].

---

Fig. 2. Views of Nb$_3$Al strand and full-size conductor developed for the TFC of JT-60SC.
2.2 Development of 30 m length Nb₃Al Full-size Superconductor

The development of a Nb₃Al strand with a high copper ratio of 4.1, which was the optimizing value to the magnetic field of 7.4 T for the TF coil, was carried out. A Nb₃Al strand of an 11 km length in total was successfully drew down to 0.74 mm diameter (filament diameter of 47 μm) without breakage by increasing the quality of jelly roll structure (Fig. 2). A procedure for finishing a long piece of long Nb₃Al strand with a high copper ratio of 4.1 has been established [7].

This strand was coated with 2 μm Cr and cabled together with a pure copper wire. The superconducting cable inserted into a 30 m length conduit with two butt-welded joints, which was composed of three unit conduits of a 10 m length. The CIC conductor has the same configuration as the designed full-size TFC conductor. Figure 3 shows a view of the 30 m length Nb₃Al full-size conductor in process of the cable insertion. The non-Cu Jₐ of developed Nb₃Al strands was 1914 A/mm² at 7.4 T, 4.2K from the Jₐ measurement. Because of the thermal contraction effect from the SS conduit, the measured critical current (Iₐ) of the conductor was decreased to 98% of Iₐ estimated from the product of the Iₐ in strands by the number of strands. In the case of the Nb₃Sn CIC conductor, the decrease of Iₐ was estimated to be 79% [6]. This result indicated the low strain sensitivity of Nb₃Al conductor in comparison with the Nb₃Sn conductor.

2.3 R&D for Demonstration of a React-and-wind Technique

In the design of JT-60SC, the react-and-wind technique with the Nb₃Al CIC conductor using stainless steel was studied for the TFC. As an advanced superconducting magnet technology relevant to the large-scale magnets of DEMO reactor, Nb₃Al superconductor is an advantageous candidate because of the low degradation of the critical current density by the thermal and bending strain [8]. From viewpoints of fabrication reliability and cost, a react-and-wind technique is essential for the fabrication of large TFC (ex. the size of 11 m x 16 m in SSTR). The large TFC can be easily fabricated with high reliability and reduced cost by the technique. Furthermore, the heat treatment time of Nb₃Al is around one third that of Nb₃Sn. Heat treatment of Nb₃Al conductor can be completed in short time (50 hr at 750°C). By the technique without a huge size furnace, an effective cost reduction for the fabrication of the large magnet is expected as compared with a wind-and-react transfer technique.

A half of the 30 m length Nb₃Al full-size conductor was used to fabricate a D-shaped double layer coil with the same bending strain of 0.4% as the fabrication of the TFC of JT-60SC (Fig. 4). In this fabrication by the react-and-wind technique, spring back of the conductor in the winding process should be taken into consideration. Due to existing spring back, an overbending is required to form the

Fig. 3. A view of the 30 m length Nb₃Al full-size conductor in trial manufacture.

Fig. 4. A Nb₃Al D-shaped double layer coil for the demonstration of a react-and-wind technique.
CIC conductor into a designed curvature in the bending process.

The fabrication process of the D-shaped double layer coil by the react-and-wind technique is as follows. 1) Bending the Nb$_3$Al conductor into a radius of 2125 mm by a roller bender. 2) Bending two terminal parts to a radius of 300 mm, which is the same curvature as the terminal joint of the designed TFC, and S-shape with a radius of 150 mm by a bender with three points. 3) Heat treatment (flat top of 50 hr at 750±5°C) with a ring-shaped muffle case in a furnace to precisely control heating temperature. 4) Bending the conductor into a designed shape (test section: R1062.5 mm) by a roller bender and a bender with three points. 5) Shaping into the D-shaped double layer. 6) Fabrication of terminal joints.

After the heat treatment, a compression stress of about 2 tons applied the cable with thermal contraction effect from the SS conduit. However, the cable in the terminal part moved only 1.6 mm. This result indicated that a locking of the cable in the conduit with the curvature of R300 mm was confirmed. Although the bending strain of the conductor at the test section (position of magnetic field application) became 0.40%, the bending strain in the load condition with the overbending became 0.57%. In the other small curvature positions, the maximum bending strain in the load condition was limited to 0.8%. The spring back was estimated to be 197 mm (load condition: R880 mm, free condition: R1077 mm) in the final bending process. The D-shaped double layer coil will be tested to verify the react-and-wind technique with $I_c$ measurement in December 2002. The effect of the overbending in the fabrication process on superconducting performances can be investigated.

3. Development of Reduced Cost NbTi Conductor

A NbTi conductor is applied for equilibrium field (EF) coils in JT-60SC because of relatively low magnetic field (< 5 T) in the windings. For the EF coils, the conductor with low AC loss is required. The NbTi strands coated with Cr can realize low AC loss [9]. However, the Cr coating is rather expensive. Development of NbTi conductor with low cost coating instead of Cr coating is an important issue. A full size NbTi conductor composed of 2 μm Ni-coated strands was newly fabricated. From the AC loss measurement of the conductor, a coupling time constant was measured to be 140 ms (Fig. 5). The Ni-coated NbTi conductor can be adopted for JT-60SC. The Ni coating is the reference candidate for the poloidal field coils of ITER. Therefore, this conductor can be also applied to ITER. The fabrication of reduced cost NbTi conductor has been established.

4. Development of Forced Cooling Divertor Component with High Heat Transfer

Realization of the forced cooling divertor under a high heat load in the range of 10 - 15 MW/m$^2$ is one of the most important issues for JT-60SC [10]. In the activity of the ITER divertor R&D, various types of the divertor targets have been developed using swirl tubes. The critical heat flux of the screw tube (M10, fin pitch of 1.5 mm) near the boiling region was evaluated so far [11]. It was found that the screw tube had a higher heat transfer than the swirl tube. However, the heat transfer coefficient of the screw tube has not been evaluated yet.

A more simplified structure can be applied for the JT-60SC divertor. A prototype flat CFC (carbon fiber composite) target with screw tubes, which had helical fins like a nut, was manufactured aiming at cost-effectively manufactured divertor target with a sufficient heat removal performance for JT-60SC (Fig. 6). The CFC (CX-2002U) tiles were brazed onto a
Cu-alloy heat sink with a 1-mm thick Cu-interlayer at one step of heat treatment (870°C in a vacuum furnace). Prior to the brazing step, four screw tubes were made directly in the Cu-alloy heat sink [12]. The heat removal performance of the CFC target was successfully demonstrated on the JAERI Electron Beam Irradiation Stand. The evaluated heat transfer coefficient of the screw tube (M10, fin pitch of 1.5 mm) at the non-boiling region was roughly 3 times higher than that of the smooth tube of 10 mm inside diameter. This corresponds to 1.5 times that of the swirl tube of 10 mm inside diameter with a tape twist ratio of 3. A heat cycle test of 10 MW/m² showed that the CFC target with the screw tubes could withstand for 1400 cycles. These results indicate that the divertor target plate with the flat CFC tiles and the screw tubes is a promising candidate for the JT-60SC.

Fig. 6. A prototype flat CFC target with heat removal of 10 MW/m².

5. Conclusions

The modification of JT-60 is planned as a full superconducting tokamak (JT-60SC). Advanced fusion technologies relevant to DEMO reactor have been developed in the superconducting magnet technology and plasma facing components for the design of JT-60SC. To achieve a high current density in a superconducting strand, Nb₃Al strands with a high copper ratio of 4 have been newly developed for the TFC of JT-60SC. The R&D to demonstrate applicability of Nb₃Al conductor to the TFC by a react-and-wind technique have been carried out using a full-size Nb₃Al conductor. The D-shaped double layer coil fabricated by the react-and-wind technique in consideration of spring back will be tested to verify the fabrication technique in December 2002. A full-size NbTi conductor with low AC loss using Ni-coated strands has been successfully developed. A forced cooling divertor component with high heat transfer using screw tubes has been developed for the first time. The heat removal performance of the CFC target was successfully demonstrated on the electron beam irradiation stand.

Acknowledgments

The authors would like to thank the members of the Japan Atomic Energy Research Institute who have contributed to the JT-60 project, and appreciate contributions from Superconducting Magnet Laboratory and NBI Heating Laboratory.

References

Advanced Fusion Technologies Developed for JT-60 Superconducting Tokamak


Japan Atomic Energy Research Institute, Naka Fusion Research Establishment Naka-machi, Naka-gun, Batori-ken, 311-0193 JAPAN

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- Objectives of JT-60 Superconducting Tokamak
- Design of JT-60SC
  - Superconducting Coils (TF, CS, EF)
  - Vacuum Vessel (In-vessel Components), Divertor
- Superconducting Magnet Technology developed for JT-60SC
  - Development of 30 m Length Nb3Al Full-size Superconductor R&D for Demonstration of a React-and-wind Technique
- Development of Reduced Cost NbTi Superconductor
  - Ni-coated NbTi Conductor with Low AC loss
- Development of Forced Cooling Divertor Components
  - Prototype Flat CFC Target with High Heat Transfer

Requirements and Issues for Design of JT-60SC

- Superconducting Coils
  - Full Superconducting Tokamak Device (ITER-FEAT, KSTAR, JT-60SC, etc.)
  - Nb3Al cable-in-conduit conductor for TF Coils --> Advanced Magnet Technology
- Stabilizing Baffle Plates
  - Utilization of Reduced Activation Materials (Carbon steel)
  - Passive Stabilizer for VDE and RWM
- Plasma Control
  - Vertical Position Control Coils for VIE Suppression
  - Sector Coils for RWM Stabilization
- Reduction of TF ripple
  - Ferritic Steel inside Vacuum Vessel (1% --> 0.4%)
- Forced Cooling Divertor
  - Water Cooling Divertor (Primary Heat Source of JT-60SC)

Utilization of the Present Facilities (Heating system, Power Supplies, etc.)

Objectives and Issues of JT-60SC

- Objectives are
  - To Establish high performance steady state operation
to demonstrate plasma applicability of reduced activation ferritic steel
1) ESTABLISHMENT OF HIGH PERFORMANCE STEADY STATE OPERATION
  - HIGH BETA PLASMA CONTROL (βn = 3.5 - 5.5)
  - STEADY STATE PLASMA CONTROL (I_p, ps = 50 - 92 kA)
  - DIVERTOR HEATSPOT CONTROL (I_p, ps = 95%, t_p, ps = 5)

2) PLASMA APPLICABILITY TEST OF ADVANCED MATERIALS
  - practical use of the advanced material with reduced activation ferritic steel

<table>
<thead>
<tr>
<th>Parameter</th>
<th>JT-60U</th>
<th>JT-65SC</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pulse length</td>
<td>15 s</td>
<td>10 s</td>
</tr>
<tr>
<td>Max. input power</td>
<td>40 MW (10 s)</td>
<td>11 MW (10 s)</td>
</tr>
<tr>
<td>Elbow shape</td>
<td>3 MA</td>
<td>12 MA</td>
</tr>
<tr>
<td>Toroidal field Bt</td>
<td>4 T</td>
<td>3.6 T (R5 = 2.8 m)</td>
</tr>
<tr>
<td>Major radius Rm</td>
<td>3.4 m</td>
<td>2.6 - 3 m (2.8 m)</td>
</tr>
<tr>
<td>Minor radius rm</td>
<td>0.9 m</td>
<td>1.7 x 0.8 m (0.80 m)</td>
</tr>
<tr>
<td>Elongation ε</td>
<td>1.8 (βn = 0.06)</td>
<td>2.1 (βn = 0.05)</td>
</tr>
<tr>
<td>Triangularity δb</td>
<td>0.4 (δb = 1.35)</td>
<td>0.5 (δb = 1.25)</td>
</tr>
</tbody>
</table>
Modification to Superconducting Tokamak

To be modified with maximum utilization of the present facilities such as torus building, heating systems and power supplies.

Shapes and Displacement of Superconducting Coils

Design condition:
- $B_s=3.8$ T at $R=2.8$ m
- Keep a flux swing of 40 Wb

TF Coil System parameters

<table>
<thead>
<tr>
<th>Items</th>
<th>Value</th>
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<tbody>
<tr>
<td>Size (outboard side)</td>
<td>3.9 m x 6.0 m</td>
</tr>
<tr>
<td>Coil Numbers</td>
<td>18</td>
</tr>
<tr>
<td>$I_{br}$</td>
<td>3.8 T</td>
</tr>
<tr>
<td>Magnetic Energy</td>
<td>1.7 GJ</td>
</tr>
<tr>
<td>Inductance</td>
<td>9.1 H</td>
</tr>
</tbody>
</table>

TF coil shape with small bending strain ($\varepsilon=0.4\%$) is realized.

<table>
<thead>
<tr>
<th>Coils</th>
<th>SC strands</th>
<th>Iop</th>
<th>Bmax</th>
</tr>
</thead>
<tbody>
<tr>
<td>TF Coils</td>
<td>Nb$_3$Al</td>
<td>19.4 kA</td>
<td>7.4 T</td>
</tr>
<tr>
<td>CS, EF4</td>
<td>Nb$_3$Sn</td>
<td>20 kA</td>
<td>7.4 T</td>
</tr>
<tr>
<td>EF1,2,3,5,6</td>
<td>NbTi</td>
<td>20 kA</td>
<td>5.0 T</td>
</tr>
</tbody>
</table>

Design of Superconductor for TF coils

Cable-in-conduit conductor forced cooling by SHe

- Nb$_3$Al strands
  - Decrease of current density with thermal and bending strains is low.
  - Demonstrate a react-and-wind (R&W) technique.
  - Time of heat treatment is short (750°C for 50 hours).
- SS316LN for conduit is used.
- Strand with higher current density, Cu/non-Cu ratio of 4.
  - Reduction of SC material and design of compact TFC.

![Diagram of Superconducting Coils](image-url)
Design of Superconductor for PF coils

- CS and EF4
  - Nb3Sn strands
  - Low AC loss (max. dB/dt = +2.4 T/s)
  - Relatively high Cu ratio (~2.3)
  - SS316LN conduits

Other EF coils
- NbTi strands
- Fine filament (dia. 11 μm) and high Cu ratio (~7)
- Low AC loss (max dB/dt = +2.7 T/s)

Conductors for TF coil, CS and EF coil

<table>
<thead>
<tr>
<th>Structure</th>
<th>for TF coil</th>
<th>for CS &amp; EF4 coil</th>
<th>for EF coil</th>
</tr>
</thead>
<tbody>
<tr>
<td>SS316LN</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Superconducting Strand</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pure copper wire</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

- Max. Magnetic Field: 7.4 T
- Nominal Current: 19.4 kA
- Operating Temp: 4.6 K
- SC Material: Nb3Al
- Coating Material: Cr
- No. of Total Strands: 324
- No. of SC Strands: 216
- No. of Cu Wires: 109
- Cu/Nb Cu Ratio: 4
- Strand Diameter: 0.74 mm
- Void Fraction: 36%
- Weight of SC strand: 32 tons

Conduit fabrication: Butt-welding}

Strand with high Jc and high Cu ratio for TF coil

Because the strand for JT-60SC is used in the region of high current density, the high Cu/non-Cu ratio is required to keep high stability.

Structure of TF coils

- Centering Force: 35.8 MN / coil (B_T = 3.8 T)
  - Wedge support in inner structure

- Overturning Force
  - Intercoil shear panels
  - Shear keys at wedge part

- Gravity support: with plate springs
  - Weight: 40 tons / coil

A compact TF coil (winding) can be designed.
Structure of CS

- Stack of 4 coils
- Self-standing support
- Preload ~ 49 MN
- Buffer zone
- Tightening at RT

Structure of TFC Conductor and Winding

Insulator between turns 1 mm²
Insulator to ground 15 mm²
Insulator between pancakes 2 mm²
Conduit 22.7 x 22.7 mm
Cable φ17 mm

Structure of TFC Winding

Structure of Vacuum Vessel and In-vessel Components

- Reduction of TFC nuclear heating ⇒ Radiation shielding with VV structure
- Installation of ferrite steel for reduction of TF ripple, (≤ 0.4%)
- Stiffness plates: Passive Stabilizer for avoidance of VDE and RWM stability
- In-vessel coils: Control of RWM and HPC/VPC

- Double-walled structure made of SS316L with low Co contamination.
- Single layer shielding with water and shielding plates.

Structure of Vacuum Vessel

- Double-walled structure with a polygon shape
- Consist of 12 mm thick SS316L steel
- Rib structure using H-type steel
- Outer wall connected by plug welding
- Pure water is filled in the space of double wall

Structure of Double-walled Vacuum Vessel
Heat load is evaluated to be 10~15 MW/m² at \( f_{\text{rad}} = 50-65\% \)

<table>
<thead>
<tr>
<th>Heating condition</th>
<th>Transient heat flux (MW/m²)</th>
<th>Required radiation loss fraction ( f_{\text{rad}} ) (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Input power (MW)</td>
<td>heat flux 15 MW/m²</td>
<td>heat flux 10 MW/m²</td>
</tr>
<tr>
<td>44.6/30</td>
<td>25.6</td>
<td>91 (92)</td>
</tr>
<tr>
<td>39.5/30</td>
<td>20.6</td>
<td>92 (94)</td>
</tr>
<tr>
<td>17.7/100</td>
<td>13.0</td>
<td>97 (97)</td>
</tr>
</tbody>
</table>

- Conduction power to the divertor = Input power \( \times (1-f_{\text{rad}}) \)
- Single Null, Piv/Pout = 1/2
- Width of heat flux at SOL \( \leq 8 \times 10^{-8} \) mm (based on scaling law in ITER physics R&D) & flux expansion \( \leq 4 \) at outer divertor

Radiation enhancement for good confinement plasma on JT-60U
- \( f_{\text{rad}} \approx 80\% \) at high density ELMy H-mode with impurity seeding
- \( f_{\text{rad}} \approx 80\% \) at high density RS with impurity seeding
- \( f_{\text{rad}} < 50\% \) at high \( \beta_p \) mode with low recycling

Power handling capability of 10~15 MW/m² is needed for high \( \beta \) operation with 15~30 MW heating.

---

**Development of 30 m Length Nb₃Al Full-size Superconductor**

- Mass production technique of high Cu ratio (~4) Nb₃Al strand was established. (~11 km; no breaking)

- In the test of full-size short sample of CICC, no degradation of Ic by thermal strain with the use of SS conduit was confirmed.
Manufacture of 30 m SS conduit

Material : SS316L
Shape : Round-In-Square type conduit
Unit length : 10m (Hot extrusion and cold work)
Eccentricity of hole tube : 0 ~ 9% (averaged 3%)

Automatic welding
(Manual welding for the corner)

After polishing

Nondestructive test :
PT, RT and check projections of welding bead inside the conduit by CCD camera.

Manufacturing technique of round-in-square type SS conduit with butt-welding was established. (10 m x 3 units)

R&D for Demonstration of a React-and-wind Technique

Advanced magnet technology with Nb$_3$Al conductor by R&W technique for large-scale magnets

Fabrication of 30 m length full-size conductor

Cable was inserted into the 30 m length conduit (Jacketing) and then it was drawn down to the designed size with a square die (Compaction).

A 30 m Nb$_3$Al full-size conductor was successfully completed

Advanced magnet technology with Nb$_3$Al conductor by R&W technique for large-scale magnets

From viewpoints of fabrication reliability and cost, a react-and-wind technique is essential for the fabrication of large-scale TFC (ex. the size of 11 m x 16 m in SSTR).

- High reliability and reduced cost by the technique as compared with a wind-and-react transfer technique.
- Heat treatment of Nb$_3$Al is around one third that of Nb$_3$Sn.
  (50 hr at 750°C without a huge size furnace)

Demerits of R&W

- Overbending is required to form the conductor into a designed curvature in bending or winding process because of spring back.
- After the heat treatment, a compression stress of about 2 tons applied the cable with thermal contraction effect. Moving of the cable in the terminal part is a mater of concern.

Manufacturing technology of the CIC conductor for JT-60SC TF coil (~ 300 m) has been established.
Demonstration of R&W technique with D-shaped double layer coil

- D-shaped double layer coil was fabricated by R&W technique using full-sized Nb$_3$Al CICC.
- External magnetic field will be applied to a test section with the same radius ($r = 1062.5$ mm, $\varepsilon \sim 0.4\%$) as curvature of the designed TF coil.
- Pool boiled cooling in a cryostat. ($T = 4.2K$)

$I_c$ measurement under the bending strain including overbend at 6.5 $\sim$ 12 T.

Fabrication of D-shaped double layer coil (1)

Bending the Nb$_3$Al conductor into R2125 mm by a roller bender.

Heat treatment: 750 ±5°C for 50 hour with a ring-shaped muffle case.

Fabrication of D-shaped double layer coil (2)

Bending into R1062.5 mm. Bending strain: 0.4% (free) Overbend: 0.57% (load condition) Spring back: 197 mm

The fabrication was completed. The test will be done to verify R&W technique in Dec. 2002.

Development of Reduced Cost NbTi Superconductor with Low AC Loss
Development of NbTi conductor with low AC loss

Candidates of coating material for NbTi strand
Cr (ITER-CSMC) : expensive
SnAg + oxidation (LHC) : low cost
Oxidation of CICC is not established
Ni (ITER-PF insert) : low cost

Because of the advantages of cost and low coupling time constant, Ni-coated NbTi strand is adopted as the first candidate.

AC loss measurement with full-size short samples.

Ni coated NbTi strand with φ 11 μm filament

NbTi strand with Cu ratio of 7 developed for EF coils.

Measured Jc of strand was sufficient for the designed value.

Advantage of Screw Tube for Heat Transfer

Cooling conditions
Inlet temperature : room temp.
Pressure : 1 MPa (center of test sample)

Pumping power per unit length (kW/m) = Pressure loss per unit length (Pa/m) \times Volume flux (m^3/sec)

Fin pitch dependence of Critical Heat Flux (CHF)

The screw tube has the advantage of pumping power compared with swirl tube.

The screw tube with M10 and fin pitch 1.5 mm has the highest CHF.
Structure of Prototype CFC Target

Thermocouples (for the evaluation of heat transfer coefficient)
- Fiber direction
- Thermocouple (ch1)
- Thermocouples (for heat cycle test) (ch2)
- Size of screw fin

Screw tube

Fabrication Procedure of Prototype CFC Target

CFC tiles (CX-2002U): 40 x 40 x 10 mm

Brazing condition
- at 870°C, maintain for 15 min
- CFC-Cu brazing material: Ag-Cu-Ti
- Cu-Cu, Cu-SUS brazing material: Ag-Cu

Header: SS304

Screw tube

Heat sink (Cu-alloy)

- CFC tiles were brazed onto a Cu-alloy heat sink with the Cu-interlayer at one step.

40 x 40 x 1 mm Cu-interlayer to reduce thermal stress between the CFC tiles and the heat sink.

Results of Experiment and FEM Analysis

The heat transfer coefficient of the screw tube was evaluated to be roughly 3 times that of the smooth tube.

Comparison between the results from the experiments and the FEM analyses (cooling water: 8.0 m/s, 0.74 MPa).

Heat Load Test of Prototype CFC target for Divertor Plates

Simplified structure
- Low coat

Structure of prototype CFC target (forced water cooling)

<Results>
- Temperature of inlet water: RT, Heat load: 10 MW/m² for 30 s, Flow speed of water: 8 m/s,
- Water pressure: 0.74 MPa

Heat cycle test was carried out for 1400 cycle at 10 MW/m² for 1.5 s by using JEBIS.
The reliability of prototype CFC target was confirmed.
Summary

- The modification of JT-60 is planned as a full superconducting tokamak.
  - Design of JT-60SC has been completed. Now under detailed design phase.
  - Engineering design of the main components has been established through R&D for JT-60SC.

- Advanced fusion technologies have been developed in the superconducting magnet technology and plasma facing components for the design of JT-60SC.
  - To achieve a high current density in a superconducting strand, Nb$_3$Al strands with a high copper ratio of 4 have been newly developed for the TFC.
  - The R&D to demonstrate applicability of Nb$_3$Al conductor to the TFC by a react-and-wind technique have been carried out. The test will be done to verify the fabrication technique in Dec. 2002.
  - A full-size NbTi conductor with low AC loss using Ni-coated strands has been successfully developed.
  - A forced cooling divertor component with high heat transfer using screw tubes has been developed for the first time.

- The modification to JT-60SC is being referred to governmental permission.
1.27 Performance of ITER as burning plasma experiment


1) International Team, ITER Naka JWS, Mukouyama, Naka-machi, Naka-gun, Ibaraki-ken, 311-0193, Japan,
2) International Team, ITER Garching JWS,
3) Toshiba Corp., Minato-ku, Tokyo
4) Kurchatov Institute, Moscow, Russia
5) Japan Atomic Energy Research Institute, Naka-machi, Ibaraki-ken, Japan

e-mail contact of main author : shimadm@itergps.naka.jaeri.go.jp

Abstract. Recent performance analysis has improved confidence in achieving $Q \geq 10$ in inductive operation in ITER. Performance analysis based on empirical scaling shows the feasibility of achieving $Q \geq 10$ in inductive operation, particularly with improved modeling of helium exhaust. Analysis has also elucidated a possibility that ITER can potentially demonstrate $Q$'s ~ 50, enabling studies of self-heated plasmas. Theory-based core modeling indicates the need of high pedestal temperature (2.3 - 4.5 keV) to achieve $Q \geq 10$, which is in the range of projection with presently available pedestal scalings. Pellet injection from high-field side would be useful in enhancing $Q$ and reducing ELM heat load in high plasma current operation. If the ELM heat load is not acceptable, it could be made tolerable by further tilting the target plate. Steady state operation scenarios at $Q = 5$ have been developed with modest requirement on confinement improvement and beta ($\beta_n(1.3) \geq 2.6$). Stabilisation of RWM, required in such regimes, is feasible with the present saddle coils and power supplies with double-wall structure taken into account. Recent analysis shows a potential of high power steady state operation with a fusion power of 0.7 GW at $Q \approx 8$. Achievement of the required $\beta_n \sim 3.6$ by RWM stabilisation is a challenge and further analysis is also needed on the reduction of the divertor target heat load.

1. Introduction

Analysis of ITER plasma performance is being carried out to confirm the integrity of core, pedestal and divertor characteristics. The core performance analysis described in Final Design Report of ITER [1] was based on empirical scaling. Recently efforts have been focused on projection with theory-based modeling in the core. The pedestal temperature has been found to play an important role in core confinement. Improved assessment of the erosion of the target has led to an increase in the tolerable ELM heat load by a factor of ~ 2. The ELM amplitudes show reduction toward high edge collisionality and high frequency, suggesting that pellet-induced or spontaneous frequent ELMs can suppress the ELM amplitudes to a benign level in ITER. Steady state operation scenarios, with less demanding confinement improvement and beta, have been developed, and analysis of RWM suggests that the present set-up of pick-up coils, vacuum vessel and coils is adequate for RWM stabilisation. Recently scenarios have been developed for high power steady state operation, in the prospect of developing a core plasma for the next step. This paper summarises recent progress in these areas for ITER projection.

2. Inductive operation

Performance analysis based on empirical scaling demonstrates the feasibility of achieving its mission of $Q \geq 10$ in inductive operation [1,2]. Divertor transport calculations by the B2/Eirene code indicate that the steady-state target heat load can be reduced to < 10 MW/m² and the helium concentration to < 3% at the separatrix [3], which corresponds to < 4.3 % at the axis. Inclusion of helium elastic scattering in the divertor plasma further enhances the
helium exhaust efficiency by a factor of ~ 3 [4], which increases Q, e.g. from 10 to 14. Figure 1 shows Q vs. $P_{\text{sep}}$ at a condition that the separatrix power higher than LH transition power at a plasma current of 15 MA with different levels of helium concentration [2], suggesting the potential of operating at Q's higher than 50 for the investigation of self-heated plasmas.

Core performance projection is also in progress with theory-based modeling e.g. Weiland [5], Multi-Mode [6] and GLF23 [7] models. Figure 2 shows Q vs. pedestal ion temperature ($T_{\text{ped}}$) calculated with Weiland and MMM95 models, showing that the goal of $Q \geq 10$ is achievable with $T_{\text{ped}} \geq (2.3 - 3.9)$ keV, while the IFS/PPPL model requires $T_{\text{ped}} \geq 4.5$ keV [2]. Analysis of the international pedestal database suggests that achievement of this high pedestal temperature is possible [2,6,8,9,10]. Figure 3 shows a scaling of pedestal pressure compared against experimental data in International Pedestal Database v. 3. This scaling projects a temperature of 5.3 keV for a pedestal density of $7 \times 10^{19}$ m$^{-3}$ [11], suggesting that $Q \geq 10$ is achievable. $Q \geq 10$ operation at a plasma current of 15 MA is associated with $\beta_n \geq 1.5$, which could trigger neoclassical tearing modes (NTMs). Present analysis suggests that stabilisation of fully-grown 2/1 and 3/2 of NTM could require electron cyclotron current drive (ECCD) power of ~ 30 MW [12]. An early detection of the island with a size w/a ~ 0.04 and subsequent ECW injection could enable mode stabilisation within the initial capability of the ECCD/ECH system (20 MW) [13].

3. ELM

The projection of the heat load with type-I ELMs in the inductive high Q operation is subject to a large uncertainty. However, recent calculations show that with a reasonable rise time (0.3 ms) of target heat load and a reasonable heat conductivity taken into account, the tolerable target heat load is ~ 1 MJ/m$^2$ [14] or ~ 6 MJ per ELM pulse with a total ELM footprint of 6 m$^2$, which corresponds to about 6% of the pedestal energy. The projected ELM heat load is 5 - 20 MJ in ITER [14]. As will be discussed in the following section, pellet injection could reduce the ELM amplitudes to a benign level. If the heat load is excessive, further inclination of the target would increase the heat load that can be tolerated. Furthermore, extension of the lifetime of the target plates is possible, e.g. operation with more benign type-II ELMs with a small decrease of plasma current [15]. Therefore in steady-state operation and long pulse hybrid operation with a reduced plasma current, a long lifetime is expected.

4. Pellet injection

High-field side pellet injection has proved to be successful in maintaining good confinement at densities close to the Greenwald density ($n_{\text{w}}$) [16,17]. Reduction of ELM heat load is also observed [18, 19]. This fuelling method will be one of the major fuelling methods in ITER and is expected to enhance the Q value and reduce the ELM heat load [20]. Projected fusion power $> 450$ MW, at an auxiliary heating power of 23 MW and $Q$ of ~ 20 can be achieved at line-averaged density below the Greenwald density with pellet injection from the high field side at a moderate pellet speed of ~ 500 m/s. The ELM heat loss is observed to decrease to 4-5% of the pedestal energy with increasing pedestal collisionality in the high density range [21]. Experiments in ASDEX-Upgrade show that the energy loss during pellet-induced ELM is reduced with ELM frequency following the same scaling as spontaneous ELM obtained in JET and ASDEX-Upgrade experiments [19]. Thus pellet injection would increase the collisionality of the pedestal and it is expected to reduce the heat load below a tolerable range in ITER [20]. Calculation shows that the increase in ELM frequency from ~ 1 Hz (without
pellets) to 4 Hz (with pellets) reduces the ELM energy loss from 10 – 20 MJ to below 6 MJ (Fig. 4). The recovery of the pedestal temperature is much faster than the pellet interval, suggesting that the core confinement remains high after the pellet. However, more work is needed to optimise the pellet parameters and to confirm the compatibility of the pellet injection with good confinement and low ELM heat load.

5. Steady state operation

Operational scenarios have been developed for steady-state operation with modest requirements on confinement and $\beta$; e.g. $H_{\text{499(5)}} \geq 1.3$, $\beta_n \geq 2.6$ [22], with $I_p = 9$ MA, $Q = 5$, $n/n_G = 0.83$, $Z_{\text{eff}} = 2.2$ and $P_{\text{ion}} = 68$ MW. The plasma parameters are shown in Table 1. Divertor transport calculations by the B2/Eirene code show that long connection length with steady-state operation at higher safety factor facilitates divertor compatibility even with a higher fusion power [3]. As the required $\beta_n$ exceeds the ideal MHD no-wall limit for these scenarios by 10-20 %, suppression of resistive wall modes (RWMs) will become a key issue.

Stability against ideal modes has been analysed for three sets of safety factor and pressure profiles shown in Fig. 5 (a) with the KINX code [22]. The minimum $q$ of these reverse shear equilibria are 2.1, 2.25 and 2.4. As neoclassical heat and particle diffusivities are assumed inside the minimum-$q$ radius, the pressure profile becomes flatter with higher minimum $q$. The most dangerous mode is an external $n = 1$ kink mode coupled to internal modes. The stabilising ideal wall position $a_w/a$ is shown against normalised $\beta$ in Fig. 5 (b). The no-wall limit increases with flatter pressure profiles with higher minimum $q$. The ideal wall radius $a_w$ for marginal stability increases significantly with a flatter pressure profile at higher minimum $q$. The effective wall position $a_w/a$ is 1.375 for the plasma shown in Table 1, shifted outward by 0.15 m with a reduced minor radius ($a = 1.85$ m). At this wall position, the marginal $\beta_n$ is 2.6 for a minimum $q$ of 2.1 and 3.6 for a minimum $q$ of 2.4. A full-bore plasma with $R = 6.2$ m and $a = 2$ m is more stable against ideal modes with $a_w/a = 1.345$, suggesting that high $\beta_n \sim 3.6$ steady state operation with $Q \sim 10$, described in the following section, could be made stable with an ideal wall.

Analytical study has been carried out with double-wall structure of the ITER resistive vacuum vessel taken into account. The double wall structure does not affect the RWM growth rate significantly, but deteriorates the feedback stabilisation. However, the present arrangement of saddle coils and power supply is adequate for RWM stabilisation for the range of $\beta_n$ anticipated for the steady state operation scenarios quoted above [23, 24].

6. High performance steady state operation

Recently high power steady state operation has attracted much interest from the viewpoint of developing a core plasma of a fusion power plant, in which the requirement is more demanding on $\beta$, $Q$, power and particle control and bootstrap current fraction. Figure 6 shows profiles of current density, safety factor, temperature and electron density calculated with the ASTRAN code for possible ITER conditions. A combination of neutral beam current drive at the core, lower hybrid current drive at $t/a \sim 0.7$ and bootstrap current provides a 12 MA weak-reverse-shear (WRS) steady state plasma, with a $q_{\text{min}}$ at $t/a \sim 0.7$. $q_{\text{off}} = 4.76$, $H_{\text{499(5)}} = 1.53$, and fraction of Greenwald density $n_e/n_G = 0.86$. The bootstrap current fraction is $54.5 \%$. The fusion power is 700 MW and the current drive power is 47 MW (NB) and 40 MW (LH), giving a $Q$ value of 8. Neoclassical heat and particle diffusivities are assumed inside the radius of minimum $q$. A burn phase of $\sim 300$ s can be sustained with the present ITER
hardware, which is adequate to reach a quasi-steady state with an optimised start-up scenario. The value of $\beta_p$ in this discharge is 3.6, which is above the no-wall limit of 2.8 and below the ideal wall limit of 3.8. Requirement on RWM stabilisation is being analysed.

In addition to beta, the divertor heat load and helium exhaust are also of major concern in high power operation. The divertor performance was estimated with a scaling [25] derived from a series of B2-Eirene runs at $q_{95} = 3$. Since this discharge has a higher safety factor ($q_{95} = 4.76$), this estimate provides a conservative value. Figure 7 shows peak target heat load, separatrix electron density, helium concentration at the separatrix, and DT throughput with a scrape-off layer power of 155 MW and a pumping speed of 10 $m^3/s$ for the discharge shown in Fig. 6. With an increase in DT throughput, the helium concentration and peak heat load are reduced substantially, reaching 0.35% and < 5 MW/m$^2$, respectively at a DT throughput of 200 Pa m$^3/s$. Although the helium density at the separatrix is maintained at a very low level, further analysis is needed on helium transport with the internal transport barrier.

A high fusion power (~ 1 GW) plasma condition, i.e. high $\beta_p$ and high power exhaust, can be simulated with a plasma with an isotopic fraction of ~ 0.2 or 0.8 at an acceptable fusion power in ITER, i.e. 700 MW. The reduced alpha heating power can be compensated for by increased additional power, as shown in Fig. 8. To maintain the fusion power below 700 MW and additional power below 115 MW, the operation point should fall into either one of the two shaded regions in the figure. The plasma current of 15 MA, $\beta_p = 3.0$, $n_{e0}/n_e = 0.9$, $n_e/n_{i0} = 1.3$, and $H_{W(n_{i0}, 2)} = 1.3$ are assumed. The expected bootstrap current is 33% and the total non-inductively driven current is 57%. This operation can be sustained for ~ 300 s.

7. Conclusions

1) Performance analysis based on empirical scaling demonstrates the feasibility of achieving $Q \geq 10$ in inductive operation, especially with improved model of helium exhaust.

2) Theory-based core modeling indicates the need of high pedestal temperatures (2.3 - 4.5 keV) to achieve $Q \geq 10$, which is in the range of projection with presently available pedestal scalings.

3) The heat load of type-I ELM in high plasma current operation could be made tolerable by high density operation and further tilting the target plate (if necessary).

4) Pellet injection from the high-field side would be useful in enhancing $Q$ and reducing ELM heat load.

5) Steady state operation scenarios to achieve $Q = 5$ have been developed with modest requirement on confinement improvement and beta ($H_{W(n_{i0}, 2)} \geq 1.3$ and $\beta_p \geq 2.6$). Stabilisation of RWM, required in such regimes, is feasible with the present saddle coils and power supplies with double-wall structure taken into account.

6) Recent analysis shows a possibility of high power steady state operation with a fusion power of e.g. 0.7 GW at $Q \sim 8$. Achievement of the required $\beta_p \sim 3.6$, above the no-wall limit (2.8) and below ideal wall limit (3.8), by RWM stabilisation is a challenge and further analysis is also needed on the reduction of the divertor target heat load.

7) With an isotopic mixture of ~ 0.2 or ~ 0.8, there is a possibility of simulating a plasma
condition with a fusion power of 1 GW, e.g. high $\beta$ and power and particle control within the capability of the ITER hardware.

Reference
[2] MUKHOVATOV, V., et al., this Conference, CT/P-03.

[22] POLEVOI, A., et al., this Conference, CT/P-08.
[23] GRIBOY, Y., et al., this Conference, CT/P-12.

Table 1. ITER plasma parameters for the steady state scenario

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
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<th>Value</th>
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<tbody>
<tr>
<td>R/a, m</td>
<td>6.35 /1.85</td>
<td>$&lt;T_e&gt;/&lt;T_i&gt;$, keV</td>
<td>11/12-10.5/11</td>
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<tr>
<td>$\delta_p/\kappa$</td>
<td>0.41/1.84</td>
<td>$w_i/w_{i,tot}$, MJ</td>
<td>273/60-255/50</td>
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<tr>
<td>$q_{95}$</td>
<td>5.16-5.13</td>
<td>$H_{\beta_S(2)}$</td>
<td>1.41-1.3</td>
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<tr>
<td>I_p MA/$&lt;n_i&gt;$, 10^{20}m$^{-3}$</td>
<td>9.674</td>
<td>Q</td>
<td>5.7-5</td>
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<tr>
<td>$\beta_n$</td>
<td>2.8-2.96</td>
<td>$P_{NP}/P_{tot}$, MW</td>
<td>34/29-33.7</td>
</tr>
<tr>
<td>$\beta_i$</td>
<td>0.72-0.63</td>
<td>$P_{i,i}$, MW</td>
<td>361/93-338/97</td>
</tr>
<tr>
<td>$&lt;Z_{eff}&gt;$</td>
<td>2.2-2.17</td>
<td>$\delta_i, \delta_i$</td>
<td>2.54-2.32</td>
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</table>
FIG. 1. $Q_{\text{max}} = P_{\text{helix}}/P_{\text{add}}$ ($P_{\text{helix}} = P_{L-H}$) vs. $H_{\text{helix}}$, with different levels of helium at a plasma current of 15 MA, showing that significant improvement of performance is expected with improved helium modeling, and that $Q > 50$ is a possibility.

FIG. 2. $Q$ versus $T_{\text{ped}}$ predicted for ITER by the Multi-Mode and Weiland models. Dashed line shows a value of $Q$ compatible with $P_{\text{sep}} = 1.3 \times P_{L-H}$, and horizontal bars show the ranges of $T_{\text{ped}}$ predicted for ITER by different pedestal scalings.

FIG. 3. Scaling of Pedestal pressure compared against experimental data in International Pedestal Database. This scaling projects a pedestal temperature of 5.3 keV for a pedestal density of $7 \times 10^{19}$ m$^{-3}$ in ITER.

FIG. 4. Dependence of fraction of pedestal energy loss $\Delta W_{\text{ELM}}/W_p$ on the pedestal collisionality $v_p$. Experimental points [20] are shown by closed diamonds together with predictions for ITER (open points) with and without pellet.
**FIG. 5 (a).** Three different $q$ profiles for ideal kink mode configuration of weak reverse shear for steady state operation. The minimum $q$'s are 2.1 (dotted), 2.26 (dashed) and 2.4 (solid). Plasma pressure profiles are also shown, which are calculated with neoclassical heat and particle diffusivity inside the minimum-$q$ radius.

**FIG. 5 (b).** Stabilising wall position $a_w/a$ vs. normalised beta $\beta_n$ for $q$=const scan of SS operational points shown in FIG. 5 (a), and $a = 1.85$ m. The no-wall limits are shown by vertical lines. The $a_w/a$ of a reduced-size plasma of ITER shifted 0.15 m outward is indicated with a horizontal dotted line.

**FIG. 6 (a).** Radial profiles of current density and safety factor for a steady state discharge with 700 MW fusion power. Combination of neutral current drive at the core, RF current drive at 70% of minor radius and bootstrap current results in weak reverse shear.

**FIG. 6 (b).** Radial profiles of ion and electron temperature and electron density for the discharge in FIG. 6(a). Neoclassical heat and particle diffusivity are assumed inside the radius of minimum $q$. 

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FIG. 7 (a). Peak target heat load against separatrix density for the condition of high power steady state discharge shown in FIG. 6. Power steady state discharge shown in FIG. 6. Increased particle throughput results in increase in separatrix density and decrease in peak target heat load.

\[ I_p = 15 \text{ MA}, \beta_n = 3.0, \frac{\langle n_d \rangle}{n_G} = 0.9, \frac{n_0}{\langle n_e \rangle} = 1.3, H_{H98(y,z)} = 1.3 \]

FIG. 8. Fusion power against DT isotope ratio. Reduction of alpha-heating power can be compensated for by increase in additional power. The operation point should be in either one of the two shaded zones, where the fusion power is within the hardware limit, i.e. 700 MW, and the required additional power is within the power available, i.e. 115 MW.
Performance of ITER as burning plasma experiment

M. Shimada 1), V. Mukhovatov 1), G. Federici 2), Y. Grigov 1),
A. Kukushkin 2), Y. Murakami 3), A. Polevoi 3), V. Pastovskiy 3),
S. Sengoku 4), M. Sugihara 4)

In collaboration with International Tokamak Physics Activity under the auspices of IAEA IFIRC
and previous ITER Physics Expert Groups

1) International Team, ITER Naka, JWS, Naka-machi, Naka-gun, Ibaraki-ken, Japan,
2) International Team, ITER Garching, JWS,
3) Toshiba Corp., Minato-ku, Tokyo
4) Kurchatov Institute, Moscow, Russia
5) Japan Atomic Energy Research Institute, Naka-machi, Naka-gun, Ibaraki-ken, Japan

Goals of ITER experiments

High Q plasma experiments
Q ≥ 10, βn < 2, NTM, H_i ~ 1, ELM, divertor

Burning steady state experiments
Q ≥ 5, βn < 3, H_i ~ 1.3, divertor, f_{el} ~ 0.5, RWM, TAE

Development of reactor plasma
Q ~ 10, P_{max} ~ 0.7 GW, steady state
βn > 3, H_i ~ 1.5, divertor, f_{el} > 0.5, RWM, TAE

Outline

High Q plasma experiments
• Performance projection with empirical v_i scaling
• Performance projection with theory-based modeling
• ELM mitigation by pellet injection

Burning steady state operation
• Scenario with modest requirement on confinement and β
• Resistive Wall Mode stabilisation

Development of reactor plasma
• Requirement on beta and confinement
• Requirement on power and particle exhaust
• Possible simulation of ~ 1 GW long pulse operation

Impurity transport of tungsten

Conclusions

High Q plasma experiments
Performance projection with empirical v_i scaling
Recent divertor modeling with DT-He elastic scattering improved the efficiency of helium exhaust [Kukushkin (PPCF-2002) and core performance [Mukhovatov, CT/P-03]

Heating concentration at the separatrix vs. DT particle throughput
Open symbols: without elastic scattering
Closed symbols: with elastic scattering
Q_{max} = P_{max}/P_{core} (P_{max} = P_{ion} - H_{cusp,ion}) with different levels of helium at a plasma current of 15 MA, showing that significant improvement of performance is expected with improved helium modeling, and that Q > 50 is a possibility.
Performance projection with theory-based modeling

Theory-based projection of core performance indicates that the goal of $Q = 10$ is achievable with $T_{\text{ped}} = (2.3 - 4.5)$ keV, which is in the range of projection from the international pedestal database. [Mukhaev, CT/P-03]

Scaling of pedestal pressure compared against experimental data in International Pedestal Database. This scaling projects a pedestal temperature of 6.3 keV for a pedestal density of $7 \times 10^{21}$ m$^{-3}$ in ITER.

ELM mitigation by pellet injection

Pellet injection could improve the core performance and reduce the ELM amplitudes to a benign level. The pedestal temperature recovers quickly ($\ll$ pellet period) after pellet, suggesting that the core confinement is maintained [Polevoi, CT/P-09].

If the heat load is still excessive, further inclination of the target and/or operation with benign type-II ELMs would extend the target lifetime.

Burning steady state experiments:
Scenario with modest requirement on confinement and $\beta$

Steady state operation scenarios to achieve $Q = 5$ have been developed with modest requirements on confinement improvement and beta ($H_{\text{limb}}/R \geq 1.3$ and $\beta_n = 2.6$) [Polevoi, CT/P-08].

<table>
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<td>6.35/1.85</td>
<td>$&lt;T_{\text{e,ped}}, &lt;T_{\text{e,pl}}$, keV</td>
<td>11/72-10.5/11</td>
</tr>
<tr>
<td>$\delta_{\text{eo}}$, $\delta_{\text{W}}$</td>
<td>0.41/1.84, 0.41/1.84</td>
<td>$W_{\text{eo}}/W_{\text{e,W}}$, MJ</td>
<td>273/60-255/50</td>
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<td>$q_{\text{in}}$</td>
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<td>$H_{\text{limb}}/R$, 1.41-1.3</td>
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<tr>
<td>$l_p$, MA</td>
<td>9</td>
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<tr>
<td>$&lt;n_{\text{e,ped}}, 10^{19}$ m$^{-3}$</td>
<td>6.74</td>
<td>$P_{\text{eo}}/P_{\text{e,W}}$, MW</td>
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<tr>
<td>$&lt;n_{\text{e,pl}}, n_{\text{e,ped}}$</td>
<td>0.81</td>
<td>$P_{\text{eo}}/P_{\text{e,W}}$, MW</td>
<td>361/93-339/97</td>
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<tr>
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<td>$t_f$, s</td>
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<tr>
<td>$l_t$</td>
<td>0.72-0.63</td>
<td>$&lt;T_{\text{e,pl}}$</td>
<td>2.2-2.17</td>
</tr>
</tbody>
</table>

Resistive Wall Mode stabilisation

With pressure and current profile control, the beta required in a steady state operation is such that:

$$0 \leq \beta_n - \beta_n^{(\text{no wall})} \leq 0.5$$

$$\beta_n^{(\text{ideal wall})} - \beta_n^{(\text{no wall})}$$

Analytical and numerical studies of RWM stabilisation have been carried out with the double-wall structure of the ITER resistive vacuum vessel taken into account.

The double wall structure does not affect the RWM growth rate significantly, but reduces the effectiveness of the feedback stabilisation.

However, the present arrangement of saddle coils and power supply is adequate for RWM stabilisation for the range of $\beta_n$ anticipated for the steady state operation scenarios quoted above [Gribov, CT/P-12, LiuBonde, TH/P-3-12].
Development of reactor plasma
Recent analysis shows a possibility of high power steady state operation with a fusion power of e.g. 0.7 GW at $Q = 8$. $I_p = 12$ MA, weak-reverse-shear (WRS), $q_{95} = 4.76$, $H_{99,95} = 1.53$, $\alpha_{\text{he}} = 0.86$, $\beta_p = 3.6$ (no-wall limit: 2.9, ideal wall limit: 3.6) [Popeko, CT/P-08], sustainable for $\sim 300$ s.

Possible simulation of $\sim 1$ GW long pulse operation
A high fusion power ($\sim 1$ GW) plasma condition (e.g. high $\beta$, power and particle exhaust) can be simulated with an isotopic fraction of $\sim 0.2$ or 0.8 at an acceptable fusion power in ITER, i.e. 700 MW. The reduced $\alpha$ heating power can be compensated for by increased additional power. $I_p$ (GS + CD) = 6.5 MA, sustainable for $\sim 300$ s.

Requirements on power and particle exhaust
The divertor performance was estimated with a scaling [Pacher (PSI-2002)] derived from a series of B2-Eirene runs, with a scrape-off layer power of 155 MW for a steady state high power discharge. With an increase in DT throughput, the helium concentration and peak heat load are reduced, reaching 0.35% and < 5 MW/m².

Impurity transport of tungsten
Neoclassical transport of tungsten impurity has been analysed with different density profiles [Murakami (PSI-2002)]. The convective velocity is inward at the periphery, where the density gradient is strong. However, the convective velocity is outward where the density is flat, due to the ion temperature gradient effect.
Consequently the tungsten impurity profile could be made hollow, which could enable reduction of divertor heat load without cooling the core.
Conclusions

1) Burning plasma experiments (Q \geq 10, inductive)

- Performance analysis based on empirical scaling demonstrates the feasibility of achieving Q \geq 10 in inductive operation, especially with improved model of helium exhaust.

- Theory-based core modeling indicates the need of high pedestal temperatures (2.3 - 4.5 keV) to achieve Q \geq 10, which is in the range of projection with presently available pedestal scalings.

- Pellet injection could improve the core performance and reduce the ELM amplitudes to a benign level. If the heat load is still excessive, further inclination of the target and/or operation with benign type-II ELMs would extend the target lifetime.

Conclusions (continued)

2) Burning steady state experiments

Steady state operation scenarios to achieve Q \geq 5 have been developed with modest requirement (H_{\text{break}} > 1.3 and \beta_p > 2.6). Stabilisation of RWM is feasible with the present saddle coils and power supplies with double-wall structure taken into account.

3) Development of reactor-core plasmas

Recent analysis shows a possibility of high power steady state operation with a fusion power of e.g. 0.7 GW at Q \sim 8. Achievement of the required \beta_p \sim 3.6, above the no-wall limit (2.8) and below ideal wall limit (3.8), by RWM stabilisation is a challenge and further analysis is also needed on the reduction of the divertor target heat load.
1.28 Design Improvements and R&D Achievements for VV and In-vessel Components Towards ITER Construction


1) ITER International Team, Boltzmannstrasse 2, 85748 Garching, Germany
2) EFDA-CSU, Boltzmannstrasse 2, 85748 Garching, Germany
3) JAERI, Naka Establishment, Naka-machi, Naka-gun, Ibaraki-ken, Japan
4) NTC Sintez, Efremov Inst., 189631 Metallostroy, St. Petersburg, Russia

e-mail contact of main author: iokik@itereu.de

Abstract. There have been several detailed vacuum vessel (VV) design improvements, such as elimination of the inboard triangular support, separate interspace between inner and outer shells for independent leak detection of field joints and revised VV support system to gain a more comfortable margin in the structural performance. The blanket design has been updated; an inter-modular key instead of two prismatic keys and a co-axial inlet-outlet cooling connection instead of two parallel pipes. One of the most important achievements in the VV R&D has been demonstration of the necessary assembly tolerances. Further development of cutting, welding and non destructive tests (NDT) for the VV has been continued, and thermal and hydraulic tests have been performed to simulate the VV cooling conditions. With regard to the R&D for the FW/blanket and divertor, full-scale prototypical mock-ups of the FW panel, the blanket shield block and the divertor components have been successfully fabricated. These results make us confident in the validity of our design and give us possibilities of alternate fabrication methods.

1. Introduction

Procurement specifications are now being prepared for ITER components whose delivery is on the critical path, such as the vacuum vessel. Although the basic concept of the VV and in-vessel components of the ITER design has stayed the same, there have been several detailed design improvements resulting from efforts to raise reliability, to improve maintainability and to save money. R&D activities have been continued to confirm the design validity and to develop alternate cost saving fabrication methods.

2. Vacuum Vessel

2.1 Vacuum Vessel Design

The VV is similar to the earlier vessel design in basic features such as structure (double wall), basic shape (torus) and material (SS 316L(N)-IG, ITER Grade) [1]. However, the VV design has been improved taking into account fabrication methods and cost benefits. Difficulties in designing blanket supports in that region have also led to the elimination of the inboard triangular support (see FIG. 1), simplifying the VV structure and also reducing the blanket module electromagnetic load (the poloidal torque of the inboard lowest module). The full isolation of the volume in the interspace between the inner and outer shells at field joint regions is considered to achieve independent leak detection during VV assembly. Access for the VV
thermal shield assembly, fabricability of the port stubs, and plasma vertical stability altogether require the design of the upper port to avoid large chamfers in connection with the main vessel. Each VV sector is now supported at the lower port to the cryostat floor (see FIG. 2), as well as supported in the toroidal direction with mechanical restraints attached between the main vessel and the TF coil. This system gives a more comfortable margin in the structural performance and provides better access for assembly and maintenance. The in-vessel diagnostics design has also progressed avoiding interference with other in-vessel components. Design of the divertor supporting structure on the vessel has also progressed. The support structure is integrated with the VV inner shell, and rails are prepared for travel of the divertor cassette in the toroidal direction, as shown in FIG. 3. A special structure is being considered in the VV field joint region to avoid interference with ultrasonic testing (UT) inspection tool.

The main concept of the vacuum vessel envisages a double wall structure. However, a single wall structure is feasible in some regions (for components near to the cryostat) and is a simpler solution in VV port extensions in particular where the blanket cooling system penetrates. Most welds in the inner shell will have butt-joint configurations with both-side access and will be radiographically inspected to assure 100% weld efficiency. However, the one-sided weld joints between the outer shell and the ribs/housings, and the field joints, cannot be radiographically inspected and so will be inspected by UT or "progress LPT (liquid penetrant dye test)", and a "code case" will be justified by testing. The current approach of the weld joint configurations is to minimize required code cases.

2.2 Vacuum Vessel R&D

One of the most important achievements in the VV R&D is demonstration of the necessary assembly tolerances. This has already been achieved for the main vessel, and more recently an integration test of the port extension has been performed with the full-scale sector model. As a continuous activity of VV R&D to demonstrate the fabrication, assembly and maintenance[2], the port extension was cut and deformation was measured. A manual plasma-cutting tool was
employed for this operation. Before and after the cutting work, dimensions of the VV section and the port section were measured in order to obtain the deformation. Results showed no global deformation except in the port edge section, where a slight expansion up to 3-4 mm was observed. The reason can be explained by an effect of local angular distortion due to welding. During the port integration test before the port extension cutting, the NDT inspection of the field joint between the VV sector and the port extension was successfully performed in a remotitized mode using a robot to move a scanning device (FIG. 4). It was demonstrated that defects as small as 2-3 mm in diameter and 5mm in length can be detected for 60mm thick austenitic stainless steel plate.

Thermal-hydraulic tests have been performed to confirm the VV cooling parameters for future licensing. The tests were focused on study of (1) heat transfer in differently oriented and non-uniformly heated rectangular channels; (2) flow distribution and stability in the parallel channels at the extremely low water velocities (few mm/s per second); (3) development of natural circulation in the entire VV cooling circuit. The VV test element as a rectangular channel (FIG. 5) 0.2 m in width, 3 m in total length, 2.48 m in heated length and with variable channel height (12.5, 25 or 50 mm) has been fabricated to examine heat transfer coefficients under typical VV fluid conditions. The test conditions are pressure=1 MPa, water inlet temperature =20 and 100°C, water velocity=6.7-170 mm/s and heat flux=1.3-24.4 kW/m². On the basis of 361 experimental results, correlations were developed describing heat transfer coefficients in the VV channels. The obtained results make us confident in the acceptable VV cooling performance, which should provide not less than 500 W/(m²K) heat transfer coefficient in the first channel at any location of the VV. Experiments with the VV two-channel model to investigate the flow distribution and its stability in the VV cooling passages are now under way. Study of the development of the natural circulation in a close to full-height (30 m) model of the entire VV cooling circuit is under preparation in order to confirm the cooling system capability of passively cooling the VV in the case of LOFI (Loss of flow incident).

As an alternative to LPT, the residue from which may compromise the vacuum purity, the novel Photothermal Camera method [3] reliably detects cracks within 0.5 mm of the surface with the inspection equipment located up to 2 metres away. A 120 W YAG laser line is raster-scanned between passes in both directions across the welded surface at a speed of 5 mm/sec and the surface temperatures concurrently recorded. By computer analysis and subtraction of the images obtained, defects are discriminated from surface irregularities and changes in reflection. In inspections carried out on weldment surfaces from narrow gap TIG in 60 mm stainless steel (shown in FIG. 6), the results proved that the Photothermal Camera is generally more sensitive and reliable than LPT and better discriminates linear (>1.6 mm) and rounded (>4 mm) indications. The method will need to be qualified as a code case in the future. Further development of advanced methods of cutting, welding and NDT for the VV has been continued in order to increase the potential for improved cost and technical performance [3]. The main achievement of the NdYAG-Laser welding with filler wire, using up to 11 kW of power has been to achieve stable and reliable welding of 60 mm thick SS with the following parameters: 13 passes; welding speed 0.6 m/min; deposition rate 90 g/min; deposited energy 11 kJ/cm.
3. FW/Blanket

3.1 FW/Blanket Design

The basic concept of the FW/blanket system of a modular configuration with a mechanical attachment system embedded in the vessel, has also been refined where it interfaces with the VV. The poloidal key design has been updated to use an inter-modular key (see FIG. 7, 8) fixed on the vessel instead of two prismatic keys. The inter-modular keys have a larger offset (1.2 m) than the prismatic keys (0.95 m) and react the radial torque (~1 MNm) with a lower contact force in the key. The blanket module design with FW panels supported by a central beam has been updated to use a co-axial inlet-outlet cooling connection to simplify the interface with the manifolds and the VV and to facilitate the leak testing of the hydraulic connection (see FIG. 9). The design of the blanket manifolds and its end supports have been developed further to mitigate stress concentrations.

FIG. 7 Inter-modular keys, hydraulic connection and flexible supports
FIG. 8 Blanket modules and manifolds
FIG. 9 Schematic view of co-axial hydraulic connector
3.2 FW/Blanket R&D

Blanket R&D has continued with the manufacture and testing of FW mock-ups and panel prototypes to further improve the engineering margins and to decrease the fabrication cost. It has been confirmed through recent R&D that FW mock-ups with Be tiles withstand a heat flux up to 2.5 MW/m² for 1000 cycles or 0.7 MW/m² for 13,000 cycles [4'], which is adequate for the equivalent design requirement of 0.5 MW/m², 30000 cycles. Four FW panel prototypes are being fabricated to demonstrate the feasibility of HIPing (Hot Isostatic Pressing) or brazing of Be to DS-Cu or CuCrZr. Two of the four FW panel prototypes have been successfully completed (see FIG. 10); one made from DS-Cu Al25 with Be tiles joined onto the Cu alloy by furnace brazing at 780°C, and the second made from CuCrZr alloy with Be tiles joined by low-temperature HIPing at 580°C. Both the Cu alloys were joined onto the stainless steel backing plate by solid HIPing at 1040°C and HIP quenching (i.e. cooling rates above 40 °C/min) was performed on the CuCrZr panel to keep acceptable mechanical properties of the Cu alloy. This challenging technology was applied to the full-scale FW panel prototype for the first time. Fifty six beryllium tiles were joined onto both prototypes, and fatigue testing will be performed in the near future. The other two FW panel prototypes, one with CuAl25 and high temperature HIPed beryllium tiles, the other with powder HIPed CuCrZr and low temperature HIPed Be tiles and one more panel with co-extruded CuCrZr/SS tubes and diffusion bonded beryllium tiles are under construction to investigate alternative fabrication routes. The fabrication method of the Be-armoured FW was developed for the curved region at the top (or bottom) end of the FW panel by manufacturing a mock-up (~300 mm x 160 mm x 95 mm) [4].

It is proposed to use casting to join the CuCrZr heat sink with SS and fast brazing to join Be to CuCrZr as an example of the FW panel manufacturing to reduce the fabrication cost. After TIG welding of steel cooling pipes, vacuum casting of CuCrZr was performed and followed by heat treatment; SA1000°C/WQ/550°C-6h ($\sigma_u$ is about 320 MPa). Fast brazing of Be tiles was carried out in high vacuum at a temperature 700°C. Required heating rate (~1.5 °C/s) was provided by an e-beam facility. To check the quality of the FW mockup (500 mm² x 110 mm x 81 mm³) after casting followed by fast brazing, X-ray inspection was done, as shown in FIG. 11. Thermal fatigue tests on the mockup have been successfully completed up to 5000 cycles at 1 MW/m² followed by 500 cycles at 1.5 MW/m².

Fabrication of full-scale shield blocks was also completed with different cooling channel layouts:
1) a shield block with drilled poloidal cooling channels (1/2 toroidal width, i.e. ~600 mm² x 1100 mm² x 370 mm) and 2) a shield block with drilled radial channels (1/4 toroidal width). For the former, gouging by water jet was applied for making the deep slots. Two quarter blocks were joined by e-beam welding. The fabricated slots satisfied the specified width (< 10 mm) such as at the gouged region 1.5-6 mm and the EB-welded region 7.5-9.5 mm. Finally, a FW panel prototype with a central beam support was fixed onto the shield block with poloidal channels as shown in FIG. 12 [4]. The FW panel prototype (~300 mm² x 1100 mm² x 71 mm³), was fabricated with satisfactory dimensional accuracy such as the slot width < +/- 0.5 mm, the panel width < +/- 1 mm and the step between adjacent fingers < 0.7 mm.

Regarding the module attachment system, the Ti-alloy flexible supports and CuCrZr electrical connectors have been fabricated and tested [5]. The bolting tool used to tighten the tip of the
bolts securing the blanket module to the vessel, including preload measurement methods, has been also developed. The required preload of 800 MPa for the Inconel 718 bolts can only be achieved with an internal heater. Experiments have been carried out on using an automated water-hydraulic bolting tool with integral bolt heater. The mock-up and tooling for the co-axial hydraulic connector, which weld/cuts with new-generation CW YAG lasers, using on-line visual process control, is under construction.

4 Divertor

4.1 Divertor Design

The ITER divertor comprises 54 cassettes onto which are mounted replaceable plasma-facing components (PFCs). The PFCs on each cassette are one inner and one outer vertical target, which intercept the plasma scrape-off layer (SOL), and a private region PFC, which restricts the flow of neutrals into the X-point and protects the gas exhaust pumping duct of the cassette from line-of-sight of the plasma. In the latest design the region beneath the dome of the private region PFC has been opened up to connect the inner and outer divertor channels allowing the free flow of neutrals from inboard to outboard [6]. Carbon-fibre composite (CFC) armour is the preferred armour for the regions of the PFCs where the SOL strikes the targets and the design heat flux 20MWm\(^{-2}\), and tungsten for all other plasma-facing surfaces of the PFCs.

4.2 Divertor R&D

High heat flux (HHF) testing of components with a flat tile configuration of CFC armour continue to fail at 18-20MWm\(^{-2}\), which leads to complete detachment within a few heat cycles. Hence, the CFC monoblock geometry, which has a capability up to ~30MWm\(^{-2}\) and has no cases recorded of tiles falling off the heat sink, is maintained as the preferred option. In contrast for the W armoured surfaces, where the heat flux <5MWm\(^{-2}\), both flat tile and lamellae monoblock geometries are suitable, having sustained up to 27 and 18 MWm\(^{-2}\) respectively.

The manufacture of CFC monoblock and W lamellae armour on the same PFC has been demonstrated on prototypical elements (see FIG. 13) with the armours joined via a cast pure Cu layer to a CuCrZr tube by low temperature HIP. HIP joining of pure Cu to CuCrZr tube at 550°C
produces optimum mechanical properties in the CuCrZr and a grain size < 200μm. Ultrasonic and thermographic inspection of the prototype's joints indicates that they are of good quality and HHF testing is scheduled to begin in September 2002.

Meanwhile the feasibility has been demonstrated of using an annular flow coolant tube (see FIG. 14), which allows a more compact arrangement at the lower end of the target and the reduction of peak heat flux by elongating the target. For the hair-pin return a hemispherical end plug radius 7.5mm gives a pressure drop of 0.1MPa at 10ms⁻¹, or 17% of the total for the ~700mm mock-up. The design has a similar incident critical heat flux (ICHF) to that of a conventional swirl tube. A CFC monoblock amoured mock-up was produced using 15Cu-25Ni-60Ti braze for the CFC/OFHC Cu joint and 52Cu-10Ni-38Mn for the OFHC-Cu/CuCrZr tube joint, both brazed in the same cycle at 980°C for 30 min and gas quenched by Ar at 1°Cs⁻¹ to maintain good mechanical properties in the CuCrZr. Some initial cracks were observed after brazing in the CFC side wall. However, during HHF testing the mock-up survived 1000 cycles of 15s at 20MWm⁻² with no degradation of the thermal performance and no observed growth in the initial cracks.

Efforts have also focused on technologies for building PFCs using the hypervapotron cooling technique, which keeps the heat sink at <500°C during operation avoiding over precipitation of the CuCrZr and hence is suitable for W flat tile armour. The elements are built by casting CuCrZr onto stainless steel. A 12% reduction of the bi-metal plate obtained by rolling, gives a finer grain structure and improved tensile strength. The hypervapotron ribs are formed by machining through the steel into the CuCrZr and all subsequent joints in the heat sink are steel to steel. The W armour tiles are pre-coated with cast pure Cu layer before being brazed using CuInSnNi alloy to the CuCrZr in a fast braze cycle. A 0.6m long mock-up of this design survived 1000 cycles at ~ 20 MWm⁻².

The potential for the entire allowable tritium inventory of ITER to be trapped through co-deposition with sputtered carbon has long been a concern. To mitigate the problem various design options have been proposed ranging from maintaining hot surfaces (~800°C) in the private region to avoid local deposition, to the inclusion of a cold trap upstream of the cryopumps to prevent their contamination. The results of experiments carried out at the Institute of Physical Chemistry, Moscow and at IPP in Germany suggest:
1) due to the high ratio of atomic hydrogen to hydrocarbons, the high density in the ITER divertor and the general surface temperature \( >100^\circ \text{C} \), the deposition of hydrocarbons and the survival of hydrocarbon radicals is negligible. Only stable hydrocarbon species will pass into the pumping duct and to the cryopumps where they will be processed in the normal manner. A cold trap in front of the cryopump is thus not necessary.

2) Surface temperatures within the private region that are greater than \( 500^\circ \text{C} \) can promote cracking of hydrocarbons increasing the amount of active radical species, which is not advisable. The hot radiative liner will therefore be replaced by an actively cooled component.

These laboratory experimental conclusions are very positive for the operation with CFC targets on the ITER divertor. However, experimental investigations continue both in the laboratories and the tokamaks to better understand the very complicated processes between hydrogen and carbon in the extreme conditions of a hot dense plasma to a cryogenic environment where the respective proportions \( \text{H}_2 \), \( \text{H}_3 \) and \( \text{C}_3 \text{H}_2 \) vary dramatically.

5. Conclusions

The ITER vacuum vessel and blanket design and R&D have progressed significantly as collaborative efforts by the International Team, the European, the Japanese, and the Russian Participant Teams. The design and fabrication methods of the VV, FW/blanket and divertor have been assured by the R&D, and additional fabrication methods have also been shown to be acceptable.

References

Design Improvements and R&D Achievements for VV and In-vessel Components Towards ITER Construction


1) ITER International Team, Duitmannstrasse 2, 85748 Garching, Germany
2) EPJU, Reitwegstrasse 2, 85748 Garching, Germany
3) JAERI, Naka Establishment, Naka-machi, Naka-gun, Ibaraki-ken, Japan
4) NTM, Sinter, Ternesy inst., 109551 Metallurgy, St. Petersburg, Russia

IAEA FEC002, 14-19 October 2002
Presented by K. Ishii
ITER International Team
Garching-Joint Work Eff, GERMANY

2.1 Vacuum Vessel Design

Detail design improvements are being pursued to take into account fabrication methods and to provide cost benefits.

2.2 FW/Blanket Design

The design of the FW/Blanket system are being revisited:
- later-model key instead of two pneumatic keys.
- Co-axial inlet outlet cooling connection between the blanket modules and the manifolds
- Blanket manifolds and its end supports to mitigate stress concentricity.

Topics

1. Introduction

Procurement specifications are now being prepared for ITER components whose delivery is on the critical path and required early in construction, such as the vacuum vessel.

2. Design

2.1 VV

2.2 FW/Blanket

2.3 Diverter

3. R&D

3.1 VV

3.2 FW/Blanket

3.3 Diverter

4. Summary
Thermal Hydraulic Tests for the ITER VV (RFTT)

- Measurement of heat transfer coefficients under typical VV fluid conditions.
- The VV test element: a rectangular channel (Fig. 1) 0.2 m wide, 3 m long, 2.48 m heated length and with variable channel height (12.5, 25, 50 mm).
- On the basis of 361 experimental results, correlations were developed.
- Heat transfer coefficient not less than 500 W/(m²K) in the first channel at any location of the VV.

The obtained results give confidence in the acceptable VV cooling performance.

<table>
<thead>
<tr>
<th>Table 1</th>
<th>Test Conditions</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pressure (MPa)</td>
<td>1.0</td>
</tr>
<tr>
<td>Water inlet temperature (°C)</td>
<td>20</td>
</tr>
<tr>
<td>Water velocity (mm/s)</td>
<td>6.7-170</td>
</tr>
<tr>
<td>Heat flux (kW/m²)</td>
<td>1.3-24.4</td>
</tr>
</tbody>
</table>

Step 2 (now under way):
- Experiments with VV two-channel model.
  - To investigate the flow distribution and its stability in cooling passages.
  - Study of the development of the natural circulation in a close to full-height (50 m) model of the entire VV cooling circuit.
  - To confirm the passive cooling capability of the VV in the case of LOFT (loss of flow incident).

Development of Be Plasma Facing Components for FW and Limiter

<table>
<thead>
<tr>
<th>Development of Be Plasma Facing Components for FW and Limiter</th>
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<tbody>
<tr>
<td>HI-Pd Be FW (HRST)</td>
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<tr>
<td>HI-Pd Be FW (HRST)</td>
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<tr>
<td>Be limiter mock-ups</td>
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<tr>
<td>HI-Pd Be FW (HRST)</td>
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<tr>
<td>HI-Pd Be FW (HRST)</td>
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<td>HI-Pd Be FW (HRST)</td>
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</tbody>
</table>

- HI-Pd Be FW (HRST)
- HI-Pd Be FW for neutron moderation (HRST)
- Be limiter mock-ups
- Test specimens of small tiles (500°C, a few microns)
- Curved mock-ups tested: 1,000 cycles at 11.5 MPa, no failure
- HI-Pd and BeC coatings on Be tiles
- BePd and BeCBe alloys spray 1 mm thick, 95% theoretical density
- Be tiles with a thickness of 1 mm, no failure limit - 2.3 MPa
Development of Be Plasma Facing Components for FW and Limiter

- FW Panel Prototype Fabrication R&D - continued -
  - Cu alloys were joined onto the stainless steel backing plate by solid HIPping at 1040°C.
  - HIP quenching (i.e., cooling rates above 40 °C/min) on the CuCrZr panel.
  - This challenging technology was applied to the full-scale FW panel prototypes for the first time.

| Table Fabrication Methods and Main Parameters of 4 FW Panel Prototypes (EUPIT) |
|-----------------|-----------------|-----------------|-----------------|
| Prototype A     | Prototype B     | Prototype C     | Prototype D     |
| Cu-alloy        | Be joining      | Cu-alloy        | Be joining      |
| DCu              | Solid HIP      | CuCrZr          | Solid HIP      |
|                  |                  | DCu             | Powder HIP     |
| Fabrication     |                  |                 |                |
| status           |                  |                 |                |
| Completed        |                  | Not completed   |                |

Dimensions of the panel: 900 mm x 254 mm x 75 mm
Dimensions of the Be tiles: 41.9 mm x 62.5 mm x 10 mm
Weight of the panel: ~140 kg
Number of cooling pipes: 2 x 8 pipes (D30, 300 mm x 300 mm)
Achieved tolerance: Flatness of the front surface: ±0.5 mm (Brass tiles)
All dimensions as specified: ±0.5 mm

Separate FW Panel Fabrication R&D -
- Blanket R&D has continued with the manufacture and testing of FW mockups and panel prototypes to further improve engineering margins and to decrease the fabrication cost.
- It has been confirmed through recent R&D that FW mockups with Be tiles withstand a heat flux up to 2.5 MW/m² for 1000 cycles or 0.7 MW/m² for 13,000 cycles.

FW Panel Prototype Fabrication R&D -
- The R&D is demonstrating the feasibility and performance of the separate FW design.
- Four different full-scale mockups have been fabricated using DCu and CuCrZr, and HIPping and brazing.
- The feasibility of powder HIPping, with particular regard to the tight dimensional tolerances required, is examined in one of the four mockups.
- One more panel with co-extruded CuCrZr/SS tubes and diffusion bonded beryllium tiles

Fabrication of Header Structure for FW panels (EUPIT)

Cast FW Panel Fabrication R&D -
- It is proposed to use casting to join the CuCrZr heat sink, with SS and fast brazing to join Be to CuCrZr as an alternative method of the FW panel manufacturing to reduce the fabrication cost.
- After TIG welding of steel pipes, vacuum casting of CuCrZr was performed and followed by heat treatment: solution-annealing at 1000°C for 40 hours (Cu = 320 MPa).
- Fast brazing of Be tiles was carried out in high vacuum at a temperature 700°C. The required heating rate (~4.5 °C/s) was provided by an e-beam.
- To check the quality of the FW panel mockup (500 mm x 110 mm x 81 mm) after casting followed by fast brazing, X-ray inspection was done.
- Thermal fatigue tests on the mockup have been successfully completed up to 5000 cycles at 1 MW/m² followed by 500 cycles at 1.5 MW/m².

Fig. 1 Cast FW panel mock-up with fast brazed Be and X-ray inspection (RPTT)
Fabrication of Shield Block Prototype and FW panel with Attachment (JAPT)

Fabrication of prototypical shield blocks was completed with different cooling channel layouts:
- 1) a shield block with drilled radial cooling channels (-300 mm x 110 mm x 170 mm)
- 2) a shield block with drilled poloidal cooling channels (-400 mm x 110 mm x 370 mm)

Gouging by water jet was applied to make the deep slots. Two quarter blocks were 6-beam welded.

Finally, a FW panel prototype with a central beam support was fixed onto the shield block.

<table>
<thead>
<tr>
<th>Weight of 1/4 shield block</th>
<th>FW panel prototype:</th>
</tr>
</thead>
<tbody>
<tr>
<td>950 kg/17 kg</td>
<td></td>
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</tbody>
</table>

- Gap width (water jet cutting in shield block): 1.5 ± 0.5 mm
- Gap width (1/4M machine in FW panel): 1.5 ± 0.5 mm
- Gap at EB welding joint between 1/4 shield blocks: 17.5 ± 0.5 mm
- Dimensional accuracy: ± 0.1 mm

---

R&D of Flexible Support

- The flexible supports react the module loads in the radial direction while being compliant in the other directions.
- The flexible cartridges are made from Ti-6Al-4V due to their high strength, low Young's modulus and adequate toughness after irradiation. The material is in annealed conditions, because lead tests have shown adequate strength without precipitation hardening heat treatment.
- Some prototypes of the titanium flexible supports were successfully produced, and mechanical tests including fatigue and buckling have been carried out.
- The measured buckling load was always higher than 1.6 MN, which is ~3 times larger than the disruption design load (500 kN).
- The fatigue test was performed with 1550 K, 1000 cycles and ±1 mm, ±1 mm, ±1000 cycles.
- The test results have demonstrated that these components meet their loading requirements.
- The dynamic loading test in the axial direction has been performed simulating the disruption load (force scale = 27 mm), and it has been confirmed that the dynamic amplification factor is very small.

---

3.3 Divertor R&D

- CFC monoblock geometry has a capability up to ~30MWm⁻³ and has no cases recorded of tiles detaching (the preferred option).
- CFC flat tile geometry is up to ~10MWm⁻³, which leads to complete detachment within a few heat cycles.
- W armoured surfaces, both flat tiles and monoblock monoblock geometries are suitable for the heat flux ~55MWm⁻², having sustained up to 27 and 18 MWm⁻², respectively.

Prototype vertical target (EUPT)
- CFC monoblock and W lamella armour on the same PFC has been demonstrated.
- Cast pure Cu layer to CuCrZr tube by low temperature HIP at 550°C
- Ultrasonic and thermographic inspection of the joints.
- HIP testing in September 2002.

Fabrication and Testing of Electrical Strap

- Punching, cold pressing and bending the copper sheets.
- The most attractive feature is the absence of welds.

Testing of Electrical Strap
- Thermal fatigue test
  - Max. Average temperature increase 120°C / 50°C
  - Tests in a solenoid a toroidal component 7.4 T
  - a poloidal component 1.3 T
  - a radial component 0.3 T

Fabrication of Electrical Strap

- Steel, copper and water/gas cooling were used for the strap.

Fig. 4 Mechanical Fatigue Test for Cyclic Displacements in the Radial Direction (left)
Fig. 5 Solenoid Facility for EM Load Tests on the Electrical Strap (right) (EUPT)
Update on the recent co-deposition laboratory experiments:

Deposition and erosion transition temperature

**IPC Magnetron experiment**

\[ \text{CH:H = 10:1} \]

**Berlin experiment**

\[ \text{CH:H = 1:100} \]

Higher ratio of H\^+ seems to lower the deposition/erosion transition temperature, therefore the 100C ITER minimum temperature should prevent deposition remote from the divertor strike points.

The role of the hot liner: Originally conceived to maintain clean plasma facing surface and to promote recombination of radicals. **BUT:**

Cracking or pyrolysis on a hot surface (>500C) may produce high sticking coefficient CH\_x compounds.

A very important result is shown with the hot liner at 700C.

Main conclusions so far:

- With the high density plasma (1-10Ps) in the ITER divertor, hydrocarbon radicals are not expected to survive.
- The radiative liner should be replace by actively cooled component that have a maximum temperature of 500C.
- Any TiC co-deposits are expected to be very local to the source and will probably be caused by cracking on hot surfaces or secondary plasmas in the private region etc. This is the focus of on going R&D in the tokamaks to verify and explain deposits found near the divertor strike points.
- The 100C VV walls and pumping ducts should remain clean as it is not expected that radicals will be present in the duct.
- All the hydrocarbons that pass to the cryo pumps are expected to be stable and thus no cold trap is required.
- However, the ITER divertor design will isolate the divertor pumping channel by radially sealing the cassette up to the pumping port entrance and provide a removable liner in the pumping duct up to the pump entrance. If after initial operation in H\_2 co-deposition is observed then the complication of a cold trap before the pump can be installed.

4. Conclusion

Although the basic design of the VV and in-vessel components of the ITER design has stayed the same, several detailed design improvements are being pursued in an effort to raise reliability, to improve maintainability, and to save money. R&D activities have been continued to confirm the design validity and to develop alternate cost-saving fabrication methods.
### Summary of JFE/IRFM Blanket R&D

<table>
<thead>
<tr>
<th>R&amp;D Objective</th>
<th>Main R&amp;D and achievements</th>
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<tbody>
<tr>
<td>FW fabrication and testing</td>
<td>- HIP: Pour 1000 cycles at 2.5 MPa, 1200 cycles at 7 MPa, 250 cycles at 15 MPa</td>
</tr>
<tr>
<td>- Be-Cu jointing technology</td>
<td>- Cured by mukago fabricated and tested 1000 cycles at 3 MPa</td>
</tr>
<tr>
<td>- Neutron irradiated Be-Cu joints</td>
<td>- HIP: Pour 6000 cycles at 0.5 MPa</td>
</tr>
<tr>
<td>- Cold-pressing technology</td>
<td>- HIP: HIP: Flux of phase within 45%, phase diffractogram within 15%</td>
</tr>
<tr>
<td>- Non-destructive testing</td>
<td>- LT from outside surface and inside cooling channels, 12 mm resolution</td>
</tr>
</tbody>
</table>

**Fabrication and testing of port linear supports:**
- Be-Cu joint mukago fabricated by HIP with high Be contents |
- HIP: Pour 1000 cycles at 7 MPa |
- 250 cycles at 15 MPa |

**Fabrication and testing of blanket module:**
- Demonstration of mukago assembly of blanket module |
- Development of blanket module with new blank pipe |

**Fabrication and testing of blanket module:**
- Demonstration of mukago assembly of blanket module |
- Development of blank pipe testing with blanket module assembly |

**Fabrication and testing of FW panel:**
- Demonstration of mukago assembly of FW panel |
- Development of blank pipe testing with FW panel assembly |

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**Fabrication and testing of port linear supports:**
- Be-Cu joint mukago fabricated by HIP with high Be contents |
- HIP: Pour 1000 cycles at 7 MPa |
- 250 cycles at 15 MPa |

**Fabrication and testing of blanket module:**
- Demonstration of mukago assembly of blanket module |
- Development of blank pipe testing with blanket module assembly |

**Fabrication and testing of FW panel:**
- Demonstration of mukago assembly of FW panel |
- Development of blank pipe testing with FW panel assembly |
1.29 Development of High Performance Negative Ion Sources and Accelerators for MeV Class Neutral Beam Injectors

M. Taniguchi 1), M. Hanada 1), T. Iga 1), T. Inoue 1), M. Kashiwagi 1), T. Morisita 1), Y. Okumura 1), T. Shimizu 2), T. Takayanagi 3), K. Watanabe 1) and T. Imai 1)

1) Japan Atomic Energy Research Institute, 801-1 Mukoyama, Naka-machi, Naka-gun, Ibaraki-ken, 311-0193 Japan
2) Doshisha Univ., 1-3 Miyako Dani, Tataru, Kyotanabe, Kyoto, 610-0321 Japan
3) Ibaraki Univ., 4-12-1 Nakanarusawa-cho, Hitachi, Ibaraki-ken, 316-0033 Japan

E-mail: tanigucm@fusion.naka.jaeri.go.jp

Abstract. Operation of accelerator at low pressure is an essential requirement to reduce stripping loss of the negative ions, which in turn results in high efficiency of the NB systems. For this purpose, a vacuum insulated beam source (VIBS) has been developed at JAERI, which reduces the gas pressure in the accelerator by enhanced gas conductance through the accelerator. The VIBS achieves the high voltage insulation of 1 MV by immersing the whole structure of accelerator in vacuum with long (~1.8 m) insulation distance. Results of the voltage holding test using a long vacuum gap of 1.8 m indicate that a transition from vacuum discharge to gas discharge occurs at around 0.2 Pa m in the long vacuum gap. So far, the VIBS succeeded in acceleration of 20 mA (H\(^+\)) beam up to 970 keV for 1 s. The high voltage holding capability of the VIBS was drastically improved by installing a new large stress ring, which reduces electric field concentration at the triple junction of the accelerator column. At present the VIBS sustains 1 MV stably for more than 1200 s. Acceleration of ampere class H\(^+\) beams at high current density is to be started soon to demonstrate ITER relevant beam optics.

Operation of negative ion source at low pressure is also essential to reduce the stripping loss. However, it was not so easy to attain high current density H\(^+\) ions at low pressure, since destruction cross section of the negative ions becomes large if the electron temperature is \(>\) 1 eV, in low pressure discharge. Using strong magnetic filter to lower the electron temperature, and putting higher arc discharge power to compensate reduction of plasma density through the filter, an H\(^+\) ion beam of 310 A/m\(^2\) was extracted at very low pressure of 0.1 Pa. This satisfies the ITER requirement of current density at 1/3 of the ITER design pressure (0.3 Pa).

1. Introduction

One of the key components of the NB system is a high power and high energy beam source which can produce 40 A D\(^+\) ion beam at the energy of 1 MeV. JAERI (Japan Atomic Energy Research Institute) has developed the negative ion sources and accelerators to realize efficient and reliable operation of the NB system.

To produce the high energy negative ion beam, the reduction of the gas pressure in the accelerator is one of the most critical issues, since the negative ions are easily neutralized before full acceleration by collisions with residual gas molecules in the accelerator. The neutralization causes not only a loss of negative ions but also acceleration of the stripped electrons. The acceleration of the stripped electrons leads to the reduction of acceleration efficiency and an increase of heat load on the accelerator grids. To overcome this problem, a vacuum insulated beam source (VIBS), which immerses the negative ion source and accelerator in vacuum, has been developed instead of the gas insulated beam source (GIBS). Comparing with the GIBS, the VIBS has larger gas flow conductance around the accelerator that results in significant reduction of the residual gas pressure in the accelerator.

Reduction of the operating pressure of the ion source is also effective to decrease the gas pressure in the accelerator. For this purpose, KAMABOKO source, a prototype of the ITER negative ion source, has been developed to achieve high current density even at low pressure (< 0.3 Pa). Normally electron temperature increases as lower the pressure in arc discharge
plasma, and destruction cross section of the ions becomes large as the electron energy increase to > 1 eV. In the present experiment, the electron temperature was lowered using strong transverse magnetic field, called "magnetic filter", which only allows low temperature electrons diffusing into the extraction region of the ion source. In the present paper, recent activity of JAERI NBI group is presented focusing on the development of VIBS accelerator and the low-pressure high current density source.

2. MeV class Vacuum Insulated Accelerator

A vacuum insulated beam source, as shown in Fig.1, has been developed to demonstrate the 1MV insulation and high current negative ion acceleration. The VIBS consist of five acceleration stages, each acceleration grid is supported and insulated by post spacers made of alumina instead of large insulator columns surrounding the accelerator structure as in the GIBS. Having no surrounding structure, the VIBS allows rapid pumping of residual gases through the accelerator's grids support that results in significant reduction of pressure in the accelerator. Hence, it is expected that the VIBS give lower stripping losses of negative ions in the accelerator than that in the original GIBS. A result of 3-dimensional Monte Carlo gas analysis shows that the stripping loss of the ions is 25% in the VIBS at operating pressure of 0.3 Pa. This value is about a half of the GIBS [1]. Thus the power losses caused by the acceleration of the electron and the neutralization before full acceleration can be reduced from 16 % to 6% by using the VIBS. The VIBS is also promising from the viewpoint of radiation-induced conductivity (RIC)[2,3,4].

As mentioned above, the VIBS has many attractive features compared with conventional GIBS. However, the VIBS for the ITER NB system forms meter-class long gaps in vacuum between 1 MV potential and ground. The pressure in the gap ranges in 0.01 Pa - 0.1 Pa during the NB operation. The previous work [5] has investigated the discharge characteristics under the above conditions and the ITER NB design has been done so that the 1 MV insulation is secured from both Paschen gas discharge [6] and Clump theory [7] for vacuum. In the case of vacuum discharge (Clump theory), the long insulation gap is appreciated, however long gap is not favorable against the Paschen discharge under high pressure. Thus careful design is required for the vacuum insulation of the VIBS. However, there are few experimental reports on vacuum insulation of MV class high voltage in meter-class long gap.

In the present work, a high voltage insulation test for vacuum gaps of 1.8 m was carried out using the accelerator column of VIBS. The flashover voltage as a function of p.d (p; gas pressure, d; gap distance) is shown in Fig.2. The flashover voltage rapidly decreases above
0.2 Pa m, where the transition of discharge mechanism occurred from vacuum discharge to gas discharge. As was designed in ref. [5], it was experimentally confirmed that the ITER NB condition locates left side (lower p.d.) of the transition area as shown in Fig.2, which is free from the gas discharge.

Thus the dominant discharge mechanism in the ITER NB condition could depend on the Clump theory; this means that the larger gap is favorable for 1MV vacuum insulation. Adapting the vacuum insulation technique described above, negative ion acceleration test had been carried out using the prototype VIBS accelerator. Up to now, the highest beam energy of 971 keV was attained with an accelerated beam current of about 20 mA for 1 s.

By using the VIBS, we have succeeded in accelerating the negative ion beam up to 971 keV. However, the negative ion current was still in a low level. This is because voltage holding performance of the VIBS was not stable enough. To sustain 1 MV stably, improvement of the stress ring at the triple junction (interface of metal flange, FRP, and vacuum) was performed for the MeV VIBS accelerator. The results of the electrostatic analysis showed that the newly developed stress ring can reduce the electric field to 1.2 kV/mm at the triple junction from 3.6 kV/mm in the original. The voltage holding performance was tested by the insulation column of the MeV VIBS with and without the new stress ring. The result of voltage holding test for one stage of the accelerator (rated voltage; 200 kV) is shown in Fig.3. With the new stress ring, the flashover voltage reached at rated voltage of 200 kV within the first several minutes whereas the accelerator without ring could not reach 200 kV even after 8 hours of operation. Moreover, with the new stress ring, the highest voltage reached more than 300 kV, where the voltage holding test was stopped to avoid possible damage of the insulator at 1.5 times higher than the rated voltage. It was confirmed that the electric field at the triple junction should be lowered with properly designed stress ring. Thus the MeV prototype VIBS at JAERI sustains 1 MV stably at present. The beam acceleration test of VIBS with newly developed stress ring is now in progress.

3. A High Density Negative Ion Production at Low Pressure

At JAERI, high current negative ion sources of Cesium seeded volume production type has been developed for the ITER NBI application. One of the sources, KAMABOKO source, has already succeeded in demonstrating ITER design current density (300 A/m²) at 0.3 Pa in a short pulse [8]. However, the grid power loading even at 0.3 Pa is close to the design limit (1MW) [9]. Thus, further reduction of operating pressure is desirable for reliable operation throughout the ITER life.

The negative ions have large destruction cross-section by collision with electrons of > 1 eV. However, electron temperature in the volume production type sources tends to increase in arc discharge of the low pressure, and consequently, negative ion current decreases steeply at the pressure lower than 0.3 Pa. To maintain plasma with low electron temperature in the ion extraction region, transverse magnetic field, so called “magnetic filter” is equipped in the negative ion sources of the volume production type. In the present work, enhancement of the
negative ion beam current was attempted by lowering the electron temperature with strong magnetic filter field.

Figure 4 shows the cross-sectional view of the plasma source used in this work. The plasma generator is a multi-cusp semi-cylindrical ion source, called "KAMABOKO" source [8,10]. The source plasma is generated by fast electrons emitted from eight tungsten filaments, and the interior wall of the KAMABOKO chamber serves as an anode. To enhance the production of H⁺ ions, about 1g of cesium (Cs) were seeded in the chamber. The negative hydrogen ion was extracted by applying the extraction voltage of 9 kV between plasma grid (PG) and extraction grid (EXG), and acceleration voltage of 39 kV between EXG and ground grid (GRG). The beam current was measured by a calorimeter, which locates at 1.3 m downstream from GRG.

The magnetic filter of three different strengths were tested; type I: 226 G cm, type II: 605 G cm and type III: 907 G cm, respectively. Figure 5 shows the pressure dependence of electron temperature measured by a Langmuir probe at extraction region (1 cm apart from PG) obtained with constant arc power of 10 kW. It was found that the electron temperature increases with decreasing the source pressure, whereas it decreases with increasing the strength of the magnetic filter. The type III filter maintain the electron temperature of ~2eV, which seems to be effective to enhance the negative ion production, even under the low operating pressure of 0.1 Pa.

The beam extraction test was performed using the type III filter. The results are shown in figure 6. The negative ion current density increases linearly with increasing the applied arc power. A high current density beam of 310 A/m² was extracted at 0.1 Pa. The operating pressure in the negative ion source was 1/3 of the design value for ITER. Although the present result was obtained by operating the source with twice higher input power density (80 kW) than the ITER design value [4], and also in a limited pulse length (0.1 s), the present result

Fig.4 Cross-sectional view of the "KAMABOKO" source.

Fig.5 Pressure dependence of electron temperature in KAMABOKO source

Fig.6 H⁺ current density obtained by using typeIII filter.
gives a prospect to realize a low-pressure high-density source. We plan to demonstrate the operation of the negative ion source for long pulse duration of 1000 s under the conditions of 310 A/m², which are ITER operation conditions, under low pressure at 0.1 Pa.

4. Conclusion

The reduction of gas pressure in the accelerator is one of the key issues to realize the high-energy negative ion beam source. For this purpose, VIBS accelerator and the low operating pressure source have been developed.

The VIBS accelerator has succeeded in accelerating the negative ion beam up to 971 keV for 1 s. The high voltage holding capability of the VIBS was improved by installing a new large stress ring, which reduces electric field concentration at the triple junction of the accelerator column. At present the VIBS sustains 1 MV stably for more than 1200 s.

As for the low operating pressure source, an H⁻ ion beam of 310 A/m² was extracted even at the low pressure of 0.1 Pa by optimizing the filter magnetic field.

These results give the prospect to realize the high power NB systems with high reliability and efficiency.

References

Development of high performance negative ion sources and accelerators for MeV class neutral beam injectors
Naka Fusion Research Establishment, Japan Atomic Energy Research Institute

**Backgrounds**
- In ITER, high power and high energy beam source which can produce 40 A D- ion beam at the energy of 1 MeV is required.
- To produce high power and high energy beams, reduction of gas pressure in an accelerator is essential since the stripped electrons cause reduction of acceleration efficiency and an increase of heat load on the accelerator grids.
  - "KAMABOKO" source at low operating pressure
- In the original design of the ITER beam source, insulation gas was used to sustain 1MW. However, radiation induced conductivity (RIC) causes the power loss of 1MW order.
- The VIBS allows a rapid pumping of residual gases in an accelerator.
  - The vacuum insulated beam source (VIBS)

**KAMABOKO source for low pressure operation**

**Issue:**
In the ITER negative ion source, operating pressure is designed to be 0.3 Pa, where the highest grid power loading is estimated to 1 MW. The grid power loading is marginal from fatigue life aspect and further reduction of operating pressure is desired for reliable operation. However, the negative ion current steeply decreases at the pressure lower than 0.3 Pa due to the destruction of the ions by collisions between fast electrons and negative ions.

![Diagram](image)

**The negative ion destruction reaction:**
\[ H^+ + e^{-} \rightarrow H + e^{-} \]

To suppress this reaction, electron temperature near the extraction region need to be reduced.

- Magnetic filter is effective to keep the electron temperature to be \( \sim 1 \text{eV} \).

**Beam extraction test at low pressure (0.1 Pa)**

The magnetic filter was strengthened to keep the electron temperature to be around 1 eV even under low pressure operation.

- The electron temperature increases with decreasing the gas pressure.
- The type III filter maintain the electron temperature of \( \sim 1 \text{eV} \) even under the low pressure of 0.1 Pa.

The strength of magnetic filter III was sufficient.

In case of type III filter, current density increases linearly with increasing the applied arc power.

A high current density beam of 310 A/m was extracted at 0.1 Pa. This fulfills the current density requirement for ITER at 1/3 of the design pressure (0.3 Pa).

By operating the ITER source at 0.1 Pa, the grid power loading can be reduced to 1/3 of the ITER design (1 MW at 0.3 Pa).
Summary

1) By strengthening the magnetic filter of KAMABOKO source, an H+ ion beam of 310 A/m² was extracted even at the low pressure of 0.1 Pa. This fulfills the current density requirement of ITER at 1/3 of the design pressure (0.3 Pa). By operating ITER source at 0.1 Pa, the grid power loading can be reduced to 1/3 of the ITER design.

2) The VIBS accelerator succeeded in accelerating 971 keV, 20 mA, 1 s beam. High voltage holding capability of the VIBS was improved by installing a new large stress ring. At present the VIBS stably sustains 1 MV for steady state (more than 8500 s). The issue of 1 MV holding in the VIBS acceleration column has been solved and we could have the prospect to achieve high current beam acceleration up to 1 MV.

1 MeV vacuum insulated accelerator

Issues:
- In the ITER NB system, SF₆ insulation gas is not applicable due to excess heat generation by radiation induced conductivity (RIC).
- Vacuum insulation technology for 1 MV in meter order long gap had not been established.

The VIBS accelerator, in which the ion source and accelerator is immersed in vacuum, has been developed.

Vacuum insulated accelerator

The VIBS allows rapid pumping of residual gases through the accelerator grid and supports that result in significant reduction of pressure in the accelerator. At present, the highest beam energy of 671 keV was attained with an acceleration beam current of 20 mA for 1 s.

Voltage holding characteristics

1) Vacuum - glow transition

A high voltage insulation test for different vacuum gas of 0.61 and 3.8 m was carried out using the accelerator column of VIBS to investigate the vacuum - glow transition.

The transition occurred from vacuum discharge to gas discharge at around 0.2 Pa m. This did not depend on the vacuum gap length examined in this work.

Voltage holding test of VIBS accelerator

The voltage holding test of the MeV VIBS accelerator column was performed with the new stress ring.

After the 6 hours conditioning, the MeV VIBS acceleration column sustained 1 MV stably for more than 8500 s.

Any dark current, out gassing, X-ray were not observed during the 1 MV holding.
1.30 Improvement of Beam Performance in Negative-Ion Based NBI System for JT-60U


1) Naka Fusion Research Establishment, Japan Atomic Energy Research Institute, 801-1 Mukouyama, Naka-machi, Naka-gun, Ibaraki-ken, 311-0193 Japan
2) Princeton Plasma Physics Laboratory, POBox 451, Princeton, NJ USA 08543, USA
3) Southwestern Institute of Physics, CNNC POBox 432, Chengdu, Sichuan 610041, China
e-mail contact of main author: umedana@fusion.naka.jaeri.go.jp

abstract. Injection performance of negative-ion based NBI system for JT-60U has been improved by correcting beamlet deflection and improving spatial uniformity of negative ion production. Beamlet deflection at peripheral region of grid segment has been found due to distorted electric field at the bottom of the extractor. This was corrected by modifying the surface geometry at the extractor to form flat electric field. Moreover, beamlet deflection due to beamlet-beamlet repulsion by space charge was also compensated by extending the edge of the bottom extractor. This resulted in reduction of the heat loading on the NBI port limiter. As the result of improvement above, the continuous injection of 2.6 MW H\textsuperscript{+} beam at 355 keV has achieved for 10 s. Thus the long pulse injection up to nominal pulse duration of JT-60U was demonstrated. This has provided a prospect of long pulse operation of negative-ion based NBI system for steady state tokamak reactor. So far, the maximum injection power of 5.8 MW at 400 keV with deuterium beam and 6.2 MW at 381 keV with hydrogen beam have been achieved in the JT-60U N-NBI. Uniformity of negative ion production was improved by tuning filament emission current so as to put more arc power in the region where less negative ion current were extracted.

1. Introduction

The negative-ion based neutral beam injection (N-NBI) system is one of promising candidates for plasma heating and non-inductive current drive of steady state/long pulse operation of tokamak reactors such as ITER. The N-NBI system for JT-60U has been operated since 1996[1] for research of current drive and heating in high density plasma by energetic (500 keV) beam. The design goal of the N-NBI system is to inject 10 MW, 500 keV D\textsuperscript{+} beams for 10 s. Recently the N-NBI has contributed to achieve a fusion triple product of 3.1x10\textsuperscript{20} m\textsuperscript{3} s keV under full non-inductive current drive with NB injection of 5.7 MW at 402 keV [2]. There were some issues remained to achieve injection goal such as less voltage holding of accelerator, excess heat load of grounded grid, beam divergence and spatial non-uniformity of negative ion production. In these issues, beam divergence and non-uniformity of negative ion production have been improved. Until 2000, pulse duration was limited to 2 s with an injection power of 5 MW because of high heat load and subsequent temperature rise of the NB port limiter which is located at about 22 m far from the ion source. In 2001, it was found that beamlets generated from edge of the grid segments were deflected by distorted electric field due to geometric step at the bottom of the extractor. The distorted electric field was compensated by filling the step with metal bar, and consequently, the deflection was corrected. Moreover, beamlets generated from the edge region were focused by protruding the metal bars from the surface of the bottom extractor. Correction of the beamlet deflection and its focusing were effective to reduce excess heat load on the NB port limiter, and enabled us to fire long pulse beam at high injection power.
In these three years, we have also tried to improve uniformity of negative ion production in the ion source [3], by tuning input power of each filament group so as to change arc power distribution in the chamber. Although uniform arc power profile was achieved by the filament tuning, a large non-uniformity in a profile of beams still remained, in particular, at the bottom of the chamber. Then the filaments were again tuned to yield 30% higher arc power dissipation at the bottom region. This was effective some how to improve the uniformity of the extraction beam. In this paper correction of beam deflection, improvement of source plasma uniformity and achievement of long pulse operation are reported.

2. Description of the N-NBI system

Detailed description of the JT-60U N-NBI system can be found else where [4]. Here only outline and key points of the system relevant to the present improvements are described. The N-NBI system has two large negative ion sources mounted on a single beamline. Design value of each ion source is acceleration of 22 A D' ion beam at 500 keV and the rated pulse length is 10 s. The ion source consists of a negative ion generator, an extractor and an accelerator. The negative ion generator is a volume production type multicusp plasma generator with small amount (5 ~ 10 g) of cesium seed to enhance the negative ion production. The cathode of the arc discharge is 48 tungsten filaments. Each of 6 filaments is grouped to compose eight filament groups, and each group is connected to a filament power supply. Thus the arc power to each filament group is independently controllable. The extractor, consisting of a plasma grid and an extraction grid, has an ion extraction area of 45 x 110 cm². The accelerator is an electrostatic three-stage accelerator. Each grid of the extractor and the accelerator is divided into five segments whose size is 45 x 18 cm², and each grid segment has 9 (vertical) x 24 (horizontal) aperture array. Only exception to that is the plasma grid. To inhibit beam extraction from top and bottom edge of the grid, where less ion current are expected due to stray magnetic field from the plasma generator [5], 3 lines and 5 lines of apertures in the top and bottom segments, respectively, were masked with a blank plate made of molybdenum. The grid segments are geometrically inclined (0.5 ° with respect to the next segment) to focus the beam generated from the large grid area to narrow NB port. The heat load on each grounded grid segment was measured calorimetry of the cooling water. During the experiment for improvement, the beams were fired toward a target plate located 3.5 m downstream of the accelerator. By monitoring temperature rise of the target measured by infrared camera, beam footprints were obtained to discuss the beamlet deflection and the uniformity of the beam.

3. Correction of beamlet deflection

The temperature rise of the NBI port limiter whose size is 46 cm height and 50 cm width, has limited the long pulse injection to the JT-60U plasma. It shows that the some components of the beam are intercepted by the limiter. Therefore, it is important to improve the beam optics and focusing. To evaluate the beam optics in detail, the beam profile was measured at 3.5 m downstream from ion source by using a target plate and an IR camera. Figure 1 (a) shows longitudinal beam deposition profile. Two peaks appeared at the both edges of each segment. The beamlets of the segment edge were deflected outward and overlapped with other beamlets. The deflected angle is estimated to be 14 mrad. There were small grooves with 5 mm in depth and 40 mm in width at the down streamside of the extractor grids segments as shown in Fig. 2. These grooves generated electric field distortion. To make the distorted electric field uniform, copper bars were embedded in the grooves. As a result, the peaks in the beam profile changed as shown in Fig.1 (b). The peaks moved from the both edges of the segment boundaries to the center between two segment boundaries. By an estimation of the beam
FIG.1 Longitudinal beam profile at 3.5m from ion source (a) original (b) with flat bar (c) with 1.5mm height bar

trajectory, such peaks do not appear between segment boundaries in the beam profile. It seemed that the beam deflection is enhanced by space charge effect of beamlet-beamlet interaction [6] which was not taken into account in the first modification. The deflection angle was estimated to be 6 mrad outward still. To correct this deflection, the thickness of the copper bar was increased to generate the electric field to steer the beamlets inward to the each segment. The suitable height of the bar was estimated by using a three-dimensional beam trajectory code and then 1.5 mm height was selected. The beam profile was improved as shown in Fig.1(c), which matched to design profile and the beamlet deflection by beamlet-beamlet interaction was corrected completely. Figure 3 shows temperature rise of the beam limiter at the NBI port. The heat loads on the limiter with the flat bar and 1.5 mm extruded bar were decreased to less than 60% and 70% of the original one, respectively. The injection port of N-NBI for JT-60U is so narrow that even slight beam deflection largely affects. This improvement enabled the long pulse injection at high beam power.

4. Uniform negative ion production

FIG.2 Grooves at the bottom of the extractor

FIG.3 Temperature rise of the beam limiter at NBI port
It is thought that non-uniformity of the negative ion production is a critical issue in large-scaled negative ion source so as to suppress grid heat load and breakdown. First filament power was tuned so as to provide uniform arc current from each group of the filaments. Figure 4 shows the heat load on GRG segments as a function of the arc power. GRG1 in Fig.4 corresponds to the top segment and GRG5 to the bottom. The heat load in each segment was normalized by acceleration power of each segment. The heat load of four segments except for the GRG5 were saturated at 8% over the arc power of 150 kW. While the heat load on GRG5 didn’t reached to the same level to those on other segments. Considering that this is due to less negative ion production at the bottom region, more arc power is necessary at the bottom region to extract uniform negative ions. In order to enhance negative ion production at the bottom segment, the arc power at the bottom region was increased, as shown in Fig. 5, by tuning each filament voltage to draw higher arc power at the bottom region. Figure 6 shows longitudinal profiles of target temperature rise obtained with the uniform arc power distribution and with enhanced power at the bottom. The profile shows that higher temperature rise in the center and two bottom segments. Moreover the width of beam in the bottom segment was wider than that of the uniform arc power, suggesting more beams from outermost aperture lines. Consequently, the beam footprint became uniform when the arc power was enhanced at the bottom region, rather than
distributing uniform arc power all over the chamber. Figure 7 shows the ratio of heat load on the segments between bottom and center for the uniform arc power and enhanced at the bottom. The ratio was decreased for both cases with increasing arc power. With the uniform arc power, the ratio changed from 1.6 to 1.4, while for the enhanced power at bottom, it decreased from 1.4 to 1.0. The ratio 1 indicates the same heat loads on bottom segment to that of center. Thus the uniformity of negative ion production was improved by putting higher arc power in the bottom region. The tuning of arc power distribution is an effective method to improve uniformity of negative ion production.

5. Longer pulse beam injection

A long pulse beam injection up to 10 seconds was conducted for the evolution of steady-state operation of the ion source. Figure 8 shows a time evolutions of the ion source parameters such as the acceleration current, arc power, temperature of plasma grid and temperature of grounded grid (GRG) surface. The beam injection started at 3.5 seconds in the chart. Though acceleration current decreases a little until 4.5 seconds after beam initiation with decreasing arc power, thereafter the current reaches steady state. Since plasma grid are not cooled, the temperature increases from 200 degrees to 230 degrees with a time. Negative ion beam current is not affected in the range of this temperature. GRG temperature reached to 300 degree in 10 seconds and then it was saturated. These results indicate capability of the long pulse operation of the ion source with control of the plasma grid temperature.

6. Summary

Injection performance of negative-ion based NBI system for JT-60U has been improved by correcting beamlet deflection and improving spatial uniformity of negative ion production. Beamlet deflection was corrected by adjusting electrical field at first acceleration gap. This resulted 60% in reduction of the heat loading on the NBI port limiter. Uniformity of negative ion production was improved by tuning filament emission current so as to put more arc power in the region where less negative ion current were extracted. As the result of improvements above, the continuous injection of 2.6 MW H\(^+\) beam at 355 keV has achieved for 10 s. Thus the long pulse injection up to nominal pulse duration of JT-60U was demonstrated. This has provided a prospect of long pulse operation of negative-ion based NBI system for steady state tokamak reactor. So far, the maximum injection power of 5.8 MW at 400 keV with deuterium beam and 6.2 MW at 381 keV with hydrogen beam have been achieved.


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Beam divergence
Improvement of N-NBI for JT-60U

Correction of electric field shape of 1st accelerator
Uniformity of negative ion production
Independent control of filament power of 8 groups
Decrease heat load on NBI port limiter
Low negative ion production at the bottom region
Enhanced arc power at the bottom
Improved uniformity of negative ion production

Abstract
Injection performance of negative ion based NBI system for JT-60U has been improved by correcting beamlet deflection and improving uniformity of negative ion production. Beamlet deflection has been corrected by adjusting electric field of first acceleration. As a result, the injection of 2.6 MW at 385 keV has been obtained. At a heat load of the limiter, arc uniformity has been improved by adjusting arc power distribution. As a result, the injection is normal pulse duration time for JT-60U and has provided maximum injection power of 5.8 MW at 400 keV with deuterium beam and 6.2 MW at 381 keV with hydrogen beam.
Negative ion source for JT-60U

Target:
- Beam energy: 500keV
- Beam current: 22A
- Pulse duration: 10 sec
- Beam divergence: 5mrad

Achieved: 17.4A with D-
20.4A with H-

Negative ion source was divided 3 parts.
Generator
Extractor
Accelerator

Structure of Negative ion source

Beam extraction area

The heat load on each Grounded Grid (GRG) segment is measured by calorimetry.

- 24 x 9 apertures in 1 segment
- Each segment is declined 0.5 degrees to next segment for beam focusing
**Original negative ion beam profile**

Longitudinal profile along the white line in IR image

**IR Image**

Peaks

Segment boundary

**Bottom**

Temperature rise [deg]

there are peaks at the both (up/down) edge of each segment

**Beam profile after correcting electric field**

Peaks moved to the segment boundaries.

This beamlet deflection is caused by beamlet-beamlet interaction

**Correction of beam deflection by beamlet-beamlet interaction**

The extrusion height was decided by calculation of 3D beam trajectory.

Beam deflection by space charge effect has been adjusted and beam profile has been improved.
Beam profile at 18m from ion source has been improved

Temperature rise of beam limiter at NBI port has been reduced

Original ion source
Up to 12MJ

With flat bar
Up to 30MJ
Heat load decreased to 40%

With 1.5mm extruded bar
Up to 37MJ
Heat load decreased to 30%

Beam profile is good agreement with simulation

Uniformity of negative ion production

• Improvement of uniformity of negative ion production
  suppress grid heat road and break down.

• Negative ion production at the plasma grid is affected by the source plasma distribution.

• Uniform arc power distribution was obtained by controlling each of eight group filament power.
Heat load on each GRG segment for uniform arc power

- All segments heat load decrease with arc power.
- Heat load on GRG5 is higher than others.
- Negative ion production of bottom region is small.
- Arc power at bottom region was made stronger by 30% by increasing filament power at bottom region.

Heat load on each GRG segment for enhanced arc power at bottom region

- Arc power distribution in the arc chamber.
- Uniform arc power vs. Enhanced at the bottom.

Comparison of beam profile between uniform arc power and enhanced at the bottom

- Uniform arc power vs. Enhanced at the bottom.
- The negative ion production of bottom region has been increased by enhanced arc power.

10 seconds injection has been achieved

- Beam energy: 355keV
- Acceleration current: 24A
- Injection power: 2.6MW
- Injection beam current: 7.3A

- Acceleration current, arc power, temperature of grid surface reach to steady state in 10 sec.
- Only 30 degrees rise of plasma grid is not problem to negative ion production.
- This results indicate capability of the long pulse operation.

*Heat load on the GRG5 reached to same level of others over 200kW arc power.
summary

- The beam deflection has been improved by adjusting the electric field of the 1st accelerator.
- The heat load on the limiter of injection port was reduced to 30% of the original one.
- The arc power enhanced at the bottom area proved to be effective to improve uniformity of negative ion production.
- A 10 second operation at 2.6 MW and 355 keV was attained successfully.
- High power, high energy negative-ion based NBI technology which is one of major technical basis for ITER has been demonstrated.
1.31 Development of Gyrotron and JT-60U EC Heating System for Fusion Reactor

K. SAKAMOTO 1), A. KASUGAI 1), YO. IKEDA 1), K. HAYASHI 1), K. TAKAHASHI 1), K. KAJIWARA 1), S. MORIYAMA 1), M. SEKI 1), T. KARIYA 2), Y. MITSUNAKA 2), M. TSUNEOKA 1), T. FUJII 1) AND T. IMAI 1)

1) Naka Fusion Research Establishment, JAERI, Naka-machi, Ibaraki-ken 311-01 Japan
2) Display Devices & Components Company, Toshiba Co., Ootawara-shi, Tochigi, 324-8550 Japan

e-mail: sakamotk@naka.jaeri.go.jp

Abstract. The progress of ECH technology, for ITER and JT-60U tokamak, are presented. In the development of gyrotron, 0.9MW/9.2sec, 0.5MW/30sec, 0.3MW/60sec, etc. have been demonstrated at 170GHz. At 110GHz, 1.3MW/1.2sec, 1.2MW/4.1sec, 1MW/5sec were obtained. It is found that the reduction of the stray radiation and the enhancement of cooling capability are keys for CW operation. Four 110GHz gyrotrons are under operation in the ECH system of JT-60U. The power up to approximately 3MW/2.7sec was injected into the plasma through the poloidally movable mirrors, and contributed to the electron heating up to 26keV(n_e=0.5x10^{19}cm^{-3}), and the suppression of the neo-classical tearing mode.

1. Introduction

The Electron Cyclotron (EC) wave is an effective method of on- and off-axis current drive and plasma profile control for fusion reactors. Injection of 20 MW EC power is planned in the ITER design to suppress the Neo-Classical Tearing Modes (NTMs). A 170GHz, 1 MW gyrotron is a key R&D technology of the ITER EC system and intensive efforts have been made in JAERI to develop the 170GHz gyrotron. During ITER EDA, successes of introducing the depressed collector[1], high order mode oscillation at 170GHz/1MW[2] and installation of a CVD diamond window on the gyrotron[3-5] opened a new stage of gyrotron development. Further advancement to suppress unnecessary modes like parasitic oscillations was achieved[6]. Together with these, integration efforts of the key EC technologies have also been devoted in the 110GHz EC system on JT-60U. The construction had been started in 1998 by use of the technology outcomes of ITER R&D and the system with four 1MW gyrotrons started its operation in 2001[7]. In the following sections, the latest results of development of gyrotrons and ECH system on JT-60U are described.

2. Gyrotron Development

2.1 170GHz gyrotron

A photograph of the 170GHz gyrotron and a cross sectional view of the gyrotron are shown in Fig.1 and Fig.2, respectively. A triode type electron gun (magnetron injection gun: MIG) and mirror magnetic field makes a rotating electrons with an energy of 70keV~ 85keV, which are injected into a cylindrical cavity and generate TE_{01,1} mode RF of more than 1MW. Q-factor of the cavity is 1530. The oscillation power is converted to the Gaussian beam using a quasi-optical mode converter and outputted through the low loss diamond window. The design value of the radiation power from the window is 94%. A gyrotron experiment was carried out on a gyrotron test stand (RFTS). The capability of RFTS is 90kV/50A for long pulse operation, and
Fig. 2: Cross sectional view of depressed collector gyrotron.

Fig. 4: SiC cylinder at beam tunnel for suppression of parasitic oscillation.

Fig. 3: Gyrotron and matching optics unit (MOU) for coupling with waveguide. Measured wave patterns at the gyrotron window and at the waveguide mouth are shown.

Fig. 5: Beam current dependence of Power and efficiency at 170 GHz. Beam voltage is ~74.5 kV.

90kV/80A up to a few milliseconds. A power supply is consisted of main power supply (MPS) and beam acceleration power supply (APS). The voltage difference between MPS and APS appears as a retarding potential on the spent electron beam at a ceramic insulator (DC break) for a depressed collector operation. A switching of the MPS is done by IGBT (Insulated Gate Bipolar Transistor). The output power is focused to the corrugated waveguide of 31.75mm in diameter using two phase-correction mirrors in the matching optics unit (MOU). As the rf power is radiated as a Gaussian beam through the diamond window, high efficiency coupling is expected. The MOU and transmission line should be evacuated to avoid the breakdown. In the experiment, the Gaussian beam was formed at the window as shown in Fig. 3. The transmitted power to the dummy load placed after 12m corrugated waveguide via three miter bends was 95% of the gyrotron power. In the previous 170GHz gyrotron, oscillation efficiency was ~24% at most because a parasitic oscillation occurred in the beam tunnel (a region between the electron gun and the cavity). The frequency of the parasitic oscillation was ~140GHz, which indicates the beam instability due to the cyclotron resonance. The power of the parasitic oscillation was sometimes in the order of a few tens kW, which causes an electron energy broadening and consequently decrease of the main oscillation efficiency. To suppress the parasitic oscillation, SiC cylinders were installed on the surface of the beam tunnel as shown in Fig. 4. Since the SiC is a good mm wave absorber, the growth of the instability is anticipated to be suppressed. The adopted SiC cylinders have a finite resistivity, no electrification would occur. As a result, the parasitic oscillation power was suppressed and the output of 1.3 MW with
31% of oscillation efficiency was achieved. In Fig.5, a beam current dependence of the output power and of efficiency (without depressed collector) is shown. The beam voltage is 74.5kV. The quasi-CW operations have been demonstrated as listed in table 1. Figure 6 is typical waveforms of 47sec operation at 0.45MW output. It is noteworthy that the temperatures of major components like cavity, window and collector stabilized within 5s and operation was quite stable. It took almost no conditioning time to extend the pulse duration from 10 s to 30 s at 500 kW, which gives a promising view to the ITER CW gyrotron. A key point of this fast conditioning is an active extraction of stray radiation power inside the gyrotron through the DC break ceramic and a sub-window. The aperture of the sub-window is 120mm. A material of the DC break ceramic and sub-window is silicon nitride (SN287, Kyocera Co., tanδ~2x10⁻⁴). The stray radiation was absorbed by water in the Teflon tube and the FX-3300 (3M), which flew around the DC break. The total power of the stray radiation that was extract through the DC break and the sub-window is ~8% of the output power. A power deposition of the stray radiation to the SiC cylinder at the beam tunnel was 0.35% of the output power (mainly to upper one). The pulse extension was prevented by pressure increase in the gyrotron. In Fig.7, the typical behavior of the pressure is shown. Basically, pressure is kept in very low level, but sudden increase of the pressure occurs. The cause was confirmed that the temperature increase of the component of poor cooling (bellows behind the steering mirror that is made of SUS), which absorbed the stray radiation, and probably exceeded a baking temperature 450°C. Further extension of the pulse duration will be achieved by enhancing the cooling capability of the minor sub-components.

2.2 110GHz gyrotron

The 110GHz gyrotron has a same appearance with 170GHz. The oscillation mode is TE_{22,6}. Q-value of the cavity is 1300. The heat load on the cavity wall is 1.2kW/cm² at 1MW operation, that is well below the criteria of the heat load of 2kW/cm². The MIG is triode, oscillation power of TE_{22,6} mode is converted to Gaussian beam using the quasi-optical mode converter. The thickness of the diamond disk is 1.715mm. At the beam tunnel, SiC cylinder that has a same size with 170GHz gyrotron is installed. The gyrotron experiment was carried out RFTS using the same magnet. Fig.8 is a beam current dependence of the output power and efficiency at 1msec operation. The beam voltage is 84.5kV. At I_{b}=57A, the output power was 1.56MW. The maximum efficiency was 33% (1.3MW) at I_{b}=47A. As with the 170GHz experiment, 94% of the output power was transmitted to the dummy load after three miter bends using a same transmission line. Long pulse operation is underway. Up to now, 1.3MW/1.2sec (efficiency of 48% with depressed collector), 1.2MW/4.1sec 1.0MW/5sec were obtained (Table 1). Since the heat load on the cavity wall has enough margins at 1MW, long pulse operation with higher power such as 1.5MW could be possible.
3. JT-60U EC System

The EC system in JT-60U has similar configuration to the ITER except a launcher. The RF power from the gyrotron is transmitted through evacuated corrugated waveguide and the CVD diamond torus window to the launcher. The RF beam is steered by movable mirrors and injected into plasma. The key points of EC system technology are 1) gyrotrons; 2) high efficiency and high power transmission; 3) fast and accurate scanning of RF beam direction to the plasma. On the first point, the performance of the gyrotron was proved as described in the previous section. However, when the gyrotron is operated in the EC system, where the several gyrotrons and other apparatus are simultaneously operated, some unexpected effects arise. As the power supply is converted from that of lower hybrid system, the voltage of the main power supply $V_{main}$ is not regulated. The stabilized voltage of the accelerating DC Power Supply (APS), which determines the beam energy of the gyrotron $V_e$, guarantees the gyrotron operation itself. The voltage perturbation of the MPS appears on the retarding potential for the depressed collector $V_{dp}$. Here, $V_{dp}=V_e-V_{main}$. When $eV_{dp}$ deeply exceeded the minimum energy of the spent beam, electron trapping becomes a problem. The electron trapping cause an increase of a leak current to the anode of MIG, which seems to be a cause of the trip of the power supply. In particular, it was frequent when the four gyrotrons start at same timing by a mutual coupling of the voltage perturbation. The influence of the voltage perturbation could be reduced by setting a ramp-up of the electron beam slow and $V_{dp}$ lower. As a result, the simultaneous operation of four gyrotrons has been

<table>
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<tr>
<th>Frequency (GHz)</th>
<th>Power (kW)</th>
<th>Duration (sec)</th>
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<tbody>
<tr>
<td>170</td>
<td>900</td>
<td>9.2</td>
</tr>
<tr>
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<td>17.</td>
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<tr>
<td>200</td>
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<tr>
<td>110</td>
<td>1300</td>
<td>1.2</td>
</tr>
<tr>
<td>1200</td>
<td>4.1</td>
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<td>1000</td>
<td>5.</td>
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![Fig.8: Beam current dependence of power and efficiency at 110GHz. Beam voltage is 84.5kV. Pulse duration is 1msec.](image)

![Fig.9: Real time control of electron temperature using antennas.](image)
succeeded. For an EC system of many gyrotrons, stabilized main power supply voltage is desired to obtain a stable and high efficiency operation. Present issue for simultaneous operation is a trip of power supply caused by a noise from other apparatus.

On the second point, the coupling of the gyrotron output power with the waveguide was optimized by the careful adjustment of two mirrors in MOU, which is important for excitation of HE_{11} mode. And periodic sag of the waveguide has been minimized using the laser beam alignment technique. Present value of the transmission efficiency is \( \sim 80\% \). The transmission line is composed of corrugated waveguide of 31.75 mm in diameter, 7 miter bends and pair of polarizers are included. Total length is \( \sim 60\text{m} \). The pressure inside the transmission line is kept in vacuum \( \sim 1\times10^{-7}\text{ Pa} \), and a breakdown was not observed. The torus CVD diamond windows of 31.75 mm and 60.5 mm in aperture have shown a capability of stable 1 MW transmission. The performance of the launching system has been studied with two antennas for the last point. The dynamic beam steering capability was confirmed as shown in Fig. 9. Two antennas control the RF beam injection independently in a poloidal direction and the change of electron heating in the center is observed. The maximum central heating is obtained in case of the on-axis heating of antenna A. The maximum injection performance of approximately 3 MW for 2.7 s was achieved. As a result, the EC system contributed to realize high electron temperature plasma of 26 keV [8], a suppression of neoclassical tearing mode [9] and current drive [10].

4. Concluding Remarks

The recent progresses of ECH technology development in JAERI are presented. The development of high power 170 GHz and 110 GHz gyrotrons has attained remarkable progress for ITER and JT-60U, respectively. At 170 GHz, power outputs of 0.9 MW/9.2 sec, 0.75 MW/17 sec, 0.55 MW/30 sec, 0.45 MW/47 sec, 0.35 MW/60 sec, 0.2 MW/133 sec, etc have been demonstrated. And at 110 GHz, 1.2 MW/4.1 sec, 1 MW/5 sec were obtained. These powers and pulse durations are limited by the temperature increase of the inner components, which was caused by the stray radiation from the built-in mode converter. The reduction of the stray radiation and the enhancement of cooling capability are essential for CW operation.

Using four 110 GHz gyrotrons, the EC system is under operation on JT-60U. High power operation of the gyrotrons with simultaneous operation and high efficiency transmission enabled a 3 MW/2.7 sec injection. By the power deposition control using movable mirrors, the EC system contributed to the electron temperature of 26 keV (\( n_e \sim 0.5 \times 10^{19} \text{ cm}^{-3} \)), and the suppression of the neo-classical tearing mode. These results give a prospect for the ITER EC system.

References

Remote Steering Launcher

- Advanced launcher -

Theory
Propagation const of wave (m,n) mode in a square corrugated waveguide (swg)

$$ k_{swg} = \frac{k}{\sqrt{1 + (m^2 + n^2)/2}} $$

- $k_{swg}(m^2 + n^2)/2$

$\Delta$ Phase in (m,n) & (m+1,n) mode

$\Delta = 2\pi(m+n)$

Best wave length $L_{swg} = 8.75$

$HE_{22}$ modes are excited if $\theta$ is small angle. When they propagate over $L_{swg}$, RS of RF beam is achieved $R^2$.

High Power Test
No trouble was confirmed up to 0.5MW/2sec.

Next experiment
Radiation tests with water bends.

-326-
1.32 Studies of ELM Heat Load, SOL Flow and Carbon Erosion from Existing Tokamak Experiments, and Projections for ITER


1) Japan Atomic Energy Research Institute, Naka-machi, Naka-gun, Ibaraki 311-0193, Japan
2) EFDA CSU, Max-Planck-Institut für Plasmaphysik, D-85748 Garching bei München, Germany
3) Lawrence Livermore National Laboratory P.O.Box 808, Livermore, CA 94550, USA
4) Forschungszentrum Jülich, IPP, EURATOM-Association, D-52425 Jülich, Germany
5) MIT Plasma Science and Fusion Center, Cambridge, MA 02139, USA
6) Max-Planck-Institut für Plasmaphysik, D-85748 Garching bei München, Germany
7) Joint European Torus, Abingdon OX13 3EA, Oxon, United Kingdom
8) ITER International Team, Max-Planck-Institut für Plasmaphysik, D-85748 Garching bei München, Germany
9) General Atomics, P.O.Box 85606, San Diego, CA 92186-5608, USA
10) University of California at San Diego, San Diego, CA 92093, USA

e-mail: asakuran@fusion.naka.jaeri.go.jp

Abstract. Three important physics issues for the ITER divertor design and operation are summarized based on the experimental and numerical work from multi-machine database (JET, JT-60U, ASDEX Upgrade, DIII-D, Alcator C-Mod and TEXTOR). (i) The energy load associated with Type-I ELMs is of great concern for the lifetime of the ITER divertor target. In order to understand the physics base of the scaling models[1], the ELM heat and particle transport from the edge pedestal to the divertor is investigated. Convective transport during ELMs plays an important role in heat transport to the divertor. (ii) Determination of the SOL flow pattern and the driving mechanism has progressed experimentally and numerically. Influences of the drift effects on the SOL and divertor plasma transport were discussed. (iii) Carbon erosion and redeposition are of great importance in particular for tritium retention via codeposition. Characteristics of chemical yield at two different deposited carbon surfaces, i.e. erosion- and redeposition-dominated areas, have been studied. Progress in understanding of the chemical erosion is reviewed.

1. Introduction

Three important physics issues for the ITER divertor design and operation are summarized based on the experimental and numerical work from multi-machine database (JET, JT-60U, ASDEX Upgrade, DIII-D, Alcator C-Mod and TEXTOR).

(i) The energy load associated with Type-I ELMs is of great concern for the lifetime of the ITER divertor target. Recently, scaling studies of the normalized ELM energy loss ($A_{ELM}/W_{ped}$) depending on plasma parameters, such as the effective electron collisionality of the pedestal plasma ($\nu^*_{ped}$) and pedestal density fraction ($n_{e,ped}/n_{GW}$), have progressed[1]. In order to understand the physics base of the scaling models, the ELM heat and particle transport from the edge pedestal to the divertor is investigated [2]. Convective transport during ELMs plays an important role in parallel heat transport and deposition to the
divertor as well as energy loss from the edge in high-density ELMy H-mode plasmas.

(ii) Control of the divertor plasma and impurity ions is strongly influenced by parallel SOL flow. Determination of the SOL flow pattern and the driving mechanism has progressed experimentally using the Mach probe measurements [3], and numerically using the SOL and divertor simulation code (UEDGE) with including drift effects [4]. Particle flux towards the divertor is influenced by the plasma drifts. Consequences for the divertor plasma characteristics in ITER are discussed.

(iii) Carbon erosion and redeposition are of great importance, in particular, for tritium retention via codeposition, which is expected to occur during the initial ITER operation with carbon target. Characteristics of chemical yield at two different deposited carbon surfaces, i.e. erosion- and redeposition-dominated areas, have been studied [5]. Progress of quantitative understanding of the chemical erosion is reviewed.

2. Studies of SOL Transport and Heat Load Due to Type-I ELM Energy Loss

ELM heat load is determined by plasma transport from the edge pedestal to the divertor as well as ELM energy loss. The role of parallel electron and ion energy transport has been investigated by heat flux measurements with infrared cameras and soft-X-ray emission from hot electrons impinging on the divertor (indicating conductive transport). Electron power pulses evaluated from the soft-X-ray emission in JET have similar duration to edge collapse time by MHD (τMHD), and it decreases with increasing ne,ped, which shows that the proportion of energy carried by hot electrons decreases[6]. Therefore, hot electrons play a decreasing role in determining the divertor ELM energy flux with decreasing Te,ped and increasing ne,ped.

Energy deposition time from the IR measurements (τIR ELM) in JET, AUG and JT-60U is well correlated with the characteristic time for ion transport from the pedestal to the divertor, \( \tau_{i,\text{Front}} = 2\pi R q s / C_s,\text{ped} \) where \( C_s,\text{ped} \) is the ion sonic speed calculated from plasma pedestal temperatures, while \( \tau_{MHD} \) remains unchanged (~250 µs). The good correlation of \( \tau_{IR}^{\text{ELM}}[\mu s] = 0.29(\tau_{i,\text{Front}}^{\text{MHD}}[\mu s])^{1.38} \) suggests that convective transport is important for ELM heat deposition to the divertor. The reason of \( \tau_{IR}^{\text{ELM}} \) larger than \( \tau_{i,\text{Front}}^{\text{MHD}} \) is probably that the averaged \( T_e \) of the plasma particles expelled with the ELM is lower than \( T_e,\text{ped} \) which is used to calculate \( \tau_{i,\text{Front}}^{\text{MHD}} \).

In JT-60U, \( \tau_{IR} (250-350 \mu s) \) is comparable to the duration of ELM-enhanced ion flux and the SOL flow velocity (i.e. Mach number increases to ion sonic level) measured with Mach probe just below the X-point [7]. The convective heat flux, \( \gamma_F [5/2 k T_e + 5/2 k T_i + 1/2 m(2 C_s)^2] \sin \theta_{div} \) where \( \gamma_F \) and \( \theta_{div} \) are the ion flux and the field line pitch angle on divertor plate, reaches up to
70-80% of the total heat flux measured by IRTV, provided that the ELM-enhanced particle flux has $T_e$, $T_i$, and $C_i$ measured at the pedestal. This is in agreement with the good correlation found between $\tau_{\text{ELM}}$ and $\tau_{\text{Front}}$. The fraction of convective heat flux would be comparable to or smaller than conductive heat flux, if $T_e$, $T_i$, and $C_i$ for the averaged exhausted plasma are used.

Above progress in the SOL transport study provides a good physics basis on which to extrapolate present experimental results of Type-1 ELM energy losses to ITER. An interpretation of $\Delta W_{\text{ELM}}/W_{\text{ped}}$ database as a function of $v^*_{\text{ped}}$ (based on MHD instability models produced by edge bootstrap current) suggested unacceptable value of $\Delta W_{\text{ELM}}/W_{\text{ped}} = 0.15-0.2$ for ITER ($v^*_{\text{ped}} = 0.03$). On the other hand, $\Delta W_{\text{ELM}}/W_{\text{ped}}$ is small (0.1-0.15[8] and 0.05-0.1[2]) for the models based on the SOL convective transport, which would be allowable for ITER divertor lifetime for an inclined target option [9]. Determination of conductive and convective processes in the SOL is crucial for quantitative evaluation of the ELM energy loss.

3. Plasma Flow and Effects of the Plasma Drifts in SOL and Divertor

Poloidal variation of the parallel SOL flow, i.e. at high-field-side (HFS), low-field-side (LFS), plasma top SOLs and the private flux region, were determined using reciprocating Mach probes (JT-60U, JET, Alcator C-mod, DIII-D and AUG), which demonstrated a consistent picture: net particle fluxes to the LFS and HFS divertors, but the SOL flow from the LFS SOL to the HFS divertor through the plasma top for the ion VB drift direction towards the divertor. At the same time, in the private flux region, drift flow to the HFS divertor is produced by large $E\times B$ drift [10]. Particle fluxes towards the HFS divertor are experimentally investigated with increasing $\bar{n}_e$ in JT-60U[3] as shown in Fig.2, where $n^0_{\text{HFS}} = 5.2 \times 10^{19} \text{m}^{-3}$. Drift flux ($\Gamma_{p,\text{drift}}^{HFS}$) is away from the HFS divertor, and its fraction decreases from 50% to 10% with increasing $\bar{n}_e$.

Thus, net particle flux ($\Gamma_{p}^{HFS}$) is always towards the HFS divertor. On the other hand, the drift flux in the private region ($\Gamma_{p}^{\text{Prv}}$) is comparable to or larger than $\Gamma_{p}^{HFS}$ under the attached divertor condition. HFS-enhanced asymmetry in divertor ion flux is produced mainly by $\Gamma_{p}^{\text{Prv}}$.

Various mechanisms producing the parallel SOL plasma flow such as $E\times B$, $B\times V_B$ and diamagnetic drifts were recently discussed. The numerical approach to understand the SOL flow pattern has progressed using the UEDGE code with the drift effects included [11]. Results for a JT-60U L-mode case [4] show that the SOL flow is produced from LFS midplane to HFS divertor, which is caused mainly by ion $B\times V_B$ drift. Mach numbers of the SOL flow are 0.1-0.2 at LFS and increase from 0.2 to 0.6 at HFS with $\bar{n}_e$ (as shown in Fig.3), which are smaller than and comparable to the measurements at $\bar{n}_e=1.5 \times 10^{19} \text{m}^{-3}$ (0.3-0.4 and ~0.4), respectively. Simulation of H-mode plasmas suggests that the effect of drifts becomes more important in H-
mode as the radial temperature gradient scale length is smaller, and that the existence of large parallel pressure gradient drives fast flow. Simulation also suggests large carbon ion flow in the private flux region similar to the plasma drift flow, which may lead to accumulation of carbon on HFS divertor plate.

Although high density core plasma \((\bar{n}_e/n_{GW}^{SO} \sim 0.85)\) is sustained in ITER, electron collisionality of the SOL plasma is relatively low \((\nu_{e,SOL} = 10^{-5-10})\) since \(T_e\) at separatrix \((\sim 150 \text{ eV})\) would be high where \(n_e \sim 3.5 \times 10^{19} \text{ m}^3\). Relatively large \(E_x\) is expected in such low \(\nu_{e,SOL}\), which corresponds to database at \(\bar{n}_e/n_{GW}^{SO} \sim 0.4\) in Fig. 2, and \(\Gamma_{p,drift}^{HFS}/\Gamma_{p}^{HFS}\) of \(\sim 30\%\) would be anticipated. At the same time, the \(E_xB\) drift flow in the private flux region is expected just below the X-point since the detachment is localized near the strike-points. Particle flux towards the divertor would be influenced by these drifts, and a design work including the drift effects will be useful to optimize the divertor and pump geometries.

4. Chemical Erosion Under Fusion Relevant Condition

Over the last two years many tokamaks and laboratory experiments have been carried out to better characterize the chemical erosion yield of carbon for ion flux density of \(10^{21}-10^{23} \text{ m}^{-2} \text{s}^{-1}\) and low impact energies. However, there are several aspects of erosion/redeposition that are uncertain and require further investigation in current machines in order to extrapolate reliably to ITER. This is mainly due to uncertainties in the photon efficiencies of the hydrocarbon radicals. Another complication arises from differences in chemical erosion yields between areas of net erosion and net re-deposition; the structure of the re-deposited carbon films (containing hydrogen isotope) also depends on the energies of the incident ions[5].

On the re-deposition dominated areas, carbon can be deposited in the form of soft layers, and re-erosion can proceed due to atomic and low energy ion fluxes. This is observed for example in the JET HFS divertor, which is routinely detached and exposed to high density and low energy hydrogen atoms and molecules. In this process, contribution of higher hydrocarbons to the total sputtered carbon atoms was found to dominate (i.e. half of the total methane). Large contribution \((\sim 80\%)\) from higher hydrocarbons was also found in simultaneous CD and C2 measurements of JT-60U LFS divertor [12] (where net erosion dominated) at surface temperature, \(T_{surf}\) of 440 and 560K. Overall chemical yields are between 7 and 20% for the two tokamaks, and significant influence of \(T_{surf}\) on the chemical yields was observed. Thus, the contributions of higher hydrocarbons and generation conditions need more attention in the future, under the high flux density.

On the net-erosion dominated areas (mostly in the LFS divertor of most devices), overall trends indicate that chemical yields between 3-5 % are a good guess. There are weak indications of flux
dependence for JET[13] and TEXTOR(limiters)[14] up to flux density of 5-10x10^22 m^2s^-1 as shown in Fig.4. For those conditions, most data show a week dependence on T_suf (in 400-600K). For JT-60U LFS divertor, only methane-origined chemical yields are plotted due to large contribution from higher hydrocarbons. These data show slight decrease of the yields with fluxes (\Gamma^\alpha, \alpha=0.1-0.4) and T_suf dependence is larger than other tokamaks. Yields for ASDEX-U decreased to ~1% as \Gamma^\alpha, \alpha=0.7 [15], and very low chemical yields (lower than 0.3-0.5%) was reported in DIII-D long-exposed tiles [16].

To date no definitive conclusion can be drawn on carbon chemical erosion yields at high flux density in ITER (several 10^23 m^2s^-1). More experiments are planned to better determine the extent to which parameters other than flux (e.g. energy, re-deposition, photon efficiency and viewing geometry in spectroscopic measurements, yields of higher hydrocarbons etc.) affect the observed erosion rates in current machines.

5. Conclusions

Understanding of important physics issues for the ITER divertor design and operation, (i) divertor heat load associated with Type-I ELMs, (ii) parallel SOL plasma transport produced by the SOL flow and drift flow, have been progressed based on the experimental and numerical work from multi-machine database. (iii) There has been some progress in the characterization of chemical erosion yields in C-clad divertor tokamaks. In order to make reliable extrapolation for ITER, the R&D program needs to better address the physics of the erosion mechanisms and the transport and redeposition of eroded material and resulting mixing effects.

References
Studies of ELM Heat Load, SOL Flow and Carbon Erosion from Existing Tokamak Experiments, and Projections for ITER


19th IAEA Fusion Energy Conference
14 - 19 Oct. 2002, Lyon, France

1Japan Atomic Energy Research Institute, Naka-machi, Naka-gun, Ibaraki 311-0193, Japan
2EIDA CSU, Max-Planck-Institut für Plasmaphysik, D-85748 Garching, Germany
3MIT Plasma Science and Fusion Center, Cambridge, MA 02139, USA
4Lawrence Livermore National Laboratory P.O.Box 808, Livermore, CA 94550, USA
5Forschungszentrum Jülich, IPP, EURATOM-Association, D-52425 Jülich, Germany
6Max-Planck-Institut für Plasmaphysik, D-85748 Garching, Germany
7Joint European Torus, Abingdon OX13 9EA, Oxford, United Kingdom
8ITER International Team, Max-Planck-Institut für Plasmaphysik, D-85748 Garching, Germany
9General Atomics, P.O.Box 6966, San Diego, CA 92186-6966, USA
10University of California at San Diego, San Diego, CA 92093, USA

1. Introduction

Three important physics issues for the ITER divertor design and operation are summarized, based on the experimental and numerical work from multi-machine database:

- JET, JT-60U, ASDEX Upgrade, DIII-D, Alcator C-Mod and TEXTOR.

(i) ELM Heat Transport from Edge to the Divertor

Scaling studies of ELM energy loss (\(\Delta W_{\text{ELM}}/W_{\text{p忠诚}}\)) depending on effective collisionality of the pedestal plasma (\(\nu_{\text{ped}}\)), density fraction (\(n_{\text{ped}}/n_{\text{tot}}\)), and characteristic time for ion transport (\(t_{\text{ion}}\)) have progressed [1, 2].

Heat transport in SOL and divertor is discussed to understand the physics base of the scaling models.


(ii) SOL Flow Pattern and Drift Effects

Control of the divertor plasma and impurity ions is strongly influenced by SOL flow. Numerical [3] and Experimental [4] studies of the SOL flow pattern and the driving mechanism are summarized.


(iii) Chemical Yields of Carbon

Carbon erosion and redeposition are important for tritium retention via codeposition (initial ITER operation).

Characteristics of chemical yields at erosion- and redeposition-dominated carbon surfaces have been studied [5].


2. Divertor operation limit

- Allowable ELM size in next-step device is determined by Materials

\[ \Delta T_{\text{ELM}} < \text{Physical Limit (sublimation, melt)} \]

- Heat load allowable for ITER Divertor lifetime (>10^3)

\( \rightarrow \)

(for an inclined target option)

based on Multi-machine database


- Experimental characterisation + Physics model have been progressed in ITER expert - ITPA works

- Multi-machine comparisons help understanding ELM physics, which is required to extrapolate ELM energy and particle losses to BPSs.


- Recent progress is presented: LOARTE, A., et al., "ELM energy and particle losses and their extrapolation to burning plasma experiments", PSI 2002
2.1. ELM Energy Loss (Convective ELM)
Type-I ELMs with small conductive losses: DIII-D & JET
Conductive losses decrease with increasing $n_{\text{pred}}$ (decreasing $T_{\text{pred}}$, increasing $v_{\text{p}}$)

- Reduction of $\Delta W_{\text{ELM}}/W_{\text{pred}}$ at high $n_{\text{pred}}$ due to decreasing $T_{\text{pred}}$, $T_{\text{p}}$, not $\Delta n_{\text{pred}}$/$n_{\text{pred}}$.
  where, $W_{\text{pred}} = 2 \cdot n_{\text{pred}} \cdot T_{\text{pred}} \cdot v_{\text{p}}$ (assuming $T_{\text{pred}}$, $T_{\text{p}}$)

Convective (Minimum) Type I ELMs:
$W_{\text{pred}}/W_{\text{ELM}} < 0.05$ is acceptable for ITER.

- Particle loss ($\Delta n_{\text{pred}}/n_{\text{pred}}$) at high $n_{\text{pred}}$ is important to determine ELM energy loss.

2.2. ELM Power and Particle Fluxes on Plasma Facing Components
- $t_{\text{ELM}}$ (ELM heat deposition time) 0.1–1 ms, which were not correlated to $t_{\text{ELM}}$ (0.25 ms)
  (Mirror coils and soft X-ray data)

- $t_{\text{ELM}}$ is longer than $t_{\text{IBW}}$ (best IRTV data in JET)
  $t_{\text{IBW}}$ arrives after $T_{\text{surf}}$ peak.

2.3. Discussions: extrapolation to ITER
- Extrapolation to ITER remains uncertain:
  low $v_{\text{p}}$ ($\sim 0.05$) and high $n_{\text{pred}}$/$n_{\text{pred}}$ ($\sim 0.8$) were not achieved simultaneously in experiments.

Model 1: $\Delta W_{\text{ELM}} \propto v_{\text{p}}$ ($e$-collisionality with $T_{\text{surf}}$, $n_{\text{pred}}$)

- $\Delta W_{\text{ELM}}$ dependence on $v_{\text{p}}$ can be due to MHD:
  $v_{\text{p}}$ increases with $v_{\text{p}}$, $\Delta W_{\text{ELM}}$ for ITER higher.

- $W_{\text{ELM}}/W_{\text{pred}} < 0.15$–0.2 is unacceptable for ITER.

Model 2: Influence of Parallel transport in $\Delta W_{\text{ELM}}$

Hypothesis:
$$\zeta'_{\text{ELM}} = \zeta_{\text{IBW}} (1 + \sqrt{3}/\zeta'_{\text{ELM}})$$


$$\Delta W_{\text{ELM}}/W_{\text{pred}} = 0.1–0.15$$
will be allowable for an inclined target option of ITER divertor!
ITER ITPA SOL and Divertor Physics Group CT/F-01 p.9

Model 3: \( \tau_{\text{ion}}^{\text{iso}} \rightarrow \text{lower } \Delta W_{\text{ELM}}/W_{\text{ped}} \text{ for ITER} \)

Experiments show large influence of SOL transport:
\[ \tau_{\text{ion}}^{\text{iso}} = \tau_{\text{ion}}^{\text{ped}} = 2 \tau_{\text{ion}}^{\text{iso}} \]

\[ \Delta W_{\text{ELM}}/W_{\text{ped}} = 0.05-0.1 \]
will be also allowable
for an inclined target
option of ITER divertor!

Experiments to distinguish between \( \nu_{\text{ion}}^{\text{iso}} \) (pedestal MHD) and \( \nu_{\text{ion}}^{\text{iso}} \) (SOL transport) are required.

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2.4 Summary & Further Work

- Reduction of \( \Delta W_{\text{ELM}}/W_{\text{ped}} \) with \( n_{\text{ped}} \) was found,
  \( \rightarrow \) reduction of \( \Delta n_{\text{ped}}/n_{\text{ped}} \)

Extrapolation of convective Type I ELM (M-II) to ITER should be investigated

- \( \Delta W_{\text{ELM}}/W_{\text{ped}} \) with \( \nu_{\text{ped}}^{\text{iso}} \) for all experiments
  \( \rightarrow \nu_{\text{ion}}^{\text{iso}} \) may determine ELM loss energy: \( \Delta W_{\text{ELM}} \)

- Hot flux during ELM was governed by SOL transport
  \( n_{\text{ped}} \) MHD activity:
  \( dW_{\text{ELM}}/dn_{\text{ped}} = -0.25(\eta_{\text{ped}}^{\text{iso}}/\Delta n_{\text{ped}})^{1/3} \) in JET/DIII-D/JT-60U

Determination of conductive and convective processes in SOL is crucial for quantitative evaluation of Type I ELM energy load.

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3. SOL flow and plasma drift

- SOL flow plays an important role on determining particle flux towards the divertor and
  impurity shielding/exhaust from the main plasma.

- Tokamak experiments [JT-60U, JET, C-40D, ASDEX-U, DIII-D]
  \( \rightarrow \) flow reversal (flow away from the divertor)
  was measured with Mach probe (ion flow)
  and spectroscopy (impurity flow) for the ion VB drift direction
towards divertor.

- Plasma drifts produce ion flux, and the poloidal variation influences parallel SOL flow.
  Drift effects (ExB, Drift and damping) have been investigated by models and 2D simulations [UEDEGE, BD].

- Comparison between experiments [JT-60U, JET] and UEDGE is shown: Determinations of parallel SOL flow and Drift effects are established in order to optimize the divertor design relevant to a reactor.

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3.1. Flow pattern comparison:

- experiment and UEDGE simulation

UEDGE:

- 2D fluid code, fluid neutral model,
  Mult-species carbon ion model,
  Phys.&Chem. Sputtering (Eckstein/Toronto rates)
  \( \leftarrow \) ExB, Drift:
diamagnetic drifts are included


- Effects of Drifts [JT-60U]

- L-mode:
  \( L=1.6 \text{MA}, B=1.27, P_{\text{therm}}=28 \text{MW} \)
  UEDGE: \( D_{\text{inj}} = 0.25 \text{eV}, \beta_{\text{Ti}} = 10^{-3} \)

Parallel flow is produced away from the divertor,
in particular, at Low-Field-Side SOL (mostly due to ion VB drift)

- Comparison between experiment [JT-60U, JET] and UEDGE is shown: Determinations of parallel SOL flow and Drift effects are established in order to optimize the divertor design relevant to a reactor.

Parallel flow patterns in JET, JT-60U

- Parallel flow towards High-Field-Side and Top. M = 0 at HFS separatrix.
- Profiles qualitatively consistent with experiments, but Mach numbers are small.

Simulations show that M increases, and at M at LFS midplane decreases.

Subsonic flow in experiments should be distinguished by improving simulation and Mach probe model.

ExB drift flux in SOL: \( J_{\text{exb}} \) decreases from 30-50% to 10% with increasing \( n_e \) (JT-60U)

- Relatively low SOL collisionality in ITER, \( \nu_e = 2\pi R_{\text{exb}} \nu_e /10 \), \( T_e = 180 \text{eV}, n_e = 3 \times 10^{19} \text{m}^{-3} \)
- \( \Gamma_{\text{exb}} / \Gamma_t \sim 30-40\% \) would be expected.

ExB drift flux in Private flux region:
- Comparable to \( \Gamma_t \)
- HFS-enhanced asymmetry in divertor ion flux.

Private ExB drift flux should be investigated in detach divertor with maintaining high \( T_e \) below the X-point.

Private ExB drift also produces Carbon ion flux to HFS divertor:
- Carbon accumulation on HFS divertor target.

Summary & Further Work

- Simulation and experiments determined that Drifts have influence on total mass flow in SOL, and parallel flow pattern is modified.

Design work including drift effects will be important to optimize the divertor and pumping geometries relevant to a tokamak reactor.

Subsonic flow in experiments should be distinguished by improving simulation and Mach probe model.

Drift flow in Private flux region may lead to accumulation of carbon on the HFS divertor:
- This influences location of Tritium co-deposition.

Experimental and simulation work is required to distinguish source/transport of redeposited carbons.
4. Chemical Erosion
Chemical erosion is a serious concern for the use of graphite materials in fusion devices: Lifetime of the target, and Tritium retention

- Chemical erosion yields have been characterized for ion flux of $10^9$ to $10^{10}$ m$^{-2}$s$^{-1}$ and low impact energy in many Tokamaks and Laboratories.


- Factors to determine Chemical Erosion Yields:
  1. Surface temperature ($T_{surf}$)
  2. Impact energy of Ion (D*), and Atom (D)

Strong $T_{surf}$ dependence decreases with lower impact energy of D* (e.g. attach divertor), but becomes strong for D (e.g. detach divertor).

4.2 Chemical erosion yields in Tokamaks

- Re-deposition dominant area: [JET HFS divertor, Detached and exposed by high density and low energy D* & H, etc.]

- Soft layer, Re-erosion due to atomic and low energy ion fluxes, Contribution of Higher Hydrocarbon (C$_2$D$_2$) to total sputtered Carbon is dominant (-1/2 of total CD)

- Overall $Y_{chem} = 7 - 20$

Strong $T_{surf}$ dependence

- Contribution from Higher Hydrocarbons (C$_2$D$_2$)

TEXTOR - 50% of CD
JET - 50% of CD
JET HFS - 50% of CD
JT-60U LFS - 40% of CD
JT-60U LFS - 80% of CD, in Erosion dominant area (JT-60U LFS), large contribution of Higher Hydrocarbons was observed by simultaneous C & CD line measurement.


- Flux dependence of Chemical erosion yields
TEXTOR: no clear dependence up to $10^{11}$ m$^{-2}$s$^{-1}$ And decrease for larger flux
JET: no clear dependence
JT-60U LFS: $Y_{cor}$ high flux dependence
ASDEX: $Y_{cor}$ low flux dependence

*DIEP: long exposed bias: very low chemical yields (lower than 0.3-0.5%) was reported.
4.3 Summary and Future work

Methane yield database (from CD band spectroscopy) was characterized with $T_{\text{surf}}$ (100-300°C) and $\Gamma = 10^{14}-10^{16}$ m$^2$ s$^{-1}$ in re-deposition and net-erosion dominant areas.

- No yet definitive conclusion at high flux density in ITER (several $10^{19}$ m$^2$ s$^{-1}$).

- Better and more determinations of energy, redeposition, photon efficiency etc. will be required accompanying with flux dependence.

- Contribution of Higher Hydrocarbon yields should need more attention under the high flux density.

<table>
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<th>Chemical Compound</th>
<th>Tsurf (°C)</th>
<th>$\Gamma$ (m$^2$ s$^{-1}$)</th>
<th>$\Delta H_{\text{erosion}}$ (kJ g$^{-1}$)</th>
<th>$\Delta H_{\text{deposition}}$ (kJ g$^{-1}$)</th>
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</table>

5. Conclusions

Understanding of important physics issues for the ITER divertor design and operation,

(i) divator heat load associated with Type-I ELMs,

(ii) SOL transport produced by SOL flow and drifts, have been progressed based on the experimental and numerical work from multi-machine database.

(iii) There has been progress in the characterization of chemical erosion yields in C-clad divertor tokamaks.

In order to make reliable extrapolation for ITER, the R&D program needs to better address the physics of the erosion mechanisms and the transportation and re-deposition of eroded material and resulting mixing effects.

Acknowledgments

Authors would like to thank contributions from expert members, who have provided experiment and numerical results and given constructive comments.

They also thank continuous support of institutes/universities related to our joint work.
2. Collaboration Papers

IAEA-CN-94/EX/P2-03

2.1 Study of Integrated High-Performance Regimes with Impurity Injection in JT-60U Discharges

K. W. Hill,1 W. Dorland,2 D. R. Ernst,3 D. Mikkelsen,1 G. Rewoldt,1 S. Higashijima,4 N. Asakura,4 H. Shirai,4 T. Takizuka,4 S. Konoshima,4 Y. Kamada,4 H. Kubo,4 and Y. Miura4

1Princeton Plasma Physics Lab., Princeton University, PO Box 451, Princeton, NJ 08543, 2Institute for Plasma Research, Univ. of Maryland, College Park, MD 20742, 3Plasma Science and Fusion Center, Massachusetts Institute of Technology, Cambridge, MA, 4Japan Atomic Energy Research Institute, Naka-machi, Naka-gun, Ibaraki-ken, 311-0193 Japan

E-mail: khill@pppl.gov

Abstract. Injection of argon in ELMy H-mode discharges has enabled extension of these plasmas in JT-60U to high density with good confinement and high radiation loss power fraction. By operation of the outer divertor strike point on the divertor dome, confinement improvement, electron density ne, and radiation-loss power fractions Prad / Pheat close to the ITER requirements have been achieved simultaneously (HH08Y,2) ~ 1; ne ~ 0.8 nGW, the Greenwald density limit; and Prad / Pheat ~ 0.8). Linear microstability analyses show a region of reduced growth rate in the argon-seeded discharge for the ITG/TEM mode near r/a = 0.64; the ETG growth rate is greatly reduced across the entire profile and completely quenched around r/a = 0.5 ~ 0.7. The reduced growth rates are related to the impurity effects, the deuteron dilution, and the temperature gradients.

1. INTRODUCTION

Achievement of high performance plasmas with high density, and acceptable heat loading of the divertor is critical for fusion reactors. Both the steady state divertor heat load and the transient heat load due to ELMs must be reduced in ELMy H-mode plasmas. D2 puffing can increase the density and reduce the divertor heat load by increasing the radiated power, but it results in confinement degradation at high density. [1] Injection of argon, however, has enabled extension of these plasmas in JT-60U to high density with good confinement and high radiation loss power fraction. [2] By operation of the outer divertor strike point on the divertor dome, confinement improvement, electron density ne, and radiation-loss power fractions Prad / Pheat close to the ITER requirements have been achieved simultaneously (HH08Y,2) ~ 1; ne ~ 0.8 times the Greenwald density limit, nGW; and Prad / Pheat ~ 0.8). Moreover large ELMs, with concomitant large heat fluxes to the divertor plates, can be suppressed. Under these conditions the fuel purity was ~ 70%. To understand the mechanisms by which impurity seeding leads to improved confinement in ELMy H-mode plasmas, linear microstability analyses have been performed using the GS2 gyrokinetic code, [3] without rotation effects, and the FULL code, [4] with and without rotation effects.

2. EXPERIMENTAL SCENARIO

ELMy H-mode plasmas with Ip = 1.2 MA, Bt = 2.5 ~ 2.6 T, Pnbi = 18 MW, elongation = 1.4, and triangularity $\delta = 0.3$ were studied in JT-60U. Discharge E39532 was a plasma in which the outer divertor strike point was kept on the divertor septum or dome ("dome-top" configuration), and both D2 and argon gas were puffed into the discharge. This shot, at time 7.35 s, was compared with a non-argon reference shot E36349 at 9.05 s in the standard divertor configuration with only D2 puffing.
Both the electron temperature, $T_e$, and the ion temperature, $T_i$, are higher in the argon-seeded discharge than in the reference shot, especially in the core, as illustrated in Fig. 1a and 1b. The electron density profile, $n_e$, is somewhat more centrally peaked, and $Z_{eff}$ is higher, due to the argon content. The confinement improvement factors, HH98(y,2), are 0.65 and 1 for the reference and Ar-seeded shots, respectively, the stored energies are 1.9 and 2.5 MJ, and the radiated power fractions are about 50% and 80%. The electron densities are 0.67 times $n_{GW}$ for both shots. The q profiles are monotonic, with positive magnetic shear.

3. MICROSTABILITY ANALYSES

Improved confinement with impurity seeding in other tokamaks has been attributed to reduction of ion thermal transport due to ExB shear suppression of turbulent fluctuations and reduction of toroidal drift wave growth rates [5,6]. Thus, we have performed linear gyrokinetic growth rate calculations without rotation and in the electrostatic limit using the GS2 code [3], for ITG/TEM and ETG modes in the JT-60U discharges, as well as ITG/TEM calculations both with and without rotation effects for the Ar-seeded discharge using the FULL code [4].

Figure 2 shows that the argon-seeded discharge has a region of significantly reduced linear growth rate for the ITG/TEM mode, $\gamma_{ITG}$, near $\rho = r/a = 0.65$, relative to that of the reference shot, and that the ETG mode is also significantly reduced across the entire profile, and is completely quenched over the range $\rho = 0.5 - 0.7$. These rates were calculated with GS2. Here $\rho_s = \rho_i/\sqrt{2}$, where $\rho_i$ is the ion gyroradius. The reference, non-argon discharge has a significant growth rate for both modes across most of the
profile from \( \rho = 0.2 \) to \( \rho = 0.9 \). Notice that for the unseeded shot (black curves) \( \gamma_{\text{ETG}} \) is 25 – 100 times higher than \( \gamma_{\text{ITG}} \). Also note that finite Debye length corrections were inadvertently not included in the ETG calculations; however, inclusion of these effects at \( \rho = 0.47 \) and \( k_\theta \rho_s = 35 \) resulted in only a 6% decrease in \( \gamma_{\text{ETG}} \), and no change in the real frequency. Near the radii of the minima of the curves in Fig. 2a, the real frequency of the instability changes from positive (ion diamagnetic direction) at smaller radii to negative (electron diamagnetic direction) at larger radii. Thus the plasma transitions from one unstable root to another, going through a minimum in the growth rate. Note that this transition is further out in radius for the non-Ar shot (\( \rho = 0.75 \)) than for the Ar-seeded shot (\( \rho = 0.64 \)). In the observed ion temperature profile for the Ar-seeded discharge (Fig. 1a), the slope of the profile shows an increase toward smaller radii which may be related to the minimum in \( \gamma_{\text{ITG}} \). The blue curve in Fig. 2a represents an analysis based on a flat \( Z_{\text{eff}} \) profile of 3.67. The lower growth rate for the blue curve apparently results from a lower deuteron fraction (about 12% lower) than that for the red curve. The red curve is based on the \( Z_{\text{eff}} \) profile of Fig. 1d. The shapes of the \( \gamma_{\text{ITG}} \) profiles reflect largely the ion temperature gradients, \( R/L_{T_i} \), for the discharges. For the reference discharge \( R/L_{T_i} \) is approximately constant, varying only from 11.3 to 12.4 over the large radial range \( \rho = 0.3 – 0.75 \), whereas, for the Ar-seeded shot \( R/L_{T_i} \) varies widely from a peak value of 18.2 at \( \rho = 0.4 \), i.e. near the maximum in the \( \gamma_{\text{ITG}} \) profile, to a minimum of 6 at \( \rho = 0.68 \), near the minimum in \( \gamma_{\text{ITG}} \).

The effect of rotation on \( \gamma_{\text{ITG}} \) as calculated by the FULL code is very small, since the toroidal rotation is small in these high density plasmas with nearly balanced neutral-beam injection. The resulting radial electric field is small, and the ExB shearing rates, \( \omega_{\text{ExB}} \), [7] are much smaller than \( \gamma_{\text{ITG}} \) as shown by the thin, dashed curves in Fig. 2a. The FULL-code \( \gamma_{\text{ITG}} \) profiles for the Ar-seeded discharge both with and without rotation are similar to the red curve in Fig. 2a. With rotation turned on, \( \gamma_{\text{ITG}} \) is lower by only about 20% at \( \rho = 0.6 \), where \( \gamma_{\text{ITG}} \) is already very small, and about 25% at \( \rho = 0.87 \). At all other radii the two FULL-code \( \gamma_{\text{ITG}} \) curves (with and without rotation) almost overlap.

4. EFFECT OF VARYING IMPURITIES AND TEMPERATURE GRADIENT

The effect on the ITG/TEM growth rate of adding or removing the argon for the two discharges is simulated in Fig. 3. In Fig. 3a the red curve is calculated from a discharge having the \( T_p \), \( T_e \), and \( n_e \) profiles of the reference, no-argon shot, with the \( Z_{\text{eff}} \) and carbon and argon density profiles from the argon-seeded shot. The black curve is the no-argon shot shown in Fig. 2a. The relationship between the red and black growth rate curves is similar to that of the deuteron fraction for the two cases. These fractions for the red and black curves are about 0.64 and 0.68, respectively, at \( \rho = 0.35 \), and 0.58 and 0.72 at \( \rho = 0.7 \). In Fig. 3b the effect of removing the argon from the argon-seeded discharge (red curve, same as in Fig 1a) is simulated in the black curve. We see that a significantly higher growth rate is predicted near \( \rho = 0.35 \), but the growth rate changes little near \( \rho = 0.65 \). In this case, the deuteron fraction in the case of the black curve is actually about 0.02 – 0.03 lower than that of the red curve.
The effectiveness of the argon seeding on increasing the critical $T_e$ gradient, $R/L_{Te\text{crit}}$, for the ETG mode is illustrated in Fig. 4. The linear growth rate is calculated for $k_x \rho_s = 35$ at the radial location $\rho = 0.47$ for both the argon-seeded and the reference shots. The critical gradient ($R/L_{Te}$ value at intersection with the horizontal axis) is increased from 2.8 without argon to 4.8 with argon, as indicated by the two left-most arrow heads. The right arrow heads indicate the measured gradients.

5. DISCUSSION

Impurity seeding allows ELMy H-mode plasmas in JT-60U to be extended to higher density while maintaining good confinement. In particular, when combined with high triangularity and dome-top operation, near-ITER requirements of confinement improvement, density, radiated power fraction, and fuel purity can be achieved. The confinement improvement more than offsets the effects of fuel dilution, resulting in increased fusion neutron output. Moreover large ELMs, with concomitant large transient heat fluxes to the divertor plates, can be suppressed. A possible explanation for this suppression, to be investigated, is access to second stability for ideal ballooning modes, as suggested in high-$\delta$ discharges without Ar injection. [3] The improved-confinement argon-seeded plasma analyzed in this work showed a significant decrease in the linear growth rate for the ITG/TEM instability near $\rho = 0.65$, a significant reduction of the ETG growth rate across the outer 80% of the plasma, and complete suppression of the ETG in the range $\rho = 0.53 - 0.7$. Simulations in which the deuteron dilution is reduced by about 12% in the Ar-seeded shot by replacing the measured C and Ar by a single, average $Z = 10$ impurity, result also in a decrease of $\gamma_{ITG}$ by about 15% across the profile in the region $\rho = 0.25 - 0.6$. Simulations in which the impurities are exchanged between the Ar and reference shots show a large decrease in $\gamma_{ITG}$ for the reference shot if Ar is added, and a smaller increase in the Ar-seed discharge if the argon is removed. A scan in which $R/L_{Te}$ is varied shows that $R/L_{Te\text{crit}}$ is almost a
factor of 2 higher in the Ar-seeded shot than in the reference shot.

Improved confinement with impurity seeding and ITER-like parameters has been achieved in other tokamaks [5, 6, 8-10] and analysis of the mechanisms for confinement enhancement have been done. [5, 6, 11, 12]. The improved confinement has been correlated with reduced measured turbulence [13], consistent with the model of turbulence causing anomalous transport. In [5] and [6] ExB shearing is reported to be an important mechanism for the improved performance. Although the FULL code analysis indicates that for the JT-60U Ar-seeded discharge ExB shearing has little effect on $\gamma_{\text{ITG}}$ in the high-density, final state, it is still possible that this mechanism is a factor at lower densities, during the evolution to the final, improved-confinement state. The effect of impurities on reducing the ETG growth rate was first studied in [6]. The possibility that ETG turbulence can be relevant to magnetic confinement was reported and analyzed by nonlinear, electromagnetic, gyrokinetic simulations in [14]. For the impurity-seeded, improved-confinement RI mode in TEXTOR, Tokar presented an explanation [12] in which the increased $Z_{\text{eff}}$ from impurity seeding reduces the velocity of the ion diamagnetic drift and, thus, the growth rate of the ITG instability. In this case the TEM becomes more important, and leads, through a pinch component of the TEM flux, to density peaking in monotonic-q, positive-shear discharges. The density peaking then leads to further reduction of the ITG instability. Further analysis of the JT-60U argon-seeded and reference discharges needs to be done to quantify the magnitude of increased ExB shearing as a function of time during the evolution of the improved-confinement, discharges, and the radial profiles of the separate ITG, TEM, and ETG instabilities and their relevance to particle and thermal transport.

ACKNOWLEDGEMENTS

Helpful discussions with C. Bourdelle, R. Nazikian, and M. Redi are gratefully acknowledged. The authors wish to thank the members of JAERI who have contributed to the JT-60 project. This work was supported partly by the US Department of Energy under contract no. DE-AC02-76-CH03073.

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Study of Integrated High-Performance Regimes with Impurity Injection in JT-60U Discharges

K. W. Hill,1 W. Dorland,2 D. R. Ernst,3 D. Mikkelsen,1 G. Rewoldt,1 S. Higashijima,4 N. Asakura,4 H. Shirai,5 T. Takizuka,6 S. Konoshima,7 Y. Kamada,7 H. Kubo,7 and Y. Miura8

1Princeton Plasma Physics Laboratory, Princeton, NJ, USA
2Institute for Plasma Research, Univ. of Maryland, College Park, MD, USA
3Japan Atomic Energy Research Institute, Naka-machi, Nakagun, Ibaraki, Japan

Presented at the 15th IAEA Fusion Energy Conference
October 14-19, 2002 - Lyon France

SUMMARY – Argon seeded discharges

Part I
• Near reactor requirements of high confinement, density, radiated power fraction, and fuel purity achieved simultaneously in JT-60U.
• Transient ELM heat load reduced by factor ~1/5 – 1/3 in dome-top configuration
• Particle confinement increased.

Part II
• Confinement enhancement with argon seeding is consistent with gyrokinetic microstability calculations.
• Reduced ITG growth rate in outer region is largely a Ti-profile effect; dilution causes a smaller effect.
• Effect of rotation is small.

MOTIVATION

• Achieve good confinement at high density in ELMy H-mode discharges
• Reduce steady-state and ELM-induced heat loads to divertor.

OUTLINE

• Reactor requirements almost simultaneously achieved in JT-60U with Ar seeding
• Experimental results from JT-60U with Ar seeding
• Linear microinstability analysis with GS2 and FULL codes
• Predictive modeling with “stiff” models for Ar seeding cases
• Summary and future work

JT-60U has achieved near ITER performance requirements simultaneously with argon seeding

<table>
<thead>
<tr>
<th>Parameter</th>
<th>JT-60U (no Ar)</th>
<th>JT-60U (Ar)</th>
<th>ITER</th>
</tr>
</thead>
<tbody>
<tr>
<td>HH(hg,2)</td>
<td>0.65</td>
<td>1.00</td>
<td>1.00</td>
</tr>
<tr>
<td>n_e/n_e0</td>
<td>0.67</td>
<td>0.80</td>
<td>0.85</td>
</tr>
<tr>
<td>P_red/P_neal</td>
<td>0.60</td>
<td>0.80</td>
<td>0.70</td>
</tr>
<tr>
<td>ELM-induced heat spikes</td>
<td>large</td>
<td>x 1/3-1/5 reduction</td>
<td>small</td>
</tr>
<tr>
<td>Fuel Purity</td>
<td>0.80</td>
<td>0.70</td>
<td>0.80</td>
</tr>
</tbody>
</table>
Two plasma configurations have been explored for high $n_e$.

With Ar injection, high stored energy and low recycling are maintained at high density.

Motivation: Dome top: outer strike point and varied dome-top efficient fueling of D and Ar due to recycling near X-point.
With Ar injection, large ELM heat flux reduced by factor 1/3 - 1/5.

around HH = 0.95

**Reference (without Ar, 0.49 nGW)**

**Standard (0.84 nGW)**

**Dome-top (0.70 nGW)**

Heat flux due to ELMs at inner strike point (kW/m²)

Time (s)

With Ar Injection

Although $f_{ELM}$ decreases, the maximum heat flux does not decrease.

Dome-top: Frequency and amplitude of ELM heat flux lower (factor of 1/3 - 1/5).

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In dome-top case, $n_D/n_e \sim 0.7$ at 0.8 $n_{GW}$: ITER $n_D/n_e \sim 0.8$.

---

**Core microstability analysis of argon-seeded discharge**

- Linear stability analyses with FULL and GS2 codes indicate:
- ITG maximum growth rate significantly lower in outer region of Ar-seeded discharge, relative to reference discharge.
- ETG maximum growth lower across entire profile.
- Rotation effect very small.
- Effect of adding argon to reference discharge or removing argon from Ar-seeded discharge (change in dilution) relatively small.

---

**Plasma profiles with and without argon seeding**

At 0.65 $n_{GW}$: Ar injection increases $n_D$ by ~25% despite a reduction in $n_D/n_e$ from ~60% to ~45%, because of a significant confinement improvement.

At 0.8 $n_{GW}$, $n_D$ reductions due to intrinsic impurity (~C) and Ar are 15% each.
Ar density optimization and carbon density reduction are required for high purity.

- $T_e, T_i, Z_{eff}$ higher with argon
- $n_e$ more centrally peaked
ITG growth rate, $\gamma$, reduced in outer region ($r/a > 0.5$) of Ar shot

- $\gamma_{ETG}$ reduced everywhere
- $\gamma_{TEM}$ much smaller than $\gamma_{ETG}$
- Rotation not a factor

Models qualitatively different in core for Ar discharge

Dilution does not significantly change ITG growth rate

Adding argon to reference discharge

Removing argon from Ar-seeded discharge

Summary – Argon seeded discharges

Part I
- Near reactor requirements of high confinement, density, radiated power fraction, and fuel purity achieved simultaneously in JT-6U.
- Transient ELM heat load reduced by factor $\sim 1/5 - 1/3$ in dome-top configuration
- Particle confinement increased.

Part II
- Confinement enhancement with argon seeding is consistent with gyrokinetic microstability calculations.
- Reduced ITG growth rate in outer region is largely a Ti-profile effect; dilution causes a smaller effect.
- Effect of rotation is small.
Future work

- Increased ExB shearing, as well as impurity-induced reduction in drift-wave turbulence important in DIII-D (Murakami); particle pinch a factor leading to density peaking in RI mode, according to modeling (Tokai).

- Analyze JT-60U discharges for effect of ExB shearing during evolution to final, high density discharges.

- Evaluate particle fluxes with FULL code for JT-60U discharges.
2.2 Role of Low Order Rational \( q \) Values on the ITB-events in JT-60U Plasmas


1) Nuclear Fusion Inst., RRC Kurchatov Institute, 123182, Kurchatov sq. 1 Moscow Russia
2) JAERI, Naka Fusion Research Est., Naka-machi, Naka-gun, Ibaraki-ken 311-0193, Japan

e-mail: contact: neudatchin@nfl.kiae.ru

Abstract. The formation of internal transport barriers (ITBs) near \( q=2,3 \) surfaces in normal (\( N_rS \)) or optimized shear discharges of JT-60U and JET is well known [1,2]. In reverse shear (RS) JT-60U plasmas, the role of \( q \) minimum (\( q_{\text{min}} \)) equal to 3.5,3,2.5,2 is not obvious for ITB formation. ITB-events (non-local confinement bifurcations inside and around ITB in a ms timescale) are found in various JT-60U \( N_rS \) and RS plasmas. Under sufficient power, ITB-events are seen at rational and not rational values of \( q_{\text{min}} \). The space-time evolution of \( T_e \) and \( T_i \) is similar even being strongly varied in space and time, suggesting same mechanism(s) of \( T_e \) and \( T_i \) transport. The temporal formation of strong ITB in H-mode under passing of \( q_{\text{min}}=3 \) (after periodical improvements and degradations via ITB-events with 8ms period) in RS mode with \( P_{\text{th}}=8 \text{MW} \) is presented. Under smaller power, ITB-events are observed only at rational values of \( q_{\text{min}} \). In a weak RS shot with \( P_{\text{th}}=4 \text{MW} \), abrupt rise of \( T_e \) is seen at \( q_{\text{min}}=3.5 \), while more cases of \( T_i \) rise are observed. The difference of the \( T_e \) and \( T_i \) evolution seen regularly under the low power, suggests decoupling of \( T_e \) and \( T_i \) transport.

1. Introduction

The formation of internal transport barriers (ITBs) near \( q=2,3 \) surfaces in normal (\( N_rS \)) or optimized shear discharges of JT-60U and JET is well known [1,2]. In reverse shear (RS) JT-60U plasmas, the role of \( q \) minimum (\( q_{\text{min}} \)) equal to 3.5,3,2.5,2 is not obvious for ITB evolution. The transient processes seen under crossing \( q_{\text{min}}=3 \) were first time reported in [3]. Later, non-local confinement bifurcations inside and around ITB (abrupt variations of transport in a ms timescale within \( \sim 0.3 \text{r/a} \)) were found in various JT-60U \( N_rS \) and RS plasmas and called ITB-events [4-6]. The maximum of heat flux variation is located near the position of \( q_{\text{min}} \). The series of ITB-events is able to create the strong ITB in H-mode (\( q_{\text{min}}=2.7 \)) with nearly doubled energy confinement time [6]. The influence of the radial electric field calculated near ITB foot on wider ITB region was highlighted in [7]. Initially, another type of non-local (in \( \sim 90\% \) of volume) abrupt jumps (bifurcations) of transport at fast "global" \( L-L \) transitions was found in JET and JT-60U plasmas with \( N_rS \) [8-9]. At \( L-L \) transitions in JT-60U plasmas with RS and ITB [5-6], the profile of the heat flux jump follows the position of the safety factor minimum and penetrates into RS region deeper for the weak ITB that for the strong one [6]. ITB-event degradation causes L-H transition [6].

2. ITB-events under sufficient NBI power in RS

Under sufficient power in JT-60U RS plasmas, ITB-events are seen at rational and not rational values of \( q_{\text{min}} \) and the space-time evolution of \( T_e \) and \( T_i \) is similar [4-6]. In the present paper, we highlight the similarity of \( T_e \) and \( T_i \) evolution by detailed comparison \( T_e(r,t) \) and \( T_i(r,t) \) behavior (see Fig. 1) during strong ITB creation via series of ITB-events: improvements A, C, F and further ITB degradation (shot 32423, 1.5MA/3.7T, L-mode edge, \( P_{\text{th}}=8 \text{MW}, q_{\text{min}}=2.7 \), see evolution of plasma parameters in [4]). The position and the evolution of \( T_e \) measured by 12-channels ECE heterodyne radiometer (data averaged in 0.5ms interval) at channel 11 (\( T_{111} \)) correspond to the \( T_{112} \) evolution (changes of timetraces at the times A, D, F, K on Fig.1). The \( T_i \) is measured with 17ms time resolution and \( \sim 0.06 \text{r/a} \) space resolution half width. The \( T_{65-9} \) evolution corresponds to the \( T_{112} \) evolution (changes of slopes at the times C, E, F, H on Fig.1). The \( T_{65} \) position lies near the \( T_{111} \) position (times A, F, K). The \( T_{65} \) position corresponds to the \( T_{110} \) position. The evolution and the similarity of \( T_e \) and \( T_i \) transport at \( t=6.5-6.68 \) time interval was described in detail [4]. The \( T_{11} \) evolution
EX/P2-06

presented on Fig.1, shows the similarity of the transport in a longer time interval, including the formation of double ITBs clearly seen before time G (weak ITB between $T_e$ and $T_{e_6}$, and strong ITB between $T_{e_{10}}$ and $T_{e_{12}}$. The same trend is observed for $T_i$ profile at $t=6.75s$ (the difference between $T_{i_{12}}$ and $T_{i_{13}}$ is equal to 1 keV and 2.5 keV for $T_{i_{13}}$ and $T_{i_{14}}$ ($r/a=0.73$)). Strong ITB destroys after time G.

The time traces of shot 32474 (1.5MA/3.7T), the evolution of $T_e$ and profiles in H-mode under passing of $q_{min}=3$ are presented on Figs 2(a-d). Four cycles of ITB-events (called periodic ITB-events or P-ITB-events) are seen on $T_e$ evolution after $t=6.68s$ (see Figs 2(e,d)). Each cycle consists of ~4ms ITB-event improvement phase ($T_{e_{3-6}}$ rise and $T_{e_{9-12}}$ decay) and ~4 ms ITB-event

Fig. 1. Similarity of $T_i$ and $T_e$ evolution in shot 32423. $P_{nbi}$ rises from $8\text{MW}$ to $10\text{MW}$ at $t=6.66s$.

Figure 2(a) Time traces of $W$, $P_{nbi}$ and $I_p$ in shot 32474. Transitionless H-mode [10] starts from $t=5.7s$. $P_{nbi}$ start of periodical ITB-events at $q_{min}=3$. (b) Positions of radiometer channels and $T_e$ ($t$) for $t=6.68, 6.9s$. (c-d) $T_e$ time traces at periodical P-ITB-events and ITB-event-improvement $I$
Fig. 3. Profiles of $T_e$, $n_e$, and $q$ before time $P$ at Fig. 2. 
Fig. 4 Profiles of $\delta \chi_e$ estimated for P, I and D ITB-events at $t = 6.9$ ms (dotted line) are shown on Fig. 3. The inversion radius (region between $T_e$ rise and decay at ITB-events on Fig. 2(d)) lies near the position of $q_{\min}$, as usual [4-6]. The $\delta \chi_e$ profiles at ITB-events P, I and D (degradation which occurs later and not shown on Fig. 2) were calculated from abrupt variations $\partial \chi_e / \partial t$ values at times of ITB-events (see method in [4]).

Fig. 5 presents modeling of periodical ITB-events with the profile of the electron heat diffusivity coefficient variation $\delta \chi_e$ shown in Fig. 5(a). The evolution of $\chi_e$ and calculated values of $\delta T_e$ at various radial positions are shown on Figs. 5 (b-c). The calculations reasonably describe the experiments shown on Fig. 3(c-d). We suppose that the periodical "global" L-H-L transitions with 20ms period (10ms H-mode phase and 10ms L-mode phase) found in JET [11] are clear physical analogue to the periodical ITB-events described above.

3. ITB-events under small NBI power in RS

Fig. 6. Timetraces of $I_p$, $W$, $H_{\alpha}$, $P_{\text{net}}$ and $q_{\min}$ in shot 36639

Under smaller power, ITB-events (in ~20 pulses studied) are connected with some low order rational $q_{\min}$ values. ITB-events are found at $P_{\text{net}} = 2.5$ MW in the latest phase.
of RS discharge 36639 (1.4MA /3.8T) under $q_{min}=2.5$ (see Fig. 6).

Moreover, the influence of some low order rational $q_{min}$ values is seen clearly for temporal ITB creation on $T_e$. The timetraces of shot 38976 (1.3MA/3.7T) are shown on Fig.7. The abrupt rise of $T_e$ is seen only once at $q_{min}=3.5$ at $t=5.87$s while more cases of $T_i$ rise are observed (after $t=6.1$s also). The timetrace of the row heterodyne data is shown on Fig.8. The rise of $T_e$ is seen at $t=5.87$s ($q_{min}=3.5$ at this time) in the wide region $0.18 < r/a < 0.42$. The profile of the electron heat diffusivity variation $\delta \chi_e$ is obtained from abrupt variation of $\partial T_e/\partial t$ values at $t=5.87$s in the same way like described in detail in [4] and is wide in space (in the region over $0.5r/a$). The $q$ profile at $t=5.9$s is presented on Fig. 9. The wide region of small shear is observed clearly. In this particular shot 38976 case, $T_i$ evolves similar to $T_e$ at $q_{min}=3.5$ and rises separately from $T_e$ at $t=6.1$s (see Fig.10). The same behavior of $T_e$ and $T_i$ is observed in the similar shot 38974. The rise of $T_e$ occurs at $t=5.92$s (close to $t=5.86$s in shot 3896).

Fig. 10 Timetraces of $T_i$ in shot 38976

The difference of the $T_e$ and $T_i$ evolution seen regularly under the low power, suggests decoupling of $T_e$ and $T_i$ transport.

4. Discussion and Conclusions
Besides well-known formation of ITBs near $q=2.3$ surfaces in NIRS or optimized shear discharges of JT-60U and JET [1-2], similar features are sometimes seen in small machines with ECR heating. The existence of the zones with improved transport near low-order rational $q$ values was reported at RTP [13]. The zone of the improved transport formed by
off-axis ECRH in T-10 (the region with low shear and q near 1 inside ~0.3r/a) is able to survive at R/L_Te = Rgrad T_e / T_e up to 23 with x_e ~0.1-0.2 m^2/s [14].

Under sufficient power in RS JT-60U plasmas, the space-time evolution of T_e and T_i due to series of ITB-events improvements and degradations is similar even being strongly varied in time and space. The same physical mechanism(s) is responsible for T_e and T_i evolution at ITB-events. ITB-events are observed under various values of q_min. The periodical ITB-events with ~8ms period are found in H-mode RS plasmas under crossing q_min=3. Probably the clearest analogues are periodical "global" L-H-L transitions with 20ms period found in JET [11].

Under smaller power in JT-60U RS plasmas, the space-time evolution of T_e and T_i could be different from each other. The transport looks different for T_e and T_i. The influence of some low order rational q_min values is seen clearly for temporal creation of the ITB on T_e and for series of small-scale ITB-events on T_e. At present, we observe ITB-events at low order rational q_min values only.

ITB-events triggers could be different in various JT-60U plasmas. The role of MHD-activity as ITB-event improvement trigger should be studied in future. The correlation of the MHD-activity and ITB-event improvement within a millisecond timescale was found sometimes (not frequently). The correlation of the coupled edge-core MHD-activity and ITB formation (unfortunately within ~100ms time interval) was reported on JET [11]. A physical mechanism of non-local bifurcations of the core transport at the ITB-events is still unclear. Further study of ITB-events (especially in low power cases) and ITB-events triggers is necessary.

References
Role of Low Order Rational q Values on the ITB-events in JT-60U Plasmas


1) Nuclear Fusion Inst., RRC Kurchatov Institute, 123182, Kurchatov sq., Moscow, Russia
2) JAEA, Naka Tokon Research Ext., Naka-machi, Naka-gun, Ibaraki-ken 311-0195, Japan

Outline

1. Introduction
2. ITB-events under sufficient NBI power
3. ITB-events under small NBI power
4. Discussion and Conclusions
+ Appendix: to clarify detail

1. Introduction

- ITB-events formation near q=2.3 surfaces in normal (HS) or optimized shear discharges of JT-60U and JET is well known [1-2]
- Variations of $T_e$ slopes under crossing $q_{min}$ observed at ITB in JT-60U RS plasmas [3]
- Abrupt variations of $\delta_T$ at ITB were seen in JT-60U HS plasmas [4]
- Detail study with 12-channels ECE radiometer:

- Abrupt in time and wide in space (inside ~ 0.3-0.4 /a) "spontaneous-like" jumps of $\chi_e i$ seen as "bipolar" perturbations of $T_e i$ ⇒ were found

for RS JT-60U plasmas with weak ITB's and called "ITB-events" [5] !!

- Variations of $\chi_e i$ were extended well to smaller (shear) > 0 region [5]

1. Introduction

- ITB-events (improvements and degradations) under sufficient power are seen under various $q_{min}$

- ITB-events able to create strong ITB in H-mode

[Diagram showing ITB-events and their impacts on plasma parameters]

[Diagram showing variations of $\chi_e i$ over time]

References:
3) Fujita et al 1993 Phys. Rev. Lett. 70 3652
Introduction

The influence of low-order rational values at small tokamaks:
- improved confinement zones near q=1,5,2, at RTP [7]
- Study of electron heat pulse propagation with
  130+140 GHz gyrotrons in T-10 sawteeth-free plasmas with low-shear zone near q=1 [8]
  1) 140GHz low-field-side off-axis - damped sawteeth
     a) Outward HPP from 140GHz on axis switching-on
     b) Inward HPP created by switching-off of the off-axis

ITB-events in NtS JT-60U plasmas (Pnbi=22MW [7]):

-5x profiles obtained from (y=1.1) jump at time of ITB-events are wide in RS and NtS.

2. ITB-events under sufficient NBI power

- ITB-events are seen at rational and not rational values of q_{min} and the space-time evolution of T_e and T_i is similar.

- Detailed comparison T_e(t,0) and T_i(t) behavior during strong ITB formation via series of ITB-events and further ITB degradation via series of ITB-events (Pnbi=3MW, 1.5T L-mode with q_{min}=2.7). Formation of double barriers after time G

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-354-
2. ITB-events under sufficient NBI power
The evolution of 1 m H-mode under passage of q95 in E32474 RS 1.5MA/3.8T shot
Timetraces of

Gradual or "transientless" H-mode started around t--5.7s
Periodic ITB-events - trigger

-8x profiles obtained from 6: Tc vs t jump at
time of ITB-events P, I, D
-periodical global L-H-L transitions with 20ms period
(10ms H-mode phase and 10ms L-mode phase) found in JET [9] are
clear physical analogue to the
periodical ITB-events described
above.

3. ITB-events under small NBI power
ITB-events are connected with some low order rational qie values.
ITB-events are found at Pbih=2.5MW in the latest phase of
RS discharge under qie=2.5.
Moreover, the influence of some low order rational qie values is seen
clearly for ITB formation on Tc.

In 1.3MA/3.7T shot with very weak RS (Pbih=2MW in the current ramp-up stage and Pbih=4MW from t=5.8s), abrupt rise of Tc is seen only once
at qie=3.5 while more cases of Tc rise are observed (after t=6.1s also).
- The rise of Tc is seen at
t=5.8s (qie=3.5 at this time) in the wide region 0.2<\(\alpha<0.4\).

- The \(\delta Tc\) is wide in space (over 0.5m).
- In this particular case, Tc evolves similar to Tc at qie=3.5 and rises separately from Tc at t=6.1s.
The same behavior of Tc and Tc is observed in another similar shot.
Conclusions
- Under sufficient NB power (above 4MW) in RS plasmas, the space-time evolution of T_e and T_i, due to series of ITB-events improvements and degradations (creation of strong ITB and further ITB splitting and final degradation) is similar even being strongly varied in time and space. The same physical mechanism(s) is responsible for T_e and T_i evolution in ITB-events. ITB-events are seen under various q_max values.
- The periodic ITB-events with ~4ms period are found in H-mode RS plasmas under covering q_max. Probably the closest analogues are periodic "global" L-H transitions with 20ms period found in JET Formation of stronger ITB occurred after periodic ITB-events

- H-mode ITB-events - small NB power (1MW) under plasma diamagnetic radius of 1.1 and 1.5, different form of RF, different type of RF, different ELM frequency, different plasma edge launched RF, different pulse length, different TF shape and other parameters

- ITB-events triggers could be different in various JT-60U plasmas. The possible role of MHD-activity as ITB-events trigger (ms time scale correlations in some NBI case) is under investigation. The correlation of the coupled edge-core MHD-activity and ITB formation (unfortunately, within ~100ms time interval) was reported on JET [11]. The physical mechanism of non-local formations of the core transport at ITB-events is still unclear and further study of ITB-events (especially in low power cases) and their triggers is under the way.


APPENDIX:
Method of \( \delta \chi_e \) analysis at ITB-event:
For slowly varied \( T_e \) background ?= \( T_e0(t) \)
\[
\frac{3}{2} \frac{\partial (n_e T_e0)}{\partial t} = \div(n_e \delta \chi_e V T_e0) + Q_e
\]
\( \chi_e = \chi_{e0} + \delta \chi_e \) for \( t \geq t_0 \), \( t_0 \) - time of ITB-event
\( \delta T_e = T_e - T_e0 \)
for small time interval after \( t \)
\( V(\chi_e0,T_e0) \) constant

Influence of density flux jump was checked analytically and numerically, regarding it now:
\( \delta \chi_{e0} + \delta \chi_e(V T_e0 + V \delta T_e0) = \delta \chi_e V T_e0 + \delta \chi_e V T_e0 \)

Equation for \( \delta T_e \) :
\[
\frac{3}{2} n_e \delta \chi_e (V T_e0 + V \delta T_e0) = - \div(n_e \delta \chi_e ? T_e0)
\]
\[
\delta \chi_e = \frac{\div(n_e \delta \chi_e) / \div(n_e ? T_e0 A(r))}{\Delta}
\]

where \( A(r) \) is enclosed surface, \( \Delta \), is a variable of electron heat flux

Accuracy of \( \delta \chi_e \) is less than of \( A \), due to uncertainties of local ?T_e0 (r)
2. 3 Tritium Distribution on Plasma Facing Graphite Tiles of JT-60U

T. Tanabe\(^1\), K. Sugiyama\(^2\), K. Masaki\(^2\), Y. Gotoh\(^3\), K. Tobita\(^3\) and N. Miya\(^3\)

1) Center for Integrated Research in Science and Engineering, Nagoya University, Furo-cho, Chikusa-ku, Nagoya 464-8603, Japan, 2) Department of Nuclear Engineering, Graduate School of Engineering, Nagoya University, Furo-cho, Chikusa-ku, Nagoya 464-8603, Japan, 3) Japan Atomic Energy Research Institute, Naka-machi, Naka-gun, Ibaraki 311-0193, Japan

E-mail contact of main author: tanabe@cirse.nagoya-u.ac.jp

Abstract: Tritium distributions on the graphite divertor tiles, the dome units and the baffle plates of JT-60U were successfully measured. Poloidally, the highest tritium level was found at the dome top tiles and the outer baffle plates, where the plasma did not hit directly. On the other hand, although the toroidal tritium profiles on each tiles appeared uniform, detailed profiles in full toroidal direction clearly showed a periodic variation corresponding to the position of the magnetic field coils, indicating the ripple loss of high energy tritons as suggested by the OFMC code. Finally, the temperature increase owing to the plasma heat load was found to release the once retained tritium.

1. Introduction

Tritium Imaging Plate Technique (TIPT) was found to be very useful to determine the tritium areal distribution on plasma facing graphite tiles. It gives very detailed tritium surface profiles and can also be used as a new diagnostic technique to investigate plasma wall interaction through tritium behavior \([1,2]\). In the present work, TIPT was applied to determine the surface tritium distributions on graphite tiles used as the first wall and the W-shaped divertor in JT-60U, in which tritium is produced by the D-D nuclear reaction.

2. Experimental

2-1. Specimens

The poloidal cross-sectional view of JT-60U with the W-shaped divertor is shown in Fig. 1. CFC graphite (CX-2002U) tiles are used for the divertor targets, the dome top and parts of the baffle tiles. The rest of the divertor region and the first wall are covered with isotropic graphite tiles (IG-340U). All tiles are fixed to metal backings by bolts. The temperature of the divertor tiles was measured by thermocouples installed in the tiles at the depth of 6-mm depth.

The high temperatures of the thermocouples were observed for high neutral beam power, which is aimed at steady-state high performance operation. During the operation, the outer divertor tiles

![Fig. 1 (a) Photograph of JT-60U inside and (b) Cross-sectional view of W-shaped divertor.](image-url)
showed higher temperatures than inner ones. The highest surface temperature of the outer divertor tiles is expected to reach approximately 1100°C. The dome and baffle tiles had a history of comparatively temperatures, nearly at the baking temperature of ~300°C.

The first wall tiles (exposed to plasma from March 1991 to Nov. 1998) and the divertor region tiles (were exposed to plasma from Jun. 1997 to Nov. 1998) were removed for the detailed tritium measurement using TIPPT. Furthermore, all tiles in full toroidal direction of the top of the dome units were also measured. Some of the tiles were also analyzed by full combustion and SEM observation for quantitative tritium analysis and deposition analysis, respectively. The total number of the discharges during the period from Jun. 1997 to Nov. 1998 was approximately 4,000 shots. The amount of tritium produced during this period, which was estimated from neutron production, was about 18 GBq [3].

Before the opening of the vacuum vessel of JT-60U, hydrogen discharges were used to remove the tritium retained in the vacuum vessel. This is followed by air ventilation before fully open to the atmosphere. Thus long term tritium retention in the JT-60U vacuum vessel was estimated to be about 50% of the total production [3].

2-2. Tritium Imaging Plate Technique

The imaging plate (IP) is a radiation image sensor based on photo-stimulated luminescence (PSL). The IP can detect tritium distributed within a depth of ~3.5 μm from the surface of graphite-based tiles. The surface of the IP was in contact with the sample tiles for a day in a dark shielded room. After the exposure, the IP was processed using an imaging plate reader to obtain a digitized tritium intensity mapping. The details of the IP technique is described elsewhere [4].

2-3. Simulation of the High Energy Tritons

The behavior of high energy tritons produced by the D-D reaction in a typical plasma operation of a high βp, H-mode was simulated by the Orbit Following Monte Carlo (OFMC) code, developed in JAERI [5,6]. In the code, coulomb collisions between energetic triton particles and the plasma are simulated using the Monte-Carlo method tracing the triton particle orbits in the magnetic fields. The fields are a combination of the axisymmetric field calculated by a two-dimensional magnetohydrodynamic (MHD) equilibrium code and the non-axisymmetric field produced by the toroidal filed ripple. The Coulomb collisions lead to pitch angle scattering and slowing down of the energetic triton particles. The launching points and pitch angles of test triton particles are also determined by Monte-Carlo method. The test triton particles with the initial energy of 1 MeV are launched at a radius based on the birth profile of the energetic tritons. The orbit of each test particle is followed until it impinges on the wall or slows down to thermal speed.

3. Results

Figure 2 shows the tritium images of the divertor units and the baffle plates together with the line profiles corresponding to the red and blue lines in the images. The tritium profiles were non-uniform in the poloidal direction, but symmetric in the toroidal direction. It was rather surprising to see the highest tritium level observed at the dome top tiles and the outer baffle plates, where the plasma did not directly hit. Tritium levels on both sides of the dome units showed steep gradients in the poloidal direction. Tritium level on the divertor tiles were very small, in particular, both the outer and inner strike points showed the lowest levels owing to the temperature escalation during plasma. We shall discuss this point in further detail later.

Such poloidal tritium profiles were quite consistent with the tritium activity determined by the combustion method, and the highest tritium levels at the top of the dome unit was measured to be around 60 kBq/cm² [7].
Fig. 2 (a) Tritium images of graphite tiles used as the divertor and the baffle plates in JT-60U. Tritium level is higher in the red region and less in the blue region; (b) Tritium line profiles along the poloidal direction; (c) tritium impinging flux to the divertor tiles calculated by OFMC code (see text).

Fig. 3 Full toroidal distribution of tritium on dome top tiles. The positions of toroidal magnetic coils are indicated as columns.

Although the toroidal tritium distribution on each tile seems uniform, the full toroidal tritium distribution from all of the dome top tiles showed certain variation as seen Fig 3. In the figure, the tritium levels of all 240 toroidal dome top tiles are plotted against the toroidal angle corresponding to their positions, exhibiting a periodic variation. In JT-60U, 18 toroidal magnets are placed as indicated in the figure. One can clearly see the corresponding similar periodicity of the tritium retention in the toroidal direction. This is the first clear evidence of the loss of high energy tritons by the ripple in the toroidal magnetic filed (i.e., “ripple loss”).

Figure 4 (a) shows the tritium images of some selected first wall tiles. Although the tritium level was lower than that of the baffle plate, it still contained several kBq/cm². The tritium levels on the outer first wall tiles were higher than those for the inner first wall tiles. And the outer midplane showed the highest level among the first wall tiles.
Fig. 4 Comparison of (a) tritium profiles for selected first wall graphite tiles and (b) flux of higher energy triton impinging to first wall by OFMC calculation.

It is interesting to know that the tritium profiles in the divertor tiles did not show a clear correlation with the deposited layers. Figure 5 compares tritium profiles with and without deposited layers on the inner divertor tile. The upper right image is a cross-sectional SEM view of the tile, indicating about 20 μm of deposition layer. The deposited layer of the bottom half of the inner divertor tile was exfoliated by an adhesive tape. One can clearly see a higher tritium level behind the deposited layer. This suggests that tritium was implanted even in those areas with less tritium implantation with the ripple loss mechanism, in other words, the tritium is not fully thermalized before impinging the target plates.

4. Discussion

As observed in the periodic variation of the full toroidal tritium distribution on the dome top tiles (see Fig.3), tritium is very likely retained as the high energy triton implanted through the ripple loss mechanism. Agreement of the measured tritium profiles with the impinging fluxes calculated by OFMC for the divertor region, (as shown in Fig. 2(c)), and the midplane tiles among the first wall tiles (as shown in Fig. 4(b)) is also consistent with the ripple loss mechanism, since the OFMC takes account the toroidal filed ripples

In contrast to the dome top tiles, the tritium levels in the divertor tiles were quite low. In particular, both the outer and inner strike points showed the lowest level. The temperature increase of graphite tiles due to the plasma heat load was around 50-100K except in the divertor regions where the maximum temperatures of 800-1200K were recorded at the inner and outer divertors, respectively. One also notes that the tritium level showed a gradient in the poloidal direction. This gradient was inversely correlated to the poloidal temperature distribution, indicating that the implanted tritium was thermally released, resulting in no tritium retention in the divertor legs [8].

Furthermore, it is also important to note that the tritium profiles in the divertor tiles did not show a clear correlation with the deposited layers as shown in Fig. 5. This is quite different from JET,
where the highest tritium level was observed in the redeposited layer, particularly, on the plasma shadowed area [9,10]. In JT-60U, deposited layers were found mainly on the inner divertor tiles, and the deposited layer can not be clearly distinguished from the substrate [11]. This may be due to high temperatures during discharges, which seems to enhance the adhesion property of the deposits on the matrix with smaller H and D content than other large tokamaks [12].

![Redeposited layer SEM photograph](image)

Surface of the redeposited layer
The exfoliated region

**Fig. 5** Comparison of tritium profiles with and without deposited layers on the inner divertor tile. The upper right image is a cross-sectional SEM view of the tile, indicating about 20µm of deposition layer. The deposited layer of the bottom half of the inner dome wing was exfoliated by an adhesive tape, indicating higher tritium level in the matrix than in the deposited layer.

Finally according to the analysis of hydrogen and deuterium depth profiles measured for the same divertor tiles used in the tritium analysis [12], hydrogen and deuterium in JT-60U are distributed quite similarly with tritium, i.e., the area with lower tritium retention also shows lower hydrogen and deuterium retention. Since all three isotopes behave similarly during thermal release above 800K, this can be viewed as another indication that the temperature effect dominated tritium retention in the divertor region of JT-60U.

Thus in essence, we are now able to explain the observed tritium distribution in JT-60U divertor tiles by the combination of the implantation of high energy tritium and the simultaneous thermal release due to the heat load [4,13].

**5. Conclusions**

Tritium distributions on the graphite divertor tiles, the dome units and the baffle plates of JT-60U were successfully measured. The highest tritium level was found at the dome top tiles and the outer baffle plates, where the plasma did not hit directly. Such high tritium retention in the dome units and the baffle plates can be well explained by the energetic triton particle loss due to the ripple loss mechanism.

According to the orbital simulation code of OFMC, about 1/3 of tritons produced by D-D reaction impinge plasma facing surfaces without fully losing its energy. In particular, the impinging flux is high on the dome area and the baffle plates. Although the toroidal tritium profiles on each tiles
appeared symmetrical, detailed profile in full toroidal direction clearly showed a periodic variation in
the direction of the toroidal magnetic fields, confirming the ripple loss of high energy triton as
suggested by the OFMC code.

In addition, the tritium retention in divertor tiles heated above 800K was actually very small,
indicating that the temperature increase owing to the plasma heat load results in the release of the
once retained tritium.

The present IP imaging could have missed some tritium adsorbed or absorbed in near surface layer
from low energy impinging after losing energy in the plasma, because surface tritium can be easily
replaced by hydrogen and water molecules

Acknowledgement

This work was carried out under joint research project of JAERI and Japanese Universities for
plasma-surface interactions in JT-60U and was partly supported by a Grant-in-Aid for scientific
research by The Ministry of Education, Culture, Sports, Science and Technology of Japan. The
authors would like to acknowledge those who made the joint project possible in JAERI and to thank
the JT-60 team for their contribution to the operation and the experiment.

References

Tritium Distribution on Plasma Facing Graphite Tiles of JT-60U


- Center for Integrated Research in Science and Engineering, Nagoya University, Furo-cho, Chikusa-ku, Nagoya 464-8603, Japan.
- Department of Nuclear Engineering, Graduate School of Engineering, Nagoya University, Furo-cho, Chikusa-ku, Nagoya 464-8603, Japan.
- Japan Atomic Energy Research Institute, Nakamachi, Naka-gun, Ibaraki 311-0193, Japan

Abstract:
Tritium distributions on the graphite divertor tiles, the dome units and the baffle plates of JT-60U were successfully measured. Poloidally, the highest tritium level was found at the dome top tiles and the outer baffle plates, where the plasma did not hit directly. On the other hand, although the toroidal tritium profiles on each tiles appeared uniform, detailed profiles in full toroidal direction clearly showed a periodic variation corresponding to the position of the magnetic field coils, indicating the ripple loss of high energy tritons as suggested by the OFMC code. Finally, the temperature increase owing to the plasma heat load was found to release the once retained tritium.

Calculation of ripple loss of high energy triton by OFMC code

Y. Gotoh et al. Fusion Eng. Design to be published
**Tritium on the first wall**

- Higher tritium on bottom tiles
- Higher tritium on outer wall
- Very high tritium on midplane of outer wall

**Tritium beneath the deposited layer** (Inner divertor tile)

(Tritium was implanted)

- Exfoliate some redeposited area on inner divertor tile SDV7ap
  - Redeposited layer with thickness of ~20 μm (Max 60 μm)

**Tritium on the first wall**

- Higher tritium on bottom tiles
- Higher tritium on outer wall
- Very high tritium on midplane of outer wall

**Full toroidal distribution of dome top tiles**

- Total 240 pieces of tiles
- Relation between tritium distribution and toroidal magnetic field coil

Particle flux by OFMC

Surface of the redeposited layer

The exfoliated region

**Full toroidal distribution of dome top tiles**

- Toroidal direction

Relative radial tritium concentration (×10^4 ppm)

Toroidal angle [degrees]
Conclusion
(1) Tritium distributions on the graphite divertor tiles, the dome units and the baffle plates of JT-60U were successfully measured. The highest tritium level was found at the dome top tiles and the outer baffle plates, where the plasma did not hit directly. Such high tritium retention in the dome units and the baffle plates can be well explained by the energetic tritium particle loss due to the ripple loss mechanism.

(2) According to the orbital simulation code of OFMC, about 1/3 of tritons produced by D-D reaction impinge plasma facing surfaces without fully losing its energy. In particular, the impinging flux is high on the dome area and the baffle plates. Although the toroidal tritium profiles on each tiles appeared symmetrical, detailed profile in full toroidal direction clearly showed a periodic variation in the direction of the toroidal magnetic fields, confirming the ripple loss of high energy tritons as suggested by the OFMC code.

(3) The tritium retention in divertor tiles heated above 800K was actually very small, indicating that the temperature increase owing to the plasma heat load results in the release of the once retained tritium.

(Implantation and thermal release)

Comparison of JT-60U and JET
2.4 Fast Particle Destabilization of TAE Type Modes in NSTX, JT-60U and Proposed Burning Plasma Devices.¹


1) Princeton Plasma Physics Laboratory, Princeton University, Princeton, NJ, USA
2) Naka Fusion Research Establishment, Japan Atomic Energy Research Institute, Japan
3) Institute for Fusion Studies, University of Texas at Austin, TX, USA

E-mail contact of main author: fcheng@pppl.gov

Abstract. The properties of fast ion driven TAEs are studied in National Spherical Torus Experiment (NSTX), JT-60U, and proposed DT burning plasma experiments in International Tokamak Experimental Reactor (ITER), FIRE, and IGNITOR, and JET DT operation. Unstable TAEs have been observed in NSTX and JT-60U experiments to cause significant fast ion loss. Theoretical studies employing the global kinetic-MHD stability codes, NOVA/NOVA-K code, and a high-n non-perturbative HINST code have been performed, and the analysis explains well the experimental results. For the proposed DT burning plasma experiments TAEs are expected to be unstable in ITER and FIRE.

1. Introduction

Of major importance in burning plasma studies is the fast ion confinement. In burning plasmas, 3.5 MeV α-particles and other fast ions transfer their energy to the background plasma. Collective instabilities destabilized by fast ion pressure gradient can cause premature loss of fast ions from the confinement system. Theory and experiment have confirmed that large amplitude Toroidicity-Induced Alfvén Eigenmodes (TAEs) [1–3] can lead to expulsion of fast ions, degrade ignition margin and produce localized heating or damage on plasma facing components. The TAE frequency is \( \omega \sim V_A/2qR \), where \( V_A \) is the Alfvén velocity, \( q \) is the safety factor, \( R \) is the major radius. TAEs can resonate with energetic particles with \( V_h \sim V_A \) and can be destabilized by the energetic particle phase space gradient (pressure gradient and positive energy gradient). For a slowing-down energy distribution, fast particles can drive TAEs if \( nq(V_h/V_A) \geq \tau L_p/R_Pn \), where \( n \) is the toroidal mode number, \( \tau \) is the minor radius, and \( \rho_h, L_p \) are the hot ion gyroradius and pressure gradient scale length, respectively. Thus, the device size, which determines \( L_p/\rho_h \) and aspect ratio, is an important parameter in determining the unstable TAE spectrum. However, TAEs can be unstable only if the fast particle drive overcomes damping effects of bulk electron and ion Landau damping, radiation and continuum damping.

In the paper we address the properties of fast ion driven TAE type modes in NSTX and JT-60U and in the proposed burning plasma experiments of ITER, FIRE, IGNITOR, and JET-DT operation. For NSTX we present TAE results observed even for a modest NBI power and compare the experimental results with theoretical analysis. For JT-60U we present theoretical interpretation for TAEs destabilized by NNB1 fast ions in normal shear discharges. For the proposed burning plasma experiments we present TAE stability analysis and summarize the parameter domain where TAEs are expected to be unstable due to α-particles produced in D-T fusion reaction. The stability of TAE-type modes is analyzed by employing global kinetic-MHD stability codes, NOVA/NOVA-K codes [4], and a high-n non-perturbative HINST code [5].

¹ This work is supported by DoE contract No. DE-AC02-76CH03073 at PPPL
2. TAEs in NSTX

TAE modes have been observed in NSTX even for modest NBI power. Because of low aspect ratio the toroidal coupling effect is strong. The Alfvén continuum gap is wide open across the minor radius and a broad spectrum of TAEs can exist [6] for each toroidal mode number \( n \). A variety of modes in the frequency range from 20 to 150 kHz with toroidal mode numbers from \( n = 1 \) to \( n = 6 \) are commonly seen in beam heated discharges. FIG. 1(a) shows the Mirnov magnetic field fluctuation and corresponding total neutron count rate and \( H_\alpha \) signal in the NSTX shot 108530. FIG. 1(b) shows a broad spectrum of TAEs observed with several frequencies for each \( n \). The plasma parameters for this shot are \( B_0 = 0.43 T \), \( R = 87 cm \), \( a = 63 cm \), ellipticity \( \epsilon = 1.74 \), triangularity \( \delta = 0.5 \), and the deuterium neutral beam is injected co-tangentially to the plasma current (but counter to the toroidal field) with an injection energy of 80 keV. At 0.09 sec the beam source A is injected with 1.6 MW power, and at 0.21 sec a second beam B is injected and the total NBI power is 3.2 MW and the steady state plasma current is about 0.65 MA. In addition to the more commonly observed quasi-continuous TAEs, bursting TAEs (indicated by vertical dashed lines in FIG. 1(b)) are also observed after the beam B is injected at 0.21 sec. These bursting modes are associated with fast neutron drops, \( H_\alpha \) micro-bursts, and 5 - 10% fast ions hitting the wall. Each burst consists of multiple modes with \( n \) range from 2 - 5 with a dominant mode being \( n = 2 \) or 3. From the bursting mode amplitude modulation one sees beating of multiple modes, but the bursting fluctuation is dominated by a single frequency mode with mode growth and decay times approximately 50 - 100 \( \mu s \). The TAEs are most commonly present in the early phase of the discharge, during the current ramp and when the density is low \( (2 - 3 \times 10^{13} cm^{-3} \) on axis). The higher \( n \) modes are generally associated with higher frequencies, possibly related to increased Doppler shift due to plasma rotation. However, the mode spacing is not nearly as uniform.

FIG. 1(c) shows a very rich TAE frequency spectrum and the radial extent (half width) of TAEs for \( n = 1 - 5 \) modes computed by the NOVA/NOVA-K code. The NSTX equilibrium is constructed with plasma profiles modeled by the TRANSP code at 0.267 sec, \( q_a = 11.4 \), \( n_a(0) = 2.54 \times 10^{13} cm^2 \), \( \beta(0) = 21.4\% \), \( \langle \beta \rangle \sim 2.88\% \). Note that for each \( n \) there are multiple TAEs. Only TAEs with frequency residing in the TAE continuum gap are chosen so that they do not suffer continuum damping. Higher frequency FAEs are excluded in the plot. Also, we restrict TAEs with peak amplitude located at radial location \( r/a \leq 0.8 \). Because the plasma toroidal rotation velocity is significant with \( V_{rot}(0) \approx 170 km/s \) at 0.27 sec and has a peak profile, the frequencies shown in FIG. 1(c) are the NOVA computed TAE frequencies Doppler-shifted by \( f = f_{TAE} + n f_{rot} \), where...
$f_{rot}$ is weighted by the square of the mode amplitude and is averaged over the mode half width. With $V_{rot} = 100 km/s$ and $R = 1 m$, we have $f_{rot} \approx 16 kHz$. These $n = 1 - 5$ TAEs with peak amplitude located inside $r/a \leq 0.5$ can be destabilized by NBI ions, in good agreement with the NSTX experimental results shown in FIG. 1(b). The calculated linear growth rates are about 10% of the real frequency for $\beta_n(0) \approx 13\%$.

![Graphs](image)

FIG. 2: TAEs observed in the JT-60U shot E32359.

3. TAEs in JT-60U

In JT-60U experiments, very rich TAE-type mode activities have been observed [7, 8]. In the Negative-ion-based Neutral Beam Injection (NNBI) (injection energy at $\leq 400keV$) experiments with normal magnetic shear, three types of modes have been found as shown in FIG. 2 for the E32359 shot. The NNB fast ion parameters are $V_i/V_A \leq 1.5$, $0.1\% \leq \langle \beta_i \rangle \leq 1\%$, which are similar to $\alpha$-particle parameters in ITER with $V_i/V_A \leq 1.5$, $\langle \beta_i \rangle \approx 0.2\%$. The first type is the quasi-continuous modes that can last through out NNB injection and is explained as TAEs with frequency variation due to variation of the profiles of magnetic safety factor and plasma density. The second type is slow frequency upward sweeping modes that last for $100$s of msec (from 3.8 to 4 sec in FIG. 2(a)) and is explained as resonant TAE (RTAE) [9]. These modes are RTAEs because their frequencies at the initial chirping stage are inside the Alfvén continuum and they occur when the fast particle drive is strong, so that these modes can overcome the continuum damping. The reason for the slow frequency chirping is the evolution of the fast particle pressure profile and corresponding shift of the RTAE radial location accompanied by the change in the mode frequency during the fast particle pressure buildup. Similar slow frequency chirping modes were also modeled by the HINSTE code to explain the observed changes of the mode frequency in TFTR. The third type is bursting modes (or called the Abrupt Large-amplitude Events (ALE) [8]) that last typically less than 0.5 msec, which are much shorter than the time scale of equilibrium profile evolution. The bursting mode frequency remains almost unchanged during bursting, but the amplitude varies very fast in a few wave periods as shown in FIG. 2(c), which is caused by the change in the fast particle velocity and/or spatial distribution during the nonlinear phase. We construct a JT-60U equilibrium at 4.3 sec with $R = 3.37 m$, $a = 83cm$, $\epsilon = 1.49$, $\delta = 0.15$, $B_0 = 1.2T$, $q_\alpha = 4.7$, $n_e(0) = 2 \times 10^{13} cm^3$, and $\langle \beta \rangle = 0.74\%$. The NOVA/NOVA-K code calculation gives $71 kHz$ for an $n = 1$ TAE, $66 kHz$ for an $n = 2$ TAE. If we include the downward Doppler frequency shift due to the plasma rotation (assuming $V_{rot} \approx 150 km/s$) resulting from the co-tangential injection of NNB and PNB to the plasma current and the toroidal field, the computed TAE frequencies are consistent with the observed mode frequencies for $n = 1$ and 2 TAE modes.

Significant fast ion loss has been found during the presence of bursting modes, in particular, during ALEs. The fast ion loss was seen indirectly in both drop in neutron emission rate and enhancement of neutral particle flux measured by the neutral particle analyzer (CX-NPA). Because about 90% of neutron emission rate is from beam-target reaction in
the NNB experiments, the sudden drop in neutron emission rate during bursting modes can only be caused by either rapid deceleration of NNB ions or by a radial transport of the beam ions to a lower background density region. But, the beam ion slowing-down time in the plasma core region is about 0.5 sec, which is much longer than the time scale of neutron emission rate drop. Moreover, from the energy spectra of the fast neutral flux, the peak flux enhancement is at about 260 keV, which is the energy of beam ions resonating with the bursting modes. These lead to the conclusion that lost fast ions are ejected while resonating with bursting modes. Moreover, the neutron flux signals increase in the peripheral region \((r/a > 0.48)\) and decrease in the central region \((r/a \leq 0.34)\) during the occurrence of ALE bursting modes, which suggests that ALE bursting modes cause a large radial transport of fast ions.

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<th>(a/m)</th>
<th>(B_0), T</th>
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<th>(T_{eo}, keV)</th>
<th>(\sigma)</th>
<th>(\beta_{a0}, %)</th>
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<th>(V_{h}/V_{A0})</th>
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<td>0.8</td>
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<tr>
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<td>2.3</td>
<td>1.66</td>
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</table>

**TABLE 1: KEY PLASMA PARAMETERS FOR BURNING PLASMA DEVICES.**

4. **TAEs in Proposed Burning Plasma Experiments**

The TRANSP code is employed to model plasma conditions for the proposed burning plasma experiments of ITER, FIRE, IGNITOR and JET-DT operation. Despite noticeable differences in the device size, most dimensionless plasma parameters appear to be quite similar for these burning plasma as seen from TABLE I, where \(\sigma = (n_D + n_T)/n_e\) is the plasma ion depletion factor, \(\rho_{a0}\) is the alpha gyroradius at the birth energy with \(B_0\). The HINST [5] is used to analyze the local TAE stability. As expected, TAEs can be driven by \(\alpha\)-particles for all these burning plasma experiments with growth rate on the order of a few percent of frequency [10]. The mode spectrum and their radial location are different for different machine sizes. For FIRE, IGNITOR and JET-DT the unstable spectrum is in the range of \(1 \leq n \leq 10\). For ITER the of unstable spectrum has higher \(n\) with \(5 \leq n \leq 12\) due to a larger value of \((\rho_{a0}/a)\sqrt{\beta_{a0}}\).

FIG. 3 shows the frequency and growth rate versus the normalized toroidal flux \(\sqrt{\Phi}/\Phi_0\) for (a) \(n = 10\) TAEs in ITER, (b) \(n = 7\) TAEs in FIRE, and (c) \(n = 5\) TAEs in JET, where TRANSP generated q-profile as well as the model q-profile, but otherwise parameters obtained from TRANSP was used. Note that for ITER, IGNITOR case is not shown because TAEs turn out to be robustly stable, though sometimes close to the marginal stability, because of weak \(\alpha\) drive due to low \(\beta_a\) and strong trapped electron collisional damping due to high density. For ITER, from FIG. 3(a), the instability region lies within \(0.35 < \sqrt{\Phi}/\Phi_0 < 0.7\), which is primarily due to the lower damping at \(\sqrt{\Phi}/\Phi_0 \approx 0.5\). Note that in ITER high energy D beams at 1 MeV energy are planned for better penetration into the plasma, and they can provide additional instability drive for TAEs for a wide range of \(n\)-values due to the large value of \(\omega_{\alpha}\) term and the strong anisotropy of the beam ion velocity space distribution. From the global NOVA-K code calculation, the unstable TAE spectrum expand towards higher \(n\)-values due to 1 MeV beam ions, and the beam ion TAE drive is similar to the alpha drive. For lower energy beam ions the drive will be weaker and at 500 keV energy the beam ion drive is reduced by one half.

For FIRE, with the TRANSP calculated central value of \(\beta_{a0} = 0.28\%\), the radial span of the TAE unstable region lies within \(0.5 < \sqrt{\Phi}/\Phi_0 < 0.65\) and the growth rate sharply
FIG. 3: TAE eigenfrequency and growth rate as functions of the minor radius in (a) ITER \((n = 10)\), (b) FIRE \((n = 7)\), and (c) JET \((n = 5)\) for TRANS and a model q-profiles.

decreases outside that region as shown in FIG. 3(b). For JET-DT plasmas the maximum growth rate, without NBI ions, is rather low at \(n \simeq 6\) as shown in FIG. 3(c), which is close to what was predicted in other studies \([11]\). Including NBI fast ions, an additional strong stabilizing effect is present due to beam ion Landau damping. We computed the TAE growth rate for a deuterium NBI beta \(\beta_0(0) = 0.6\%\) at 100 \(keV\) injection energy, we found at \(\sqrt{\Phi/\Phi_0} = 0.5\) the \(n = 5\) TAE is stabilized primarily due to damping on beam ions, consistent with the previous study \([11]\).

Finally, because TAEs typically have a global structure, a more accurate stability calculation will require taking an appropriate average over the minor radius. The nonlocal calculation has been performed with the NOVA/NOVA-K codes, and TAEs are usually found to be more stable than from the HINST code. However, for ITER and FIRE TAEs are still expected to be unstable.

5. Summary

In summary, we have performed studies of TAEs for presently operating devices, NSTX and JT-60U, as well as for the proposed DT burning plasma experiments of ITER, FIRE, IGNITOR and JET-DT operation. For NSTX and JT-60U bursting type TAEs are found to cause significant fast ion loss. Theoretical investigations yield TAE frequencies and stability, consistent with experimental observations. For the proposed DT burning plasma experiments, the global and local calculations predict that TAEs would be unstable for ITER and FIRE.

TH/7-1Rb:
Fast Particle Destabilization of TAE Type Modes in NSTX,
JT-60U and Proposed Burning Plasma Devices
C. Z. Cheng 1)


19th IAEA Fusion Energy Conference, Lyon, France, October 14-19, 2002

Outline

1. Rich TAE spectrum observed in NSTX and theoretical analysis.
2. Chirping frequency and bursting modes in JT-60U in the TAE frequency range.
3. TAE study in burning plasma proposals: ITER, FIRE and IGNITOR.

TH/7-1Rb:
Fast Particle Destabilization of TAE Type Modes in NSTX,
JT-60U and Proposed Burning Plasma Devices

1) NSTX shot with \( B = 0.434 \ T, R_0 = 87 \ cm, a = 63 \ cm, P = 3.2 \ MW \).
2) \( n = 1 - 5 \) TAEs usually observed.
3) Bursting TAEs observed with \( 5 - 10\% \) fast ion loss after \( 0.21 \) msec.
4) Burst TAEs dominated by single mode with burst time \( \approx 200 \) \( \mu \)sec.

TH/7-1Rb:
NOVA predicts frequencies of unstable TAEs in agreement with observations

1) Unstable TAEs with \( n = 1 - 6 \).
2) Higher \( n \)'s are stabilized due to finite orbit width and Larmor radius.
3) Growth rates are \( \gamma/\omega \approx 1 - 4\% \).
4) Unstable modes are global and peaked at \( \tau/a \approx 0.5 - 0.7 \).
JT-60U N-NBI injection shows strong bursting TAE activity

- JT-60U: \(\beta_n = 0.1 - 1\%\), \(v_\|/v_A \leq 1.5\) for N-NBI ions
- Slow frequency chirping modes observed during 3.8 - 4.2 sec is explained as RTAEs
- Bursting TAEs observed after 4.2 sec
- Bursting TAEs dominated by single mode
- NOVA predicts bursting TAE frequency \(f = 71 \text{kHz}\) for \(n = 1\), and \(f = 66 \text{kHz}\) for \(n = 2\), in agreement with observations after \(f_{\text{cd}} \approx 7 \text{kHz}\) Doppler shift is considered.

JT-60U bursting TAEs lead to fast ion transport and loss

- Fast ion loss observed indirectly in both neutron flux drop (~10%) and neutral particle flux enhancement (Shinohara, 2002)
- Peak neutral particle flux enhancement at about 200keV, energy of beam ions resonating with bursting TAEs
- Neutron flux signals increase in \(r/a \geq 0.48\) and decrease in \(r/a \leq 0.34\) during bursting TAEs
- Bursting TAEs cause large fast ion radial transport

Planned Burning Plasma Devices: ITER, FIRE and IGNITOR

TAEs in ITER, FIRE and IGNITOR burning plasmas (BP)

- Plasmas were modeled with TRANSP plasma analysis code (BUDNY, NF '02).
- Use the same \(q\)-profile, \(q = 1 + 2.8(\psi/\psi_0)^{3/2}\).

1. Kinetic nonperturbative local code HIINST.
2. Ideal MHD global NOVA code and perturbative kinetic NOVA-K code are employed (GORELENKOV, submitted to NF, 2002).
HINST predicts local TAE instability in ITER and FIRE

- HINST includes α drive, ion, electron Landau dampings, radiative damping, electron collisional damping.
- Unstable TAE spectrum is shifted toward high n's in ITER
  - ITER is strongly unstable
  - FIRE is weakly unstable
  - JET is stable if beam damping is included
  - IGNITOR is robustly stable

NOVA-K predicts global AE instability in ITER and FIRE

- In FIRE with high temperature $T_e = 22 keV$, triangularity induced AEs are found unstable due to lower Landau damping.
- In ITER 1MeV tangentially injected beam ion drive is comparable with the alpha drive.
  - Lowering NBI energy to 0.5MeV reduces drive.
  - 0.5MeV energy is enough for good beam penetration.

Summary

1. Bursting TAEs were found to cause significant fast ion losses in NSTX and JT-60U.
2. Theoretical investigation yields TAE frequency and stability consistent with observations.
3. For the burning plasmas HINST and NOVA predict AEs to be unstable in ITER and marginally unstable in high temperature ($T_e = 22 keV$) FIRE plasmas.
2.5 High Mach Flow Associated with Plasma Detachment in JT-60U


1) Keio University, 3-14-1 Hiyoshi, Kouhoku-ku, Yokohama 223-8522, Japan
2) Japan Atomic Energy Research Institute, Ibaraki, Japan
3) Max-Planck-Institut für Plasmaphysik, Greifswald, Germany
4) Max-Planck-Institut für Plasmaphysik, Garching, Germany

e-mail contact of main author: akh@ppl.appi.keio.ac.jp

Recent new results of the high Mach flows associated with plasma detachment are presented on the basis of numerical simulations by a 2-D edge simulation code (the B2-Eirene code) and their comparisons with experiments in JT-60U W-shaped divertor plasma. High Mach flows appear near the ionization front away from the target plate. The plasma static pressure rapidly drops, while the total pressure is kept almost constant near the ionization front, because the ionization front near the X-point is clearly separated from the momentum loss region near the target plate. Redistribution from static to dynamic pressure without a large momentum loss is confirmed to be a possible mechanism of the high Mach flows. It has been also shown that the radial structure of the high Mach flow near the X point away from the target plate has a strong correlation with the DOD (Degree of Detachment) at the target plate. Also, we have made systematic analyses on the high Mach flows for both the “Open” geometry and the “W-shaped” geometry of JT-60U in order to clarify the geometric effects on the flows.

1. Introduction

To control plasma flows in the SOL and divertor region is one of the most important issues for the steady-state operation of the future fusion reactors. Plasma flows in the SOL and divertor region affect divertor performances in many aspects, such as, impurity shielding, helium exhaust, divertor in-out asymmetry, main plasma recycling, etc.

High parallel flows associated with plasma detachment have been observed in several tokamak experiments[1,2]. Large Mach flows up to Mach 1 or even larger have been measured near the X-point away from the target plate in these experiments. (Henceforth, abbreviation “HMAD” will be used for such high Mach flows in the detachment state.)

In Ref.[3], a 2D numerical study of HMAD by using the B2-Eirene code package[4-6] was done for the “Open” divertor geometry in JT-60U. To understand the physical mechanism of HMAD, detailed comparisons of the numerical results with those by a simple 1D analytic model were also made in Ref.[3]. Redistribution from static pressure to dynamic pressure without a large momentum loss has been shown to be a possible cause of HMAD observed in the numerical simulations. However, for the Open divertor geometry, flow measurements in the divertor region were not made. It was impossible to make the direct comparisons of the numerical results with the experimental results.

Recently, flow measurements with the fast movable Mach probe near the X-point have been made for the “W-shaped” divertor geometry in JT-60U[2]. To verify the physical mechanism discussed in Ref.[3] and to obtain more robust conclusions, comparisons with the experimental measurements are indispensable. In the present study, we have done the numerical calculations for the W-shaped geometry and their direct comparisons with the experimental results are made. In addition, geometric effects on HMAD are studied by comparing the numerical results for the W-shaped geometry with those for the Open geometry.
2. Numerical Model

Typical L-mode discharges for the Open divertor (Open-Div) and the W-shaped divertor (W-Div) geometry with similar main plasma parameters were chosen to evaluate the geometric effects on HMAD. Figure 1 shows the numerical grid near the divertor region for (a) the Open and (b) the W-shaped geometry. Bulk ion species $D^+$, all carbon impurity ion species $C^+ - C^{6+}$, and neutral species $\text{D}, \text{D}_2, \text{C}$ are considered in the analysis. At the core interface boundary, i.e., at the innermost flux surface of the grid inside the separatrix in Fig.1, the bulk ion density $n_D$ and the total input power $P_{in}$ are specified. For the boundary conditions of the target plate, the usual Bohm condition is used. The remaining simulation models/conditions, such as transport model, are almost the same as those in Ref.[3]. To simulate the attached state and the detached state, $n_D$ has been changed for each numerical run, while the remaining conditions are kept fixed.

![Fig.1 Numerical grid near the X point in the divertor region: (a) Open divertor geometry (Open-Div) and (b) W-shaped divertor geometry (W-Div).](image)

3. HMAD in the W-shaped Divertor Geometry and Its Physical Mechanism

Figure 2 shows 2D spatial profiles of the parallel flow velocity $u_{\parallel}$ for $D^+$ near the X-point in the W-shaped divertor. The spatial profiles are compared between (a) the attached plasma case ($n_D=1.0 \times 10^{19}$ m$^{-3}$) and (b) the detached plasma case ($n_D=2.0 \times 10^{19}$ m$^{-3}$). The total input power ($P_{in}=2.5$ MW) is the same for both cases. The flow velocity is shown as the local Mach number ($M \equiv u_{\parallel} / C_s$), i.e., $u_{\parallel}$ is normalized by the local isothermal sound speed $C_s$. In Fig.2, the positive direction of the velocity is defined as the direction from the inner divertor plate to the outer divertor plate in the edge plasma region. Thus, the negative sign means the flow is directed towards the inner target plate, while the positive sign means it is directed towards the outer plate.

![Fig.2 2D spatial profiles of parallel Mach number in the divertor region for the W-Div.](image)
In the attached plasma case, the Mach number in the bulk of divertor region is still low as shown in Fig.2(a). The Mach number reaches $M \sim 1$ only near the target plate. On the other hand, in the detached plasma case, high Mach flows appear near the X-point away from the target plate.

To understand the formation mechanism of HMAD in Fig.2(b), basic divertor characteristics are compared between (a) the attached state and (b) the detached state in Fig.3-Fig.5. Typical 2D profiles of $T_e$ and ionization source $S_i(D^i$ions/m$^3$/s) are shown, respectively, in Fig.3 and Fig.4.

Fig.3 2D profiles of electron temperature $T_e$ in the divertor region for the W-Div.

Fig.4 2D profiles of the ionization source density $S_i$ in the divertor region for the W-Div.

Fig.5 2D profiles of the momentum loss density $S_m$ in the divertor region for the W-Div.
In the attached case, $T_e$ is still high in the divertor region and $S_i$ is localized near the target plate. On the other hand, in the detached case, $T_e$ drops rapidly towards the target plate and becomes $T_e$ $<$ 5eV in front of the target plate. Due to this large decrease in $T_e$, the ionization front moves away from the target plate as shown in Fig.4(b). The HMAD region in Fig.2(b) is almost coincident with the ionization region in Fig.4(b) where the static pressure drops strongly due to the large decrease in $T_e$. Figure 5 shows 2D profiles of the momentum loss $S_m$ (N/m²) for D⁺ ion fluid due to the interaction with neutrals, e.g., CX-collision. In the detached case, it should be noted that the region where $S_m$ is large in Fig.5(b) is almost separated from the ionization region in Fig.4(b). As a result, the total pressure is kept almost constant along the field line near the ionization region away from the target. Thus, the pressure gradient force due to the large drop of the static pressure possibly drives HMAD near the ionization front, i.e., redistribution from static pressure into dynamic pressure is a possible cause of HMAD observed in the simulation.

4. Comparison with Experiments and Effect of Divertor Geometry on HMAD

The radial profiles of the parallel Mach number are shown, respectively, in Fig.6 (a) for the Open-Div and Fig.6(b) for the W-Div. The $M$-profiles are plotted along the path shown in Fig.1(a) and (b) by arrows. The following interesting common features can be seen; 1) as the separatrix electron density $n_{sep}$ at the mid-plane increases, the Mach number becomes larger, 2) the Mach number first starts increasing near the separatrix and then the peak moves radially outward, and finally, 3) the peak value becomes quite large ($M \sim 1$). In the W-shaped geometry, the radial profiles of the parallel flow were measured by the fast movable Mach probe near the X point[2]. The measurements were done along almost the same path in the numerical simulation. The experimental results are shown in Fig.6(c) for each line average density $\bar{n}_e$ of the main plasma. The Mach number is estimated from the probe data by using the Hutchinson's formula[8].

The qualitative features obtained in the numerical simulation, i.e., 1), 2) and 3) described above, agree well with the experimental results in Fig.6(c). For the largest $\bar{n}_e$ in Fig.6(c), the impurity radiation near the X point is enhanced (X-point MARFE) and the detachment region is extended more radially outward from the separatrix in comparison with the case of $\bar{n}_e$ = 2.6 $\times 10^{19}$ m⁻³. Also in the simulation, X-point MARFE appears for a larger $n_{sep}$ than in Fig.6(b) and the peak of the $M$-profile moves further outward and the peak value becomes larger.

The radial $M$-profiles for the Open-Div and the W-Div in Fig.6(a) and Fig.6(b) have a close relation to the detachment characteristics. To make a discussion more quantitative, Fig.7 compares the radial profiles of the DOD [7] at the target plate. The DOD value at each point on the target plate is mapped to the upstream point in Fig.1 where the radial $M$-profile is plotted. The DOD is a figure of merit for the particle flux detachment. The DOD
\( \equiv \frac{Cn_{\text{sep}}^2}{\Gamma_d} \) is defined by the ratio of the particle flux in the attached state, which scales as \( Cn_{\text{sep}}^2 \) (C is a proportional constant), to the particle flux \( \Gamma_d \) in the detached state. Thus, if the DOD becomes larger than unity, then the detachment starts. The value larger, the detachment becomes deeper. At the low and medium \( n_{\text{sep}} \), the plasma is still attached besides the region very close to the separatrix for the Open-Div, while the detachment has already started in relatively wide region for the W-Div. However, at the highest density case, the DOD profile is more peaked for the W-Div. The radial extent of the high Mach flow with \( M=1 \) for the W-Div is also more peaked than that for the Open-Div as shown in Fig. 6 (a) and (b). In the Open-Div without the baffle plate and the doom structure, recycling neutrals are relatively free and tend to spread out radially. The DOD values for the Open-Div near the separatrix become smaller than those for the W-Div, while they become larger at the outer part of the target plate. As a result, the DOD profile becomes broader for the Open-Div.

\[ \text{(a) Open-Div} \]

\[ \text{(b) W-Div} \]

Fig. 7 Radial profiles of the DOD (Degree of Detachment) at the outer target plate.

5. Conclusions and Future Study

In the numerical simulations for the W-shaped divertor geometry, HMAD appears near the ionization front away from the target plate, where \( T_e \) rapidly drops, as in Ref. [3] for the Open divertor geometry. Direct comparisons with the experiments in the present study strongly support our explanation for the formation mechanism of HMAD proposed in Ref. [3]. In addition, by comparing the radial profiles of DOD at the target plate with those of the Mach number away from the target plate, it is shown that the radial profile of HMAD has a strong correlation with the DOD at the target plate.

However, in the experiments, relatively large Mach flows have been observed even in the attached state. The cause of such relatively high Mach flows in the attached state has not been clearly understood yet. One of the possible causes is effect of various kinds of drift in the SOL and divertor region. These effects are not taken into account the present analysis. In the future, these effects will be taken into account in the analysis.

References

Motivation/Background

- Ion-captured plasma flows in the SOL and divertor region is one of the most important issues for the steady-state operation of the future fusion reactors.
- Plasma flows in the edge region affect the divertor characteristics in many aspects: impurity shielding, divertor in-out asymmetry, helium exhaust, plasma recycling, etc.

Large Mach flows up to Mach 1 or even larger have been measured near the X point away from the target plate.

Henceforth, abbreviation "Mach" will be used for such high Mach flows in the detachment state.

Purpose / Tool / Target

- Purpose of this study
- To understand the formation mechanism of HMAD
- To clarify macroscopic effects on HMAD

- Tools
  - 2D SOL and Divertor Simulation Code Package B2-EIRENE code [3-5]
  - 1D Simple Analytic Model along B

- Analysis of JT-60U SOL and divertor plasma
- Comparisons of numerical results with experiments in JT-60U W-Shaped divertor
- Comparisons of numerical results for the Open divertor geometry and the W-Shaped geometry

Numerical Model for JT-60U Analysis (1)

- MHD equilibrium data for numerical mesh generation

Typical L-mode discharges for the Open divertor (Open-Div) and the W-shaped divertor (W-Div) with similar main plasma parameters were chosen to study the geometric effects on HMAD.

Numerical Model for JT-60U Analysis (2)

- Ion and neutral species:
  - B2 multi-fluids code for ions: D+, C+-C0+
  - Eirene Monte Carlo code for neutrals: D, D2, C

- Boundary conditions:
  at the Core Interface Boundary
  - total input power $Q_{in} = 2.5-2.75$ MW ($Q_{in} = Q_p$)
  - D+ density $n_{D+} = 1.0 - 2.7\times10^{19}$ m⁻³
  at the Target Plates
  - usual Bohm conditions are assumed.
  at the Wall-Side Boundary
  - density and temperature scale lengths are specified.

- Transport model:
  - parallel heat transport: classical with flux limit
  - radial transport: anomalous particle transport: * D = 0.3 m²/s
  - momentum transport: $\eta = \text{nnD}$
  - thermal transport: $\chi = \chi = 2.0$ m²/s

To obtain a reasonable fit to the upstream density profile, (the 1st and 2nd SOL observed in JT-60U experiments), an artificial outward particle flux is taken into account.

$$G_r = -D\frac{\partial n}{\partial r} + n \bar{v}_r, \quad \bar{v}_r > 0 \quad \text{(outward radial flow)}$$

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**Numerical Model for JT-60U Analysis (3)**

Fig. 1(a) Open-Div  
Fig. 1(b) W-Div  

JT-60U geometrical configuration. Numerical grids for the analysis and vacuum vessel. The Doon structures, baffle plates and divertor plates are also shown for the W-Div.

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**Basic Divertor Characteristics (1)**

- Basic comparisons of divertor characteristics between the Open-Div and the W-Div

Fig. 2 Electron temperature $T_e$ profile at the outer target plate for $n_p=1.0 \times 10^{20}$ m$^{-3}$ (high-recycling attached state).

Fig. 3 Electron density $n_e$ profile at the outer target plate for $n_p=1.0 \times 10^{20}$ m$^{-3}$ (high-recycling attached state).

In the W-Div, the $T_e$ profile is broader, while the $n_e$ profile is more peaked than those in the Open-Div. These basic features agree well with the experimental features.

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**Basic Divertor Characteristics (2)**

- Geometric effects on neutral density

Fig. 4 2D profile of neutral density ($D_1$) for the Open-Div ($n_p=1.0 \times 10^{20}$ m$^{-3}$, high-recycling attached state).

Fig. 5 2D profile of neutral density ($D_1$) for the W-Div ($n_p=1.0 \times 10^{20}$ m$^{-3}$, high-recycling attached state).

Due to the closed geometry of the W-Div with the baffle plates and the doom structures, recycling neutrals are effectively supplied from the plate towards the separatrix.

---

**2D Structure of Plasma Flow (1)**

- HMAD in the Open-Div

(a) Attached State  
(b) Detached State

Fig. 8 2D-profile of parallel Mach number ($M=\frac{n_p}{C_s}$)

In Fig. 7 (Open-Div) and Fig. 8 (W-Div), the 2D spatial profile of the flow velocity $u_z$ for D$^+$ fluid are compared between:

(a) attached plasma and (b) detached plasma.

The flow velocity is shown as the local Mach number $M_z$.

In Fig. 7, 8, the positive direction of the velocity is defined as the direction from the inner divertor plate to the outer divertor plate in the edge plasma region.

Thus, the negative sign means the flow is directed towards the inner target plate, while the positive sign means it is directed towards the outer plate.
2D Structure of Plasma Flow (2)

- HMAD in the W-Div

(a) Attached State

(b) Detached State

![Figure 8](image)

Fig. 8 2D-profile of parallel Mach number ($M = u_c/c_d$)

The following common boundary conditions are used for both geometries:

(a) attached plasma case: $n_f = 1.0 \times 10^{17}$ m$^{-3}$

(b) detached plasma case: $n_f = 3.0 \times 10^{17}$ m$^{-3}$

The following common features are seen in Fig. 7 and Fig. 8:

(a) attached plasma case

- $M$ in the bulk of divertor region is still low.
- $M$ reaches $M=1$ only near the target plate.

(b) detached plasma case

- High Mach flows appear near the X-point away from the target plate. (HMAD)

To understand the physical mechanism, basic parameters are compared in the following figures for the W-Div.

Formation Mechanism of HMAD (1)

(a) Attached State

(b) Detached State

Fig. 9 2D profile of electron temperature ($Te$)

![Figure 9](image)

Fig. 10 2D profile of Ionization source density ($S_i$)

In the attached case, $T_e$ is still high in the divertor region and $S_i$ is localized near the target plate. On the other hand, in the detached case, $T_e$ drops rapidly towards the target plate and becomes $T_e \leq \text{eV}$ in front of the target plate. Due to this large decrease in $T_e$, the ionization front moves away from the target plate.

Formation Mechanism of HMAD (2)

(a) Attached State

(b) Detached State

![Figure 11](image)

Fig. 11 2D profile of momentum loss density ($S_m$)

($S_m$ for D$^+$ ion fluid due to the interaction with neutrals, e.g., CX-collision.)

![Figure 12](image)

Fig. 12 Comparison between the ionization region and momentum loss region in the detached state

The ionization region and the momentum loss region are spatially separated.

Simple 1D Analytic Model along B (1)

- Model

![Figure 13](image)

- Basic Eqs.

Region A:

$$\frac{\partial (nu_i)}{\partial s} = 0$$

$$\frac{\partial (mn_iu^2 + p)}{\partial s} = -mn_i v_e u$$

$$T = T_e$$

Temperature at the target

Region B:

$$\frac{\partial (nu_i)}{\partial s} = S_i$$

$$\frac{\partial (mn_iu^2 + p)}{\partial s} = 0$$

$$T = T_e(1 + \frac{s - L_m}{L_T})^2$$


Simple 1D Analytic Model along \( B \)

Integrating basic Eqs. for each region

Region A: \( s : 0 \rightarrow s \)
\[ M = M_t \quad (s = 0) \]
\[ M = 0 \quad (s = L_m + L_q) \]

Region B: \( L_m \leq s \leq L_m + L_t \)
\[ M^2 = \frac{M_t}{M_i} (1 + M^2 + |M_t| \frac{s}{L_m}) + (1 + \frac{s}{L_m})^2 = 0 \]

\[
\frac{1}{M} \frac{dM}{ds} = \frac{A_1(s, M_t) M - A_2(s)}{M^2 - B(s)}
\]

Solutions depend on the parameter

\[ C = \frac{V_s(L_m)}{C_M}, \quad C_{\text{crit}} = \left( \frac{L_T + L_q}{2L_T L_q} \right)^2 - 2 \]

\( C > C_{\text{crit}} \rightarrow \) subsonic solution with \( M = -1 \)
\( C \leq C_{\text{crit}} \rightarrow \) possibility of supersonic flow

Comparison of the results by the 1D analytic model with those by the numerical simulation

1D Analytic Result

Numerical Result

Fig. 14 Subsonic solution with \( M = -1 \)

\[ L_m = 0.2m, L_t = 0.8m, L_T = 0.2m \]

Fig. 15 Supersonic solution

\[ L_m = 0.2m, L_t = 0.8m, L_T = 0.2m \]

Fig. 16 Results of simple 1D analytic model:
(a) the local Mach number, 
(b) the normalized static pressure \( \frac{p}{\rho \Omega^2} \)

Fig. 17 1D profiles of
(a) the local Mach number and
(b) the normalized static pressure as a function of the poloidal distance \( s' \) from the outer target plate along a field line. These plots are obtained from the 2D numerical simulation results.
Formation Mechanism of HMAD

Appearance of the ionization front associated with plasma detachment

Separation of the ionization region from the momentum loss region

Static pressure drops in the ionization region due to temperature decrease, while total pressure is kept almost constant.

Increase in dynamic pressure

HMAD
(High Mach Flows in the Detachment Stage)

Critical Parameter for the onset of HMAD

\[ C = \frac{\tau_p}{\tau_{el}} \rightarrow \] transit time through the momentum loss region
\[ \tau_{el} \rightarrow \] characteristic time for the momentum loss

Comparisons with Experiments (1)

- Experimental measurements in JT-60U W-Div

Fig.18

- Radial profiles of parallel flows were measured by X-point Mach probe [2].

- Parallel Mach number has been estimated by Hutchinson's formula [8]

\[ \frac{\dot{f}_p}{f_p} \text{ (down)} \text{ Ion saturation current on the downstream side} \]
\[ \frac{\dot{f}_p}{f_p} \text{ (up)} \text{ Ion saturation current on the upstream side} \]


Comparisons with Experiments (2)

(a) Numerical Result
(b) Experimental Result

Fig.19 Comparisons of radial M-profiles

- Interesting Common Features
  1) As the separatrix electron density \( n_{e} \) at the mid-plane increases, the Mach number becomes larger,
  2) Mach number first starts increasing near the separatrix and then the peak moves radially outward, and finally
  3) the peak value becomes quite large (\( M \sim 1 \)).

Effects of Divertor Geometry (1)

(a) Open-Div
(b) W-Div

The M-profiles are plotted along the red path.

Fig.20 Effects of divertor geometry on radial M-profiles

The radial M-profiles for the Open-Div and the W-Div shown in Fig.20(a) and Fig.20(b) have a close relation to the DOD (Degree of Detachment) at the target plate.
Effects of Divertor Geometry (2)

(a) Open-Div

(b) W-Div

Fig.21 Radial profiles of DOD at the outer target plate
- DOD : Degree of Detachment [9]

DOD ~ 1: Attached State  DOD > 1: Detached State

low and the medium $n_{ef}$:
- (a) Open-Div : still attached
  besides the region very close to the separatrix
- (b) W-Div : the detachment has already started
  in relatively wide region

the highest density case:
- (a) Open-Div : the DOD profile is more broad
- (b) W-Div : DOD profile is more peaked

Open-Div : Without the baffle plates and the dome structures
recycling neutrals are relatively free and tend to spread out radially.

$\uparrow$ The DOD values become smaller near the separatrix than those for the W-Div, while they become larger at the outer part of the target in this highest density case.


Summary and Future Study(1)

- High Mach flows associated with plasma detachment (HMAD) are studied by the 2-D SOL and divertor simulation code (B2-Elene code).
- Their comparisons with experiments in JT-60U W-shaped divertor plasma.
- A simple 1D analytic model has been also used and the results are compared with the numerical results to understand the formation mechanism of HMAD.
- In addition, effects of divertor geometry on HMAD are studied systematically by comparisons of the results for the Open-geometry with those for the W-shaped geometry.

- HMAD:
  In the numerical simulation under typical L-mode discharge condition of the JT-60U W-shaped divertor geometry, HMAD appear near the ionization front away from the target plate, as in the experiments.

- Formation Mechanism of HMAD:
  Ionization region near the X-point is clearly separated from momentum loss region near the target plate.

  Redistribution from static to dynamic pressure without a large momentum loss is confirmed to be a possible mechanism of HMAD.

Summary and Future Study(2)

- Comparisons with Experiments:
  Qualitative features of the radial profile of HMAD obtained by the 2D simulations agree well with those by the experiments.
  As the main plasma density increases,
  - Mach number first starts increasing near the separatrix
  and then the peak moves radially outward,
  - and finally the peak value becomes quite large ($M - 1$).

- Effects of Divertor Geometry on HMAD:
  - Divertor geometry has strong effects on the radial structure of HMAD.
  - Interesting correlation between the radial HMAD-profile away from the target and the DOD-profile at the target plate has been observed in the simulation.

- Future Study:
  - In the experiments, relatively large Mach flows have been observed even in the attached state.
  - The cause of such relatively high Mach flows in the attached state has not been clearly understood.
  - One of the possible causes is effect of various kinds of drifts and currents in the SOL and divertor.
  - In the future, these effects will be taken into account in the analysis.
2.6 Irradiation test of diagnostic components for ITER application in a fission reactor, Japan Materials Testing Reactor


1) Institute for Materials Research, Tohoku University, Sendai, 980-8577 Japan
2) Japan Atomic Energy Research Institute, Naka, 311-0193 Japan
3) Japan Atomic Energy Research Institute, Tokai, 319-1195 Japan
4) ITER-JWS-Garching, Garching, 85748 Germany
5) CIEMAT, Madrid, 28040 Spain
6) CEA Cadarache, Saint-Paul-lez-Durance, F13108 France
7) SCK/CEN, Mol, B-2400 Belgium
8) TRINITI, Moscow, 142092 Russia
9) GA, San Diego, 92186-4156 USA
10) ITER-JWS-Naka, Naka, 311-0193 Japan

e-mail; shikama@imr.tohoku.ac.jp

abstract. Radiation effects on components and materials will be one of the most serious technological issues in fusion systems realizing burning plasmas. Especially, diagnostic components, which should play crucial roles to control plasmas and to understand physics of burning plasmas, will be exposed to high-flux neutrons and gamma-rays. Dynamic radiation effects will affect performance of components substantially from beginning of exposure to radiation environments, and accumulated radiation effects will gradually degrade their functioning abilities in the course of their services. High-power-density fission reactors will be only realistic tools to simulate the irradiation environments expected in burning-plasma fusion machines such as the ITER, at present. Some key diagnostic components, namely magnetic coils, bolometers, and optical fibers, were irradiation-tested in a fission reactor, JMTR, to evaluate their performances under heavy irradiation environments. Results indicate that the ITER-relevant diagnostic components could be developed in time, though there are still some technological problems to overcome.

1. Introduction

The International Thermonuclear Experimental Reactor (ITER) is the first theater, where diagnostic components will be exposed to intense irradiation environments associated with high-flux high-energy-neutrons. Radiation effects will influence performance of diagnostic components substantially at the onset of fusion nuclear reactions, and successful control and operation of burning plasmas will strongly depend on development of radiation-hardened diagnostic components, and quantitative and qualitative understandings of radiation effects there. Radiation effects in diagnostics-related materials have been extensively studied in the course of ITER-EDA (Engineering Design Activity), effectively coordinated by the corresponding ITER central team [1]. Succeeding to the successful compilation of materials database on radiation effects [1], studies of radiation effects in diagnostic components were launched under international collaborations. Especially for fission reactor irradiation tests, which are time- and resource-consuming, and demanding sophisticated-technologies but are indispensable for development of radiation resistant diagnostic components, several international collaborations were set up. There, the Japanese ITER home team played crucial roles, utilizing Japan Materials Testing Reactor (JMTR) in the Oarai Research Establishment.
of Japan Atomic Energy Research Institute (JAERI), under close collaborations among universities, JAERI and industries.

The JMTR has neutron fluxes and gamma-ray dose rates, similar to those expected near burning plasma regions in ITER. Also, its structure is suitable for in-situ measurements, namely real-time studies of performance of materials and components under a reactor operation. Examples of international collaborations executed in the JMTR are a JUPITER-TRIST-ER (Japan/USA Project on Irradiation Tests Utilizing Reactors, Temperature Regulated In-Situ Test of Electrical Resistivity) project in Japan/USA collaboration for study of radiation effects in electrical insulators [2-4], international round robin tests of radiation resistant optical fibers [5-7], and irradiation tests of magnetic coils under Japan/USA collaboration [8-11] and of bolometers under Japan/EU collaboration [8,12,13]. In the present paper, recent results on performance of key diagnostic components, namely the magnetic coil, the bolometer, and the optical fibers, under the ITER-relevant irradiation conditions, are reported.

2. Irradiation tests of diagnostic components

2.1 Magnetic coil and bolometer

These two components are expected to play crucial roles for controlling plasma with a long burning duration in ITER. The magnetic coil is an essential tool to monitor a magnetic field in the ohmic-heating scenario with long-duration plasma discharges. In the meantime, the strong radiation distribution at the divertor must be known in fine details to control long-duration plasma-discharges free from disruption. The bolometer is the tool to realize this indispensable monitoring.

Several irradiation effects, such as radiation induced electrical conductivity (RIC) and radiation induced electromotive force (RIEMF) will introduce serious disturbances[10,14]. In-situ studies of performance of the magnetic coils revealed that the RIC is not a problem when a coil is made of a mineral insulating cable (MI-cable) [8]. Magnetic measurements could be carried out up to a few MHz under the ITER relevant irradiation conditions in JMTR and the coil survived neutron fluence comparable to that expected in the whole life of ITER. In the meantime, some results showed that effects of the RIEMF may cause serious problems in magnetic-field measurements for a long plasma discharge duration, because it will generate a substantial drift voltage in some occasions [11]. Fig. 1 shows drift voltage in magnetic coils made of 1.5mm outer diameter MI-cable, measured by an advanced digital integrator, during JMTR power-up period. Here, the maximum fast (E>1MeV) and thermal (E<0.683eV) neutron fluxes were 5x10^17/n/m^2s, and 2.5x10^18/n/m^2s, respectively, at a reactor full power of 50MW. A gamma-ray dose rate was estimated 3.5kGy/s for iron at a reactor full power at the peak position in the irradiation rig. The neutron fluxes and the gamma dose rate are nearly proportional to the reactor power when the reactor power changed. Irradiation temperature changed 300K with 0 power of the reactor to above 900K with 50MW reactor power. Drift voltages showed complicated dependence on the reactor power, namely intensity of radiations and their magnitudes were far larger than those expected from electrical circuit analysis with the RIEMF values in simple-configuration MI-cables. Extensive discussions were made among concerned research groups in the ITER-EDA and also stimulated experiments were carried out to check effects of the RIEMF on the drift
voltage, quantitatively. Recent results showed that the drift voltage generated by the ITER-relevant radiation environment was less than $\mu$V and a magnetic coil, satisfying the ITER design criteria, could be developed in time, with selection of appropriate materials and coil configurations and dimensions [10,11]. Concerning materials, aluminum (Al) and copper (Cu) should be excluded from the systems as possible as can be, as they generate short-life beta-emitters. Here, it should be noted that the copper and the aluminum are one of the best electrical conductors and aluminum oxide ($\text{Al}_2\text{O}_3$) is the most popular electrical insulator. Materials composing a sheath in MI-cables will be a major player in generating the RIEMF and heating-up the coil through nuclear heating. Several designs are under consideration, but a coil made of a small diameter (for example 0.5mm outer diameter) MI-cable, which is composed of a stainless steel or nickel-base super alloy sheath, a nickel center lead and a magnesia (MgO) electrical insulator layer, will be one which will decrease the RIEMF as well as decrease the internal impedance of the coil which will resultanty decrease the drift voltage. Also, the fine MI-cable will decrease a nuclear heating rate and improve technical uncertainty caused by localized heating of the coil system. The usage of finer MI-cables will make numbers of turns more and improve sensitivity of the coil. In the meantime, the detailed analysis indicated that the stability of the voltage integrator is another issue to be improved [11].

![Drift voltage extrapolated to those for 1000 seconds measurements.](image)

**Figure 1** Drift voltage extrapolated to those for 1000 seconds measurements.

*Closed rhombus; alumina insulator 0.25mm diameter copper center lead, closed square; magnesia insulator, 0.25mm diameter copper center lead, closed triangle; magnesia insulator 0.5mm diameter copper center lead, open square; magnesia insulator, 0.75mm diameter copper center lead.*

The radiation-hardened bolometer, whose structures were shown in Fig. 2, was developed by modifying a JET-bolometer and was irradiation-tested in the JMTR. There, gold meanders were vapor deposited onto a muscovite ($\text{KAl}_3\text{(Si}_3\text{Al})\text{O}_10\text{(OH,F)}_2$) thin plate. Its performance under ITER relevant irradiation conditions was qualified and quantified as shown in Fig. 3. The resistence of gold meander responded linearly to the input power under JMTR full power operation. An input power on the bolometer could be quantified with a suitable response time under the JMTR irradiation. A structure of the bolometer could withstand the 3 irradiation cycles, corresponding to the expected irradiation dose in the ITER. A few technical problems were found, such as increase of electrical conductivity of a gold meander due to nuclear transmutation of gold into mercury, and poor performance of electrical contacts between the gold meanders and measuring wires. Dimensional stability and mechanical integrity of a mica substrate is another concern. Alternative thin ceramic substrates were under development in
the EU [15]. However, it was concluded that these technical problems could be overcome by conventional techniques easily and the ITER-relevant bolometer could be developed in time.

![Diagram of a developed bolometer](image)

Figure 2 Structures of a developed bolometer. A front plate and a ground plate were made of copper. A pressure plate was made of aluminum nitride (AlN). A gold meander was on a mica substrate.

![Resistance change graph](image)

Figure 3 Resistance change as a function of input power under JMTR irradiation

2.2 Optical fibers

Improvement of radiation resistance of the optical fibers, made of fused silica (SiO₂), is remarkable in the course of ITER-EDA. At the beginning of ITER-EDA, it was a general
consensus that the optical fibers were too vulnerable to radiation effects to use them near burning plasma. Then, the design criteria claimed that optical fibers would be used out of the bio-shield. However, recent results obtained under the international round robin experiments [5-7] are yielding promising results and some optical fibers, such as Russia-made hydrogen loaded KU-1, could be used even for visible application near burning plasma with a limited life. Fluorine doped fibers showed good radiation resistance in visible regions but recent reactor irradiation tests revealed that they had higher sensitivity to the micro bending loss. For infrared applications, several optical fibers could be found with a life-time far beyond the ITER whole operation period. Fig. 4 shows examples of application of optical fibers for optical diagnostics in irradiation environments. Radiation induced luminescence of Cr$^{3+}$ in alumina was measured through an optical fiber under a Co-60 gamma ray irradiation. Realization of application of optical fibers near burning plasmas will give large technological impacts on reduction of cost and on resolving technological problems associated with limited space near the plasma, in ITER.

![Graph](image)

Irradiation test of diagnostic components for ITER application in a fission reactor, Japan Materials Testing Reactor


Abstract: Radiation effects on components and materials will be one of the most serious technical issues in fusion systems realizing burning plasmas. Magnetic materials, which would play crucial roles in control plasmas and sustain the plasma (burning plasma), will be exposed to high-flux neutrons and gamma-rays. Dynamic radiation effects will affect performance of components substantially in the beginning of exposure to radiation environments and accumulated radiation effects and probability degrade their functioning within the course of their services. High-power-density fission reactors will be good test beds to simulate the irradiation environments expected in burning plasmas. Several diagnostic components must be developed as interlaboratory test materials for ITER or similar high-flux irradiation facilities for the next generation of fusion reactors. The components include magnetic components, neutron detectors, and so on. The magnetic components are used to measure magnetic field, and the neutron detectors measure neutron interaction with the components. The diagnostics are used to evaluate the performance of the components and materials in the nuclear reactor environments.

Optically

Magnetic code: X-ray photon at x⁺1, 2 code as indicated in an irradiation cell of Japan Materials Testing Reactor

Entrainment of Magnetic code under J-MTR irradiation

References


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2.7 Formation of an Advanced Tokamak Plasma without the Use of Ohmic Heating Solenoid in JT-60U

Y. Takase,1 S. Ide,2 S. Itoh,3 O. Mitarai,4 O. Naito,2 T. Ozeki,2 Y. Sakamoto,2 S. Shiraiwa,1 T. Suzuki,2 S. Tanaka,5 T. Taniguchi,1 M. Aramasu,1 T. Fujita,2 T. Fukuda,2 X. Gao,6 M. Gryaznevich,7 K. Hanada,3 E. Jotaki,3 Y. Kamada,2 T. Mackawa,5 Y. Miura,2 K. Nakamura,3 T. Nishi,1 H. Tanaka,5 K. Ushigusa,2 and the JT-60 Team

1University of Tokyo, Tokyo 113-0033 Japan
2Japan Atomic Energy Research Institute, Naka, 311-0193 Japan
3Kyushu University, Kasuga 816-8580 Japan
4Kyushu Tokai University, Kumamoto 862-8652 Japan
5Kyoto University, Kyoto 606-8502 Japan
6Institute of Plasma Physics, Academia Sinica, Hefei, P.R. China
7EURATOM/UKAEA Fusion Association, Abingdon, OX14 3DB, UK

c-mail contact of main author: takase@k.u-tokyo.ac.jp

Abstract. An integrated scenario consisting of (1) a novel plasma start-up method using the vertical field and shaping coils, (2) an intermediate noninductive ramp-up stage, and (3) controlled transition to a high-density, bootstrap-dominated, high-confinement plasma has been demonstrated for the first time on the JT-60U tokamak. It was shown that plasma current can be ramped up even with a negative vertical field (in the direction opposite to that required for toroidal equilibrium) provided that there is a strong source of plasma. The plasma created by this technique had both internal and edge transport barriers, and had $\beta_p = 3.6$ ($\varepsilon_B = 1$), $\beta_n = 1.6$ (marginally stable to the $n = 1$ kink-ballooning mode), $H_{1500} = 1.6$ and $f_{\text{fus}} \geq 90\%$ at $I_p = 0.6$ MA and $a_e = 0.5a_0$. In these experiments, inboard turns of the shaping coil supplied about 20% of the total poloidal flux input, but further improvement is possible. This result opens up the possibility of OH-less operation, which is a requirement for ST reactors, and can also make a substantial improvement in the economic competitiveness of conventional aspect ratio tokamak reactors.

1. Introduction

In conventional tokamak operation, an Ohmic heating (OH) solenoid is used to start up and ramp up the plasma current ($I_p$) by induction. If $I_p$ ramp-up and sustainment could be accomplished without the use of OH solenoid, a substantial improvement can be achieved in the economic competitiveness of a fusion reactor by enabling a more compact design with higher magnetic field [1,2]. In particular, elimination of the OH solenoid is a necessity for a low aspect ratio spherical tokamak (ST) reactor [3].

Plasma start-up and $I_p$ ramp-up by electron cyclotron (EC) and lower hybrid (LH) waves and the vertical field ($B_v$) coil alone (RF tokamak) were first achieved on the WT-2 tokamak [4]. Several experiments have confirmed such a start-up scenario and its variations, and recently a quasi-steady-state plasma was maintained for 30 seconds on the TRIAM-1M tokamak [5]. However, these plasmas have so far been limited to low density and low plasma current. It has been suggested that plasma heating and associated $B_v$, ramp-up can provide an efficient...
means of $I_p$ ramp-up, especially in ST plasmas [6-8]. An integrated scenario consisting of (1) a novel plasma start-up method using the vertical field and shaping coils, (2) an intermediate noninductive ramp-up stage, and (3) subsequent transition to a high-density, bootstrap-dominated, high-confinement plasma with $\beta_p = 3.6$, $\beta_n = 1.6$, $H_{\text{LHCB2}} = 1.6$ and $f_{\text{BS}} \geq 90\%$ has been demonstrated on JT-60U [9]. An example is shown in Fig. 1.

The poloidal field coil configuration of JT-60U is shown in Fig. 2, together with typical examples of a large bore plasma (blue) used during the lower hybrid current drive (LHCD) $I_p$ ramp-up phase, and an inward shifted small bore plasma (red) used during the neutral beam (NB) heated high performance phase. Locations of the flux loops (in particular loop voltages measured by loops 2 and 8, $V_{22}$ and $V_{88}$, will be used in this paper) and poloidal field pick-up coils are also shown. In these experiments the current in the F-coil, which corresponds to the OH solenoid, was kept constant at zero throughout the entire discharge (Fig. 1). The main vertical field coil (VR) and the triangularity control coil (VT) were used for $I_p$ ramp-up, position control, and shaping control. The divertor coil (D) was used to create a divertor configuration, while the horizontal field coil (H) was used for vertical position control. The VT and VR coils supply poloidal flux to increase $I_p$, while the D coil acts to reduce $I_p$. The contribution of the inboard VT coil, outboard VT coils, and the VR coils to the vertical field $B$, and the poloidal flux $\Psi$ (evaluated at a nominal major radius $R = 3.4$ m) are:

$$ B_z (T) = (-0.537 + 1.948) I_{VT} (MA) + 8.720 I_{VR} (MA) $$

$$ \Delta \Psi (Wb) = (30.1 + 88.1) \Delta I_{VT} (MA) + 257.6 \Delta I_{VR} (MA) $$

The two coefficients in the parentheses for the VT coil correspond to contributions from the inboard and outboard turns of the VT coil, respectively. In the experiment described in this paper, the inboard VT coil provided about 20% of the total poloidal flux.

**FIG. 1.** Integrated scenario from plasma start-up to achievement of advanced tokamak plasma without the use of OH solenoid.

**FIG. 2.** JT-60U coil configuration and typical equilibria for the LHCD phase (blue) and the NB heating phase (red). The OH solenoid (F coil) was not used in this experiment. Locations of flux loops and poloidal field pick-up coils are also shown.
2. Plasma current start-up

In the example shown in Fig. 1 ($B_T R = 13.45$ Tm), a plasma with $I_p = 0.2$ MA was formed by a combination of preionization by EC (110 GHz) and LH (2 GHz) waves and induction by VR and VT coils. The VR and VT coil currents were ramped linearly from +0.1 to +1.1 kA and from −7.3 kA to +6.5 kA, respectively (positive current is defined in the direction that produces $B_r$, required for equilibrium), from $t = 2.10$ to 2.25 s. Such an operation is necessary because if both coils were ramped from zero, the resultant $B_r$ would become too high to hold the plasma in equilibrium. These current ramps provided a loop voltage of up to 6 V at loop 8 (inboard midplane) and 12 V at loop 2 (close to the upper outboard VT coil). The VT coil set produces poloidal field minima (poloidal field “nulls”) at two locations, at the inboard midplane and the outboard midplane (Fig. 3). The VT and VR current ramps shift the field minima towards the outboard side. The existence of a field null facilitates $I_p$ start-up.

Figure 4 shows the evolution of the vacuum field (i.e., without plasma) reconstructed from magnetic measurements inside the vacuum vessel. The ramp of VT and VR coil currents started at $t = 0.100$ s, and took 70 ms for this discharge, in stead of 150 ms for the case shown in Fig. 1. The left column shows the evolution of the poloidal flux contour, whereas the right column shows the flux profile on the midplane (in red). For comparison, the flux profile calculated from the coil currents alone (ignoring the vacuum vessel eddy currents) is shown in black. $B_r$ is initially negative (wrong direction to hold the plasma in equilibrium), and does not reverse sign until $t = 0.19$ s, but $I_p$ started to ramp up at $t = 0.105$ s. In a discharge that had neither EC nor LH, $I_p$ did not start rising until $t = 0.19$ s, approximately when the field null formed. Therefore, it can be concluded that a strong source of plasma is required for $I_p$ to start up in the absence of proper $B_r$ for establishing a toroidal equilibrium.

**FIG. 3.** Contours of poloidal flux and the magnitude of poloidal field just prior to initiating $B_r$ ramp.

**FIG. 4.** Vacuum poloidal flux contours (left) and flux profile on the midplane (right). The red curve is reconstructed from magnetic measurements while the black curve is calculated from coil currents alone.
For a typical average $B_e$ of 10 mT in a 4T toroidal field, the length along the field line from the vacuum vessel center to the vacuum vessel wall is approximately 600 m, which corresponds to about 30 toroidal revolutions. When plasma current starts to flow, the negative $B_e$ pushes the plasma outward. The eddy current induced in the vacuum vessel by this motion acts to push the plasma back, but this alone is not sufficient. During this time, a continuous source of plasma by EC and/or LH is needed in order to maintain or increase $I_p$ [10]. The plasma is in dynamic equilibrium rather than static equilibrium. When $B_e$ becomes positive and large enough, it becomes possible to maintain a toroidal equilibrium. In the example shown in Fig. 1, plasma current started to ramp up at 2.11 s. At $t = 2.15$ s, plasma is located slightly to the low field side of the vacuum vessel center. Magnetic configurations at several time slices, reconstructed using the FBI filament code (which takes into account the vacuum vessel eddy currents) [11], are displayed in Fig. 5. A divertor configuration is formed initially with the outboard VT coils acting as divertor coils. The plasma moves outward during the start-up phase until 2.20 s and becomes limited by the outboard wall. Plasma is shifted to the center of the vacuum vessel again as $B_e$ is increased.

This method of $I_p$ start-up is compared with the usual start-up using the OH solenoid in Fig. 6. For the case of $I_p$ ramp-up to 285 kA with the F coil (OH solenoid), the flux inputs from the F coil and the inboard VT coil were 1.68 Vs and 0.07 Vs, whereas the outboard VT coils and VR coils supplied 0.21 Vs and 0.61 Vs, respectively. The flux input from the inboard coils (F and inboard VT coils) was therefore 1.75 Vs out of the total input of 2.57 Vs (i.e., 68%). In comparison, for the case of ramp up to 270 kA without the F coil, the inboard VT coil supplied 0.46 Vs, and the outboard VT coils and VR coils supplied 1.36 Vs and 0.35 Vs, respectively. Therefore, the flux input from the inboard coils was 0.46 Vs out of 2.17 Vs (i.e., 21%). An important role of the inboard VT coil in this scenario is to create a field null, but this should be achievable by coils located on the inboard side, but at the top and bottom of the torus instead of the midplane.

FIG. 5. Reconstructed magnetic configurations at several time slices during the initial current formation phase.

FIG. 6. Comparison of start-up without (top) and with (bottom) the F coil (OH solenoid). $V_2$ and $V_B$ were measured by flux loops 2 and 8 (see Fig. 2).
FIG. 7. Start-up by EC alone. FBI reconstruction at $t = 0.3 \text{ s}$ (right).

It has also been demonstrated that it is possible to start up the plasma current by EC alone. In the example shown in Fig. 7, $I_p$ was ramped up inductively by VT and VR coils, as in the case shown in Fig. 1, but without LHCD. It was possible to maintain a constant $I_p$ at 200 kA for 300 ms, but the injected power was not sufficient to ramp up $I_p$ further. The FBI reconstruction at 0.3 s is also shown. The termination of the discharge in this case was caused by a slow positional drift (radially inward, and downward) of the plasma because plasma position was not feedback controlled. This can easily be remedied, and it should be possible to ramp up $I_p$ further with higher EC power.

The usual "RF tokamak" operation, in which initial current is formed by EC ionization and a positive $B_n$, was also tried. This attempt was not successful, possibly because of insufficient EC power for the large JT-60U volume. The method described in this paper requires much less RF power and is much more reliable.

3. Noninductive ramp-up

A transition to a diverted configuration starts at 2.4 s and is accomplished by 2.5 s (Fig. 5). Thereafter, the plasma configuration (plasma position, X-point, etc.) is feedback controlled. Further ramp-up to 0.4 MA was achieved by 6 s, by a combination of electron heating and current drive by EC and LH waves. During this phase, a large bore plasma (Fig. 2) is required to maintain acceptable LH coupling. This intermediate phase is similar to regular noninductive ramp-up, but a current hole [12] is already formed during this phase. The conversion efficiency from the total external noninductive input energy $\frac{1}{2} P_{\text{exl}} \text{ dt}$ to the total poloidal magnetic field energy $W_m = (L_{\text{ext}} + L_{\text{int}}) I_p^2/2$ is 3.6%, averaged over the time interval 2.6 to 5.0 s. Here, $P_{\text{Ni}} = P_{\text{LH}} + P_{\text{EC}}$ is the total noninductive input power. (Because EC and LH powers were nearly the same, the conversion efficiency would be larger by a factor of two if only the LH power is considered to be useful for $I_p$ ramp-up). The input power from the poloidal field coils $P_{\text{exl}}$ was approximately 40% of $dW_m/dt$. Therefore, the usual definition of current ramp-up efficiency was $(dW_m/dt - P_{\text{exl}})/P_{\text{Ni}} = 2.2\%$. This is a rather low efficiency, and points out that it is desirable to make the maximum use of induction by outboard PF coils.
4. Transition to advanced tokamak

A transition from a low-density noninductively driven phase to a high density, nearly self-sustained (bootstrap dominated) phase begins at 6 s when the current becomes high enough (0.4 MA) to confine the injected beam ions. Density was increased by gas puffing from 5.8 to 7 s to reduce the beam shine-through, and 85 kV NB injection was started from 6 s. The equilibrium was shifted from a full cross section LH configuration to an inward shifted NB configuration (see Fig. 2) from 6.5 to 7 s, and LH was turned off at 6.9 s. This equilibrium shift allows more central NB power deposition, reduced orbit loss, and higher density limit. Tangential beams were injected first because of their smaller shine-through fraction. Perpendicular beams were injected under stored energy feedback, which resulted in the $P_{NB}$ waveform shown in Fig. 1. This was necessary to avoid the $\beta$ collapse caused by excessive heating (discussed later). In addition to the noninductive current drive effect, $I_p$ ramps up due to the flux provided by the current increase in VR and VT coils (the latter effect is dominant). Addition of the 376.5 kV negative ion based neutral beam (NNB) contributes to further ramp-up by current drive and $\beta_p$ increase (NNB dropout at $t = 7.8$ s was not intentional).

As shown in Fig. 8, the plasma generated by this scenario had an internal transport barrier (ITB) and an edge transport barrier (H mode). The current density in the plasma core is nearly zero ("current hole"), and the $q$ profile is deeply reversed with $q_{min} = 5.6$ at $r/a = 0.7$ and $q_{95} = 12.8$ (Fig. 9). The current density inside $r = 0.4$ is small but the exact value is uncertain. A preliminary evaluation of the bootstrap current fraction yielded $f_{BS} = 90\%$ as a lower bound, conservatively setting the bootstrap current inside the current hole region to zero. Such high bootstrap fraction and confinement improvement factor are favorable for realizing steady-state operation of a fusion reactor [13]. At $t = 8.5$ s (time of maximum stored energy), $\beta_p = 3.6$ ($\epsilon \beta_p = 1.0$), $\beta_N = 1.6$, and $H_{19982} = 1.6$ were achieved at $\bar{n}_e = 0.5 n_{GW}$. These profiles and confinement improvement factor are typical of high-confinement reversed magnetic shear (RS) plasmas in JT-60U, such as the high-performance RS plasma with $f_{BS} = 80\%$ and $H_{19982} = 2.2$ sustained for 6$T_e$ (2.7 s) by NBCD at $I_p = 0.8$ MA [14].

**FIG. 8.** Profiles of electron density, electron temperature, ion temperature, and safety factor at time of maximum stored energy ($t = 8.5$ s). Both external and edge transport barriers are evident.

**FIG. 9.** Flux surfaces (left) and current density and pressure profiles (right) determined from equilibrium analysis during the high performance phase ($t = 8.5$ s).
The result of stability analysis by ERATO using the measured profiles (assuming that both $\nabla P$ and $j_{\parallel}$ are nearly zero inside the current hole region) is shown in Fig. 10. As can be seen from the figure, the growth threshold for an $n = 1$ kink-ballooning mode is around $\beta_n = 1.6$ for these profiles. This calculation is consistent with the observation that in a similar discharge with higher NB power and lower $B_T$ (3.8 T in stead of 4.0 T), $I_p$ ramped up to 0.7 MA, but ended in a $\beta$ limit disruption at $\beta_n = 1.7$ (Fig. 11). In this experiment, the duration of the noninductive ramp-up stage was limited by the plasma pulse length, and further heating and $B_p$ ramp-up resulted in a $\beta$ limit disruption. In order to ramp up $I_p$ further by heating under the same condition, it is necessary to increase the $\beta$ limit (e.g., by wall stabilization). However, it should be possible to raise $I_p$ arbitrarily (limited only by the available power) by extending the noninductive current ramp-up period.

![Fig. 10. Growth rate of the $n = 1$ kink-ballooning mode for the equilibrium just before the disruption of shot E041711 (left). Eigenfunctions of the kink-ballooning mode (right).](image)

![Fig. 11. Plasma that ended in a $\beta$ limit disruption at $\beta_n = 1.7$.](image)
5. Conclusions

In conclusion, plasma start-up, \( I_p \) ramp-up, and transition to a bootstrap dominated advanced tokamak with high \( \beta \) and high confinement \( (\beta_p = 3.6, \beta_n = 1.6, H_{198/2} = 1.6 \) and \( f_{BS} \geq 90\%) \) was demonstrated in JT-60U. This result gives confidence in reducing, and eventually eliminating the OH solenoid in ST and tokamak fusion reactors. In the present experiment, the triangularity control coil with turns on the inboard midplane was used to control the plasma shape. The inboard turns of this coil contributed about 20% of the total poloidal flux input. Demonstration of this start-up technique without using any coils on the inboard midplane is a remaining task. Extension of \( I_p \) ramp-up to higher plasma currents (i.e., lower \( q \)) and achievement of higher \( \beta_p \) without compromising the bootstrap current fraction is also a topic of future research. Since \( B_s \) ramp down (caused for example by a stored energy loss) will ramp down \( I_p \) due to the same mechanism, and therefore degrade confinement, a more serious issue is the development of a control algorithm that can react to abnormal events such as a \( \beta \) collapse.

Acknowledgments

This work was carried out as University-JAERI cooperation. Fruitful discussions with the members of the JT-60 Innovative Operations Group and participants of the “15th TRIAM Workshop on Plasma Ramp-up Experiments without the Center Solenoid” are gratefully acknowledged.

References

Formation of an Advanced Tokamak Plasma without the Use of Ohmic Heating Solenoid in JT-60U

Y. Tokase for the JT-60 Innovative Operations Group

ST Reactors Require CS-less Operation

- CS-less operation is a requirement for ST reactors

ARIES-ST
(1.0 GeV)

R = 3.2 m
R/a = 1.6
I_p = 31 MA
B_T = 2.1 T
\beta_n = 54%
f_{nuc} = 2.9 GW
Neutron wall load = 4.1 MW/m²
Recirculating power fraction = 0.32

Fixed Components
Replaceable Components

Outline

- Motivation
  - Why do we want CS-less operation?
- JT-60U CS-less Ramp-up Experiment
  - Integrated scenario consisting of:
    - I_p startup (inductive)
    - I_p rampup by aninductive overdrive
    - Controlled transition to advanced tokamak plasma (mostly inductive)
- Remaining issues

Examples of CS-less Tokamak Reactors

- Improved economic competitiveness may be realized by a CS-less design.

ITER
(18000 ton)
P_F = 0.5 GW

A-SSTR2
(28000 ton)
P_F = 4 GW

VECTOR
(-9000 ton)
P_F = 3.7 GW

Simulation for an ST Reactor

- CS does not exist in an ST reactor
  - Combine noninductive CD and inductive drive by outer PF coils.
- \( L_1 \) ramp-up by \( B_z \) ramp-up is highly effective once burning starts.
- But initial \( L_1 \) ramp-up until start of burn must be provided by other means.

O. Matsumi, Y. Takano
Fusion Science and Technology (January 2008 issue)

Flux and \( B_z \) Contributions from VR and VT Coils

- Flux @ \( R = 3.4 \) m
  \[ \Delta \Phi (Wb) = (30.1 + 88.1) \Delta I_{VT} (MA) + 237.6 \Delta I_{VR} (MA) \]
  \( \text{VT in VT out} \)
  \( \text{VR in VR out} \)
- In the present experiment:
  \( I_{VT} = 7.3 \rightarrow 6 \) kA
  \( I_{VR} = 0 \rightarrow 11.5 \) kA

Flux contribution from the inner VT coil is ~20% in these experiments

- Vertical Field @ \( R = 3.4 \) m
  \[ B_z (T) \approx (-0.537 + 1.9481) I_{VT} (MA) + 0.720 I_{VR} (MA) \]
  \( \text{VT in VT out} \)
  \( \text{VR in VR out} \)

JT-60U Coils and Heating/CD Systems

- OH solenoid \( I_r \) current is kept at zero (no flux input)
- EC/LH preionization as well as the vertical field coil \( \text{ECA/LNB} \) and the triangular field coil \( \text{EO41632} \) are used for \( L_1 \) ramp-up.
- Full cross section divertor configuration is used for \( L_1 \) ramp-up by LBCD
- Inward shifted divertor configuration is used for the high-power NB heated advanced tokamak phase.

CS-less Formation of High-Performance Plasma Demonstrated

- Noninductive ramp-up
  \[ V_x (V) \]
  \( P_{\text{heat}} (MW) \)
  \( P_{\text{R}} (MW) \)
  \( P_{\text{inj}} (MW) \)
  \( T_e (eV) \)
  \( I_x (MA) \)
Magnetic Configuration Just Before I_p Start-up

- "Field null" (minimum |B_z| region) is formed by VT coil (2 locations)
- "Field null" moves radially outward by VT and VR coil ramps

Vacuum Field Evolution

41497: no I_p start-up (no gas)
41499: I_p start-up at 0.15 s
- A small "field null" exists on the inboard midplane before VT and VR coil ramp
- Initially B_z is in the wrong direction
- B_z does not reverse sign until 0.1 s after start of VT and VR ramp
- Poloidal equilibrium, but no toroidal equilibrium?

Right: Flux profile on the midplane
Black: coil currents only
Red: coil currents and eddy currents
Left: flux contours (with eddy currents)

Configuration Evolution During CS-less I_p Start-up

- I_p ramp-up accomplished by EC/LH preionization and VT/VR coil ramps
- Transition to divertor configuration and further I_p ramp-up by LHCD

Advanced Tokamak Equilibrium and Pressure and Current Density Profiles

- Start NB heating when I_p becomes high enough for beam ion confinement
- Deeply reversed shear configuration with "current hole" is formed
- Current hole and ITB already formed during LHCD ramp-up

\[ I_{\text{con}} = 5.6 \mu \text{H} \]
\[ I_{\text{con}} = 1.4 \mu \text{H} \]
\( (l = 0.87) \)
Density and Temperature Profiles of High-Performance Plasma

- Reversed shear for \( r/a < 0.7 \)
- ITB + H-mode

\[
\begin{align*}
I_p &= 0.6MA \\
\beta_n &= 1.6 \\
I_{E4} &= 20\% \\
\gamma &= 0.6 \\
\alpha &= 13 \\
\gamma_{max} &= 5.6 \text{ at } r/a = 0.7
\end{align*}
\]

Y. Takase, et al.
J. Plasma Fusion Res. (Rapid Communications)
78, 719 (2002)

Summary of CS-less Ramp-up

- Strong preionization by EC (fundamental) and/or LH is required for effective \( I_p \) start-up
- Inductive ramp-up by VT and VR coils is effective
  - Inner VT coil provides 20% flux (VR and outer VT coils provide 80%)
  - \( I_p \) start-up by VR ramp alone is possible but less effective (~ 50 kA)
  - Static field by VT followed by VT and VR ramp-up is more effective
  - Formation of "field null" is effective
  - Further improvement of start-up scenario is possible
- LHCD is effective for noninductive \( I_p \) ramp-up
  - Decouples \( \Delta \Phi \) and \( B_n \) from PF coils
  - Maintenance of 250 kA (but not further ramp-up) was possible by EC alone
  - \( I_p \) ramp-up by EC alone should be possible but requires higher power
- An integrated scenario with controlled \( I_p \) ramp-up, transformation to advanced tokamak plasma, and controlled ramp-down is demonstrated:
  - ITB + H-mode plasma with \( \beta_n = 1.6 \) and \( \gamma = 4.0 \)

Stability Limited by \( n = 1 \) Kink-Ballooning Mode

- No \( \beta \) collapse observed at \( \beta_n = 1.6 \) in E41711
- \( \beta \) collapse observed at \( \beta_n = 1.7 \) in E41710

Remaining Issues

- Demonstrate a scenario that uses no turns on the inboard midplane
  - Use inboard top/bottom coils
- Extension to higher \( I_p \), higher \( \beta_n \)
- Application of CS-less operation to ST
- Develop control algorithm that can react to \( \beta \) collapse, etc.
2.8 Rokkasho: Japanese Site for ITER

Japanese ITER Site Forum*; S. Ohtake1, M. Yamaguchi2, S. Matsuda3 and H. Kishimoto3

1) Ministry of Education, Culture, Sports, Science and Technology (MEXT), 1-3-2
   Kasumigaseki, Chiyoda-Ku, Tokyo, 100-8966 Japan
2) Aomori Prefectural Government, 1-1-1 Nagashima, Aomori-Shi, Aomori-Ken, 030-8570
   Japan
3) Japan Atomic Energy Research Institute (JAERI), 2-2-2 Uchisaiwai-Cho, Chiyoda-Ku,
   Tokyo, 100-0011 Japan

e-mail: hiroshik@naka.jaeri.go.jp

Abstract. The Atomic Energy Commission of Japan authorized ITER as the core machine of the Third Phase
Basic Program of Fusion Energy Development. After a series of discussions in the Atomic Energy Commission
and the Council of Science and Technology Policy, Japanese Government concluded formally with the Cabinet
Agreement on 31 May 2002 that Japan should participate in the ITER Project and offer the Rokkasho-Mura site
for construction of ITER to the Negotiations among Canada (CA), the European Union (EU), Japan (JA), and the
Russian Federation (RF). The JA site proposal is now under the international assessment in the framework of
the ITER Negotiations.

1. Introduction

The site proposal for ITER is to be made in accordance with the “the site requirements and site design
assumptions” defined by the extended-EDA Parties (EU, JA, RF) during the
ITER Engineering Design Activities (EDA)11, which include the issues such as land, heat sink
and water supply, electric power supply, transportation and shipping, technological and socio-
cultural infrastructure, regulation and decommissioning. In addition, some aspects are
considered for international joint assessments such as the relation between the ITER legal
entity and the host country, licensing process, operation and decommissioning costs.

Japanese Government (MEXT) set up a site assessment committee in mid-2001 and
completed its assessments for three candidate sites in Japan, i.e., Naka, Rokkasho, and
Tomakomai. As a conclusion, it was confirmed that the Rokkasho site shown in Fig. 1 could
satisfy the generic site requirements and site design assumptions with some additional works.
The key features of the Rokkasho site are described in this paper.

* Comprised of Japanese Government (MEXT), Aomori Prefectural Government, and JAERI.
2. Technical Aspects of Site Requirements

2.1. Land and geotechnical characteristics

The land area of 40 hectares is required basically and additional 30 hectares are necessary for a temporary use. The geological investigations indicate as shown in Fig. 2 that the site has a good supporting rock-bed (Neogene Takahoko Strata) at a shallow depth with a long-term stable bearing capacity larger than 200 tons/m².

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*Long term bearing capability of Takahoko Strata > 200t/m²*

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**Fig. 1. Rokkasho-Mura site for ITER.**

**Fig. 2. Geological profile around Rokkasho-Mura site.**
The design basis earthquake is given at the free bed rock surface with a shear wave velocity over 700 m/s. Seismic isolation with laminated rubber bearings is adopted against the design basis earthquake with a peak acceleration of about 230 gal. The structure and its assembling are shown in Fig. 3.

**Fig. 3. Seismic isolation of Tokamak building.**

2.2. Heat rejection and cooling water supply (Fig. 4)

The site is located beside a fresh water lake and is also close the seashore as shown in Fig. 1. Water supply for either direct cooling or evaporation cooling is afforded sufficiently to reject the 450 MW average heat. A standard option of cooling tower system requires the fresh water supply of 16 m$^3$/min. (23,000 m$^3$/day), which is drawn from the Takahoko Lake. The direct cooling can be done by rejecting the heat into the Mutsu-Ogawara Port.

**Fig. 4. Possible heat rejection of ITER at Rokkasho site.**
2.3. Electrical power supply

A steady state as well as pulsed electric power required for the ITER operation will be supplied by a 275 kV utility line. The utility power line around the ITER site is given in Fig. 5. Reactive power compensators and high-pass filters will be required to suppress the voltage fluctuations and the higher harmonics generation. A fast active power will be supplied by a variable-speed motor-generator with a flywheel, which is effective in reduction of the network frequency disturbance as well as of the output fluctuations of the nearest neighbor nuclear power generator for a pulsed load of ITER up to 270 MW.

![Fig. 5. Utility power line for ITER.](image)

2.4. Shipping and Transportation

The Matsu-Ogawara Port faced to the Pacific Ocean locates about 5 km away in the east from the site as shown in Fig. 1. A 5,000 tons class ship can be docked at this Port and components as heavy as 1,000 tons can be unloaded and transported to the site by using a barge ship as shown in Fig. 6 through the existing public road.

The outer most PF coils and the assembled CS coil will possibly be transported, indicating no on-site manufacturing work of these coils.

![Fig. 6. Shipping by barge ship for 1,000 tons class equipment.](image)
2.5. Tritium transport

Tritium is firstly assumed to be procured from Canada. One sea-container can carry three 50g-T transport packages and consequently six shipments per year will be sufficient for the ITER consumption of Tritium; 0.9 kg-T/year averaged over 20 years of operation. The transport package will be approved as a BU-type package stipulated in the IAEA regulation.

3. Scientific and Industrial Infrastructures

A large-scale nuclear-fuel reprocessing plant is now under construction in Rokkasho-Mura and a few thousands of engineers and technicians are working here. Sufficient industrial infrastructures and workforce are available. People in the region tend to have good understanding in the nuclear energy development with these circumstances.

Fusion research in Japan is promoted as a national program. Many fusion scientists and fusion laboratories/institutes are ready to support the ITER project when constructed in Japan.

4. Socio-cultural Infrastructure

Establishment of an international school is planned in Rokkasho-Mura, which will provide instruction for the children of foreign residents. A comfortable residential environment will be provided in the surrounding area as well. There are many medical facilities, including hospitals, clinics, and dental offices, and treatment in English is available. Various sports and cultural attractions, as well as shopping centers, are found in this area.

5. Licensing Aspects

The ITER facilities as a principle will be regulated by laws and regulations concerning the nuclear safety regulation, which comes under the Atomic Energy Basic Law, because the facilities generate radiation and hold radioactive materials such as Tritium. Taking into account the intrinsic safety features of ITER, the Nuclear Safety Commission and the regulatory authority are to take a reasonable approach for ensuring the ITER safety. The major safety requirements are appropriate radiation protection, prevention of accidents and mitigation of consequences of accidents. It is not necessary in the case of ITER to assume preparedness for nuclear disaster, which is obligated to the fission facilities.

6. Rad-Waste Disposal

Japanese Government and the local governments (Aomori Prefecture and Rokkasho-Mura) stated to accept the final disposal of the entire ITER rad-waste in the site or in its
vicinity. The actual disposal (Fig. 7) will be conducted along the technical guidelines provided by the Atomic Energy Commission of Japan in 1998.

![Diagram of the disposal area](image)

*Fig. 7. Low level rad-waste disposal at Japan Nuclear Fuel Ltd.*

7. Summary

The Rokkasho site satisfies the ITER site requirements and site design assumptions defined in the ITER Engineering Design Activities (EDA). The flexibility of land area and its topology, sufficient capability of electric power and water supply, transportation capability of heavy components and high quality and affordable skills and workforce ensure the safe and reliable construction and operation of ITER. A specific benefit of the site is the availability of the rad-waste disposal in the site or its vicinity. The Nuclear Safety Commission and the regulatory authority of Japan consider to regulate the ITER facilities to ensure its safety in flexible and practical manners.

Reference

Life in Rokkasho

Community & Housing

International School

Medical Service
Rokkasho of Aomori Welcomes World Scientists for ITER

Advantageous site-characteristics

- Wide area of the site provides flexibility for ITER operation.
- Heavy and large components can be transported via existing roads.
- Cooling water supply is sufficient for ITER steady state operation.
- Meteorological characteristics of the site are mild enough.
- The entire ITER rad-waste can be disposed in the site or its vicinity.
- A comfortable living environment for foreign residents and visitors.
国際単位系（SI）と換算表

### 表1 SI基本単位および補助単位

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### 表2 SIと併用される単位

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1 eV = 1.60218 × 10^{-19} J
1 u = 1.66054 × 10^{-27} kg

### 表3 固有の名称をもつSI補足単位

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### 表4 SIと共に略称的に

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### 表5 SI換算表

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### 拡張表

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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年発行による。ただし、1 eVおよび1 uの値はCODATAの1986年推奨値によった。
2. 表4は電気、原子、液、密度を含むいるが2次元の単位なので、この表は略称した。
3. 表5は、JISでは流体の圧力を表す場合に圧力2等のカテゴリに分類されている。
4. EC開発委員会意図書「SIおよび公称単位」mmHgを表2のカテゴリに入れている。

（86年12月25日現在）