20th IAEA Fusion Energy Conference
Vilamoura, Portugal, 1-6 November 2004

IAEA-CN-116/OV/1-1

Overview of JT-60U Progress towards Steady-state Advanced Tokamak

S. Ide and the JT-60 Team

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

Overview of JT-60U Progress towards Steady-state Advanced Tokamak

S. Ide and the JT-60 Team

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. Recent experimental results on steady state advanced tokamak (AT) research on JT-60U are presented with emphasis on longer time scate in comparison with characteristics time scales in plasmas. Towards this, modification on control in operation, heating and diagnostics systems have been done. As the results, $\sim$ 60 s $I_{\rm p}$ flat top and an $\sim$ 30 s H-mode are obtained. The long pulse modification has opened a door into new domain for JT-60U. The high normalized beta ($\beta_{\rm N}$) of 2.5 is maintained for 15.5 s in a high $\beta_{\rm p}$ H-mode plasma. A standard ELMy H-mode plasma is also extended and change in wall recycling in such a longer time scale has been unveiled. Development and investigation of plasmas that are relevant to AT operation has been continued in a former 15 s discharges as well in which higher NB power ($\leqq$ 10 s) is available. Higher $\beta_{\rm N} \sim 3$ is maintained for 6.2 s in high $\beta_{\rm p}$ H-mode plasmas. High bootstrap current fraction ($f_{\rm BS}$) of $\sim$ 75% is sustained for 7.4 s in an RS plasma. On NTM suppression by localized ECCD, ECRF injection preceding the mode saturation is found to be more effective to suppress the mode with less power compared to the injection after the mode saturated. The domain of the NTM suppression experiments is extended to the high $\beta_{\rm N}$ regime, and effectiveness of m/n=3/2 mode suppression by ECCD is demonstrated at $\beta_{\rm N} \sim 2.5$ - 3. Genuine centre-solenoid less tokamak plasma start up is demonstrated. In a current hole region, it is shown that no scheme drives a current in any direction. Detailed measurement in both spatial and energy spaces of energetic ions showed dynamic change in the energetic ion profile at collective instabilities. Impact of toroidal plasma rotation on ELM behaviors is clarified in grassy ELM and QH domains.

1 Introduction

The major objectives of the JT-60U project are to establish scientific basis for ITER and explore new plasma domain leading to realization of an attractive fusion reactor. Toward realization of a steady state (SS) tokamak fusion reactor, it is essential to increase the fraction ($f_{\rm BS}$) of the bootstrap (BS) current relative to the total plasma current ($I_{\rm p}$); a value $f_{\rm BS} \gtrsim 70\%$ is typically required [1], and the normalized beta ($\beta_{\rm N}$); $\beta_{\rm N} \sim 3.5$ is required. This is so-called advanced tokamak (AT) concept. Development of AT relevant plasmas, one with weak magnetic shear ("high $\beta_{\rm p}$ plasma") and one in reversed magnetic shear ("RS plasma"), has been pursued. Formation of the internal and the edge transport barriers, that is the H-mode pedestal, (ITB and ETB) are ones of effective method of achieving high parameters. Towards high $\beta_{\rm N}$, MHD instabilities are to be overcome. Non-inductive current drive and current profile control are key issues towards SS operation. Heat and particle handling is a key for divertor feasibility. We have made effort to investigate and integrate these segments towards our objectives. Furthermore recent modification in control system on the JT-60U facilities enables us to explore plasmas of longer pulse duration. In this paper, recent JT-60U experimental result after the 19th IAEA Fusion Energy Conference [2] are reviewed with emphasis on the extension of pulse duration in comparison with characteristics time scales in plasmas.

2 Extension of JT-60U pulse duration and machine status

As described above, investigation of plasma behavior in longer time scale in comparison with various characteristics time is an important issue in the AT research. The characteristics time in JT-60U plasmas extends from the energy confinement time (τ_e), order of several hundreds

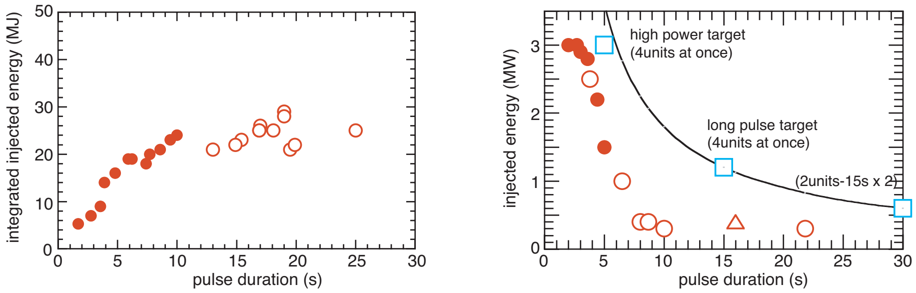

FIG. 1: Progress in heating systems shown as the injected power versus injected duration, (a) ECRF and (b) N-NB. Closed circles represent data before the 19th IAEA conference, while open circles represent data before the 19th IAEA conference.

milli seconds, to wall saturation time (τ_{wall}), order of several tens of seconds. Among the characteristics times, one of the most important is so-called the current relaxation time (τ_R), a time scale of the toroidal current density profile ($j(\rho)$, where ρ is the normalized flux radius) to saturate, or the one turn loop voltage profile ($V_\ell(\rho)$) to flatten in the entire plasma cross-section. Although τ_R varies depending on the electron temperature (T_e), the effective charge (Z_eff) and so on and also location where external current is supplied, it ranges around a few to a few tens of seconds in JT-60U plasmas. Therefore conventional JT-60U NB pulse length of 10 s may not be long enough in view of investigation in $j(\rho)$ or plasma characteristics which are closely related to the change in $j(\rho)$. In order to enable research in such a long time scale, modification on control in operation, heating and diagnostics systems of the JT-60U facility was done without major hardware upgrade. As the results, the maximum pulse length of a discharge is extended from 15 s to 65 s, the maximum duration of both parallel P-NB (four units) and N-NB (negative-ion source NB) injections are extended to 30 s (formerly 10 s). Pulse length of the perpendicular P-NB (seven units) is unchanged (10 s) except one that is used for the charge exchange recombination spectroscopy (CXRS), but they can be injected arbitraly in 30 s from the beginning of the first P-NB injection. Therefore six P-NB units can be injected for 30 s. The maximum duration of RF (ECRF and LHRF) pulses are extended to 60 s. Since no major modification is done on the hardware, there are a several limitations. Since the capacity of the poloidal coil power supply (motor generator) tolerance of feeders against joule heat are unchanged, sustainable I_p, triangularity (δ), height of the divertor X-point and so on are limited. The toroidal magnetic field at the machine center (B_{t0}) is also limited to 3.3 T for about 30 s and 2.7 T for 65 s. The NB power per unit is reduced from about 2.5 MW to 2 MW. Heating systems require conditioning in order to extend their pulse lengths. The pulse length of the P-NBs is fully extended as planned. The pulse length of N-NB has reached up to 25 s with injection power of 1 MW. The ECRF pulse has reached 21.8 s using four Gyrotrons in series. A grill mouth made of carbon is newly placed at the LHRF luncher. It yet has been under

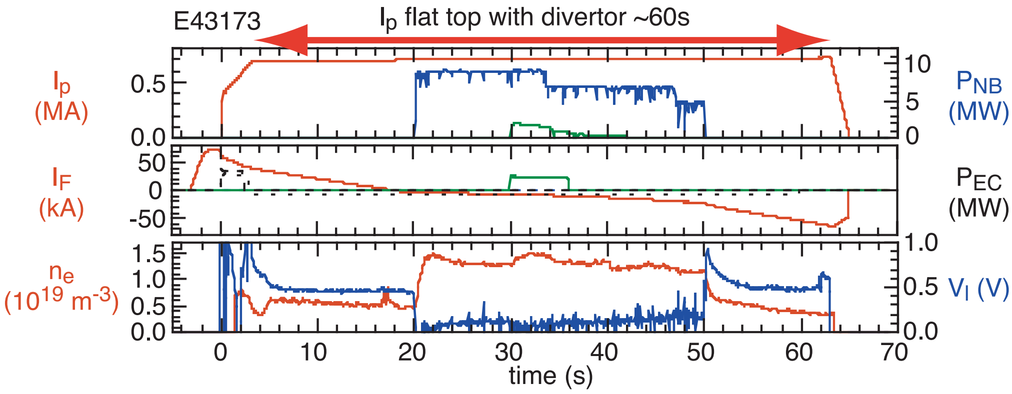

FIG. 2: Waveforms of a JT-60U 65 s discharge (E43173). (a) the plasma current (I_p) and the positive and negative NB powers (P_NB), (b) the Ohmic heating coil ('F-coil') current (I_F) and the ECRF power (P_EC, a.u.). (c) the line averaged electron density ($\bar{n}_\mathrm{e}$) and the surface loop voltage (V_ℓ).

conditioning, so far the maximum injection power 1.6 MW (about 70% of the previous value) and the total injection energy of 11 MJ (before carbon mouth placed it was 10 MJ). The progress of the ECRF and N-NB systems is shown in Fig. 1. The total input power of 360 MJ has been achieved. A 65 s discharge with $I_{\rm p}$ = 0.7MA flat top of $\sim$ 65 s is obtained (Fig. 2). And a 30 s NB heated plasma has been obtained with $I_{\rm p}$ up to 1.0 MA.

3 Towards steady state sustainment of AT relevant plasmas

In JT-60U, large efforts in extending various AT relevant plasma had been continued within 10 s of heating pulse. Towards development of the ITER hybrid operation scenario [3], the long pulse experiments will largely contribute. The purpose of the hybrid scenario is to maximize fluence in a year, therefore extending pulse length with help of non-inductive current drive keeping as high fusion power as possible is desired. Steady state scenario in ITER and steady state reactor is determined as a discharge fully sustained with non-inductive current drive, the bootstrap current and externally drive current(s). In practical or economical view, $f_{\rm BS}$ is expected to be large enough, $\gtrsim$ 50% in ITER and $\gtrsim$ 70% in a reactor. Towards these steady state development, conventional NB operation, in which pulse length is shorter but higher power is available, is contributing. In those AT domain plasmas, the target or resultant current profile has important meaning. The major issue here is the MHD instabilities. Towards the integration of the present developing AT relevant scenarios to ITER or reactor relevant situations, compatibility of such AT plasmas to the divertor feasibility is also a major issue.

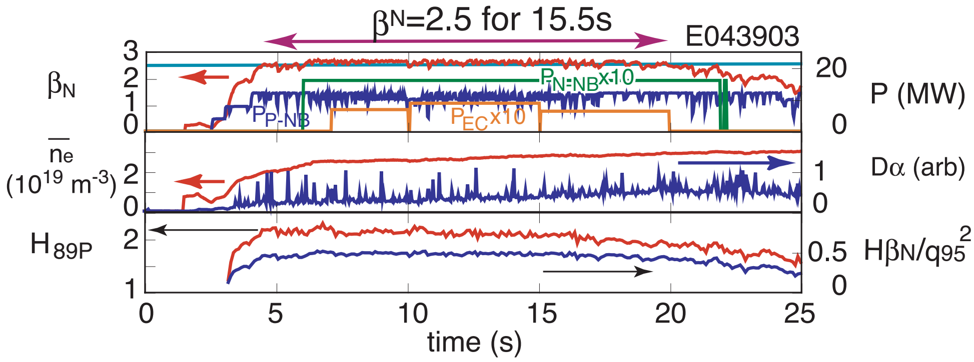

FIG. 3: Waveforms of a high $\beta_{\rm N}$ long susteinment discharge (E43903). (a) the normalized beta ($\beta_{\rm N}$) and the positive and negative NB and ECRF powers, (b) the line averaged electron density ($\bar{n}_{\rm e}$) and the intensity of the D_α line. (c) the confinement improvement factor to the L-mode scaling ($H_{\rm 89P}$) and a figure of merit of the fusion performance ($H_{\rm 89P} \times \beta_{\rm N}/q_{95}^2$).

3.1 Extension of high $\beta_{\rm N}$ sustainment

One of the major issues for the extension of a JT-60U pulse length is to extend duration of high $\beta_{\rm N}$ sustainment. We had succeeded to sustain $\beta_{\rm N}$ = 2.7 for 7.4 s Ref. [4]. By optimizing high $\beta_{\rm p}$ ELMy H-mode plasma of $I_{\rm p}$ = 0.9 MA at $B_{\rm t}$ (the toroidal magnetic field at the plasma major radius) = 1.7 T and q_{95} (the safety factor at the 95% toroidal flux) $\sim$ 3.4, $\beta_{\rm N}$ = 2.5 is succeeded to be maintained for 15.5 s (Fig. 3) [5]. It corresponds to $\sim$ 1.7 $\tau_{(}$R). The evolution of $\beta_{\rm N}$ was carefully optimized in order to reach as high as possible but avoiding the neo-classical tearing mode (NTM) to appear, since the sustainable power is not enought to raise $\beta_{\rm N}$ with the NTM existing.

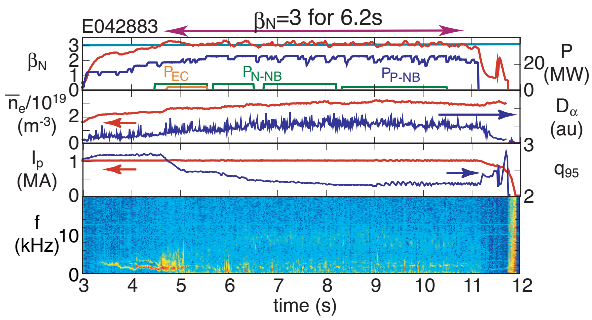

FIG. 4: Waveforms of a high $\beta_{\rm N}$ = 3 discharge (E42883). (a) the normalized beta ($\beta_{\rm N}$) and the positive and negative NB powers, (b) the line averaged electron density ($\bar{n}_{\rm e}$). (c) the intensity of the D_α line. (d) the magnetic fluctuation, no clear activity is observed during $\beta_{\rm N}$ = 3.

Although localized ECCD has been proved effective on stabilizing an NTM [6], the ECRF power that can be sustainable for 20 s is too low to apply to this duration, therefore it is not applicable. By the optimization no distinct NTM is observed for this 15.5 s. As shown in the figure, P-NB power gradually increases so does the total injection power, this is attributed to the decrease in confinement. This is indicated as a waveform of the confinement improvement factor to the L-mode scaling (H_{89P}) [7]. The reason why the confinement is getting worse can be attributed to the increase in the wall recycling which appear as an increase in the D_α line brightness (Fig. 3 (b)). The wall recycling will be discussed later. Although the confinement gradually degrades, a factor $H_{89P}\beta_N/q_{95}^2$, which is a figure of merit of the fusion performance, $\gtrsim 0.4$ is kept for 15.5 s as shown in Fig. 3 (c). It is noted that $H_{89P}\beta_N/q_{95}^2 \sim 0.4$ corresponds to the ITER standard ELMy H-mode scenario with Q = 10, while $H_{89P}\beta_N/q_{95}^2 \sim 0.3$ corresponds to the ITER steady state scenario with Q = 5.

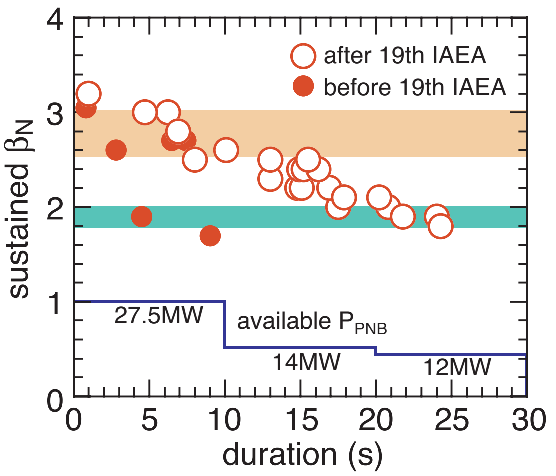

FIG. 5: Progress of sustainment of high β_N, sustained β_N is plotted against sustaining period. Closed circles indicate the results obteined before the last IAEA conference, while open circles represent the result after the conference. Upper hatched belt indicates the ITER hybrid and steady state domain, while lower one indicates the ITER reference H-mode domain.

Not only the extension towards time axis, effort to extend the domain into higher β_N has also been carried out. Since higher power is required, this experiment has been carried out with conventional 10 s P-NB injection setting in which the maximum total P-NB power can be 27.5 MW and that of N-NB can be 4-5 MW. In this experiment, NTMs are to be avoided as well. For this, the experiment was carried out at rather low q_{95} of below 3. Due to the low q_{95} but $q_0$is kept around unity, q = 1.5 or 2 that are rational surfaces at which NTMs can occur mostly, location shifts quite outwards at which the pressure gradient can be low enough so as NTMs not to occur. Also broadening of heating profile was intended. As the result, β_N = 3 is sustained for 6.2 s (Fig. 4) without distinct NTM. In Fig. 5 plotted is progress in the susteined β_N against sustaining duration.

3.2 Extension towards steady state operation with high f_{BS}

As mentioned before, in ITER steady state scenario, f_{BS} aroud 50% or higher is expected. With this range of f_{BS}, various q profile can be consistent. That is from one that with quite flat q profile in the core region to one that has very high q_0. A flat q profile with $q_0 \sim$ 1.5-2.5 is preferably accepted, in view of MHD stability, alpha particle confinement, f_{BS}, easier external $j(\rho)$ control and so forth. In JT-60U AT relevant plasma of this kind has been developed based on the high β_p ELMy H-mode Ref. [8]. By ex-

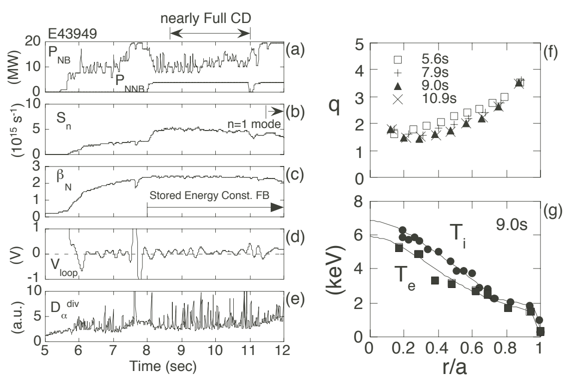

FIG. 6: Typical waveforms of a weak shear ELMy H-mode discharge with large bootstrap current fraction under nearly full non-inductive current drive: (a) injected P-NB and N-NB power, (b) neutron production rate (S_n), (c) normalized beta (β_N), (d) surface loop voltage and (e) deuterium recycling emission at the divertor (D_α). (f) Temporal evolutions of q profile shows current profile became stationary. (g) T_e close to T_i thanks to electron heating by N-NB.

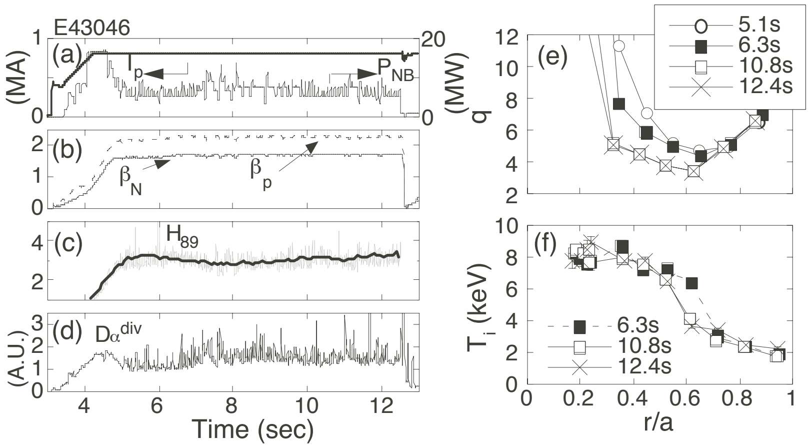

FIG. 7: Typical waveforms of a reversed shear ELMy H-mode discharge with large bootstrap current fraction under nearly full non-inductive current drive: (a) plasma current ($I_{\rm p}$) and injected NB power ($P_{\rm NB}$), (b) normalized beta ($\beta_{\rm N}$: solid curve) and poloidal beta ($\beta_{\rm p}$: dotted curve), (c) H factor (H_{89}) and (d) deuterium recycling emission at the divertor (D_α). Temporal evolutions of (e) q profile and (f) ion temperature profile ($T_{\rm i}$) show the current and the pressure profiles became stationary.

tending the pulse length of N-NB, this domain of operation is investigated [9]. Although the fraction ($f_{\rm CD}$) of the non-inductively driven current to the plasma current is not 100%, $f_{\rm CD} >$ 90% with $f_{\rm BS}$ $\sim$ of 50% is maintained for 2.3 s at $\beta_{\rm N}$ $\sim$ 2.4 (Fig. 6). Thanks to the off-axis BS current, q_0 gradually elevates from near unity to 1.5 and q profile becomes flatter. In the discharge confinement suddenly decreased ($\sim$ 11 s) and the duration of good performance is limited by this. The cause is not clear yet and under investigation.

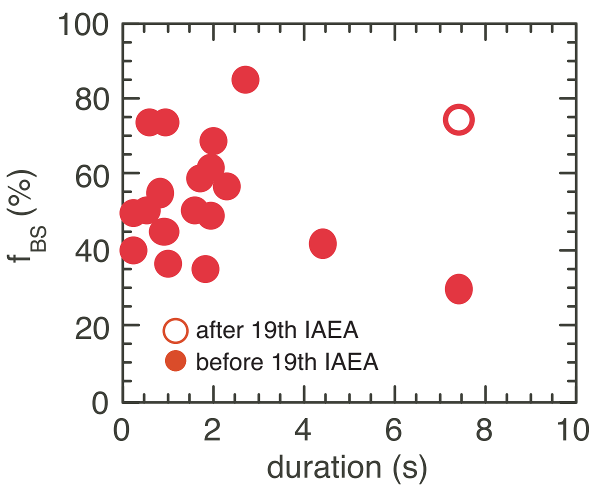

FIG. 8: Progress of sustainment of high $f_{\rm BS}$, sustained $f_{\rm BS}$ is plotted against sustaining period. Closed circles indicate the results obteined before the last IAEA conference, while open circles represent the result after the conference.

In a reactor design, $f_{\rm BS}$ of 70% or higher is expected. In order to raise $f_{\rm BS}$ an RS configuration is expected to be benefitial. In JT-60U, high $f_{\rm BS}$ of 80% was sustained for 2.7 s in a high confinement RS plasma ($I_{\rm p}$ = 0.8MA, $B_{\rm t}$ = 3.4 T) with $f_{\rm CD}$ $\sim$ 100% [10]. By optimizing the similar plasma a high $f_{\rm BS}$ of $\sim$ 75% is successfully sustained for 7.4 s ($\sim$ $2\tau_{\rm R}$) with very high confinement of $HH_{98({\rm y},2)}$ = 1.7 (Fig. 7) [9]. The duration corresponds to $\sim 2\ \tau_({\rm R})$. Continuous off-axis heating to maintain the ITB and the optimization of injection of on-axis counter (to the $I_{\rm p}$ direction) parallel P-NB to avoid disruption are found to be the keys for the sustainment. In the discharge, the duration is limited by the NB pulse length. In Fig. 8 plotted is the susteined $f_{\rm BS}$ against sustaining duration.

3.3 MHD studies to access higher beta

As described in Subsection 3.1, in positive shear (PS) discharges with q_0 close to unity, the NTM is one of the most critical modes that prevents sustainable $\beta_{\rm N}$ from approaching to the ideal MHD limit. On JT-60U, we have demonstrated that localized current drive by ECCD are effective to suppress NTMs [6]. By further study it is found that ECRF injection before the mode fully develops (early injection) is more effective than the injection after the mode gets fully developed (late injection) for the first time (Fig. 9) [11, 12]. Decrease in the current density at the mode island as the mode grows and compensation of the lost current by ECCD are confirmed by the Mortional Stark Effect (MSE) measurement [13].

On the other hand among RS plasmas, some discharges disrupt at low β_{N}, ~ 1 or even lower. This disruption should be avoided a reliable operation. We have studied these disruptions at low β_{N}, and reported that the double tearing mode (DTM) could be a cause of them [14,15]. By improved measurements and analysis in temperature and magnetic fluctuations and the q profile (or the equilibrium) it is found that this low β_{N} disruption can also be triggered by mode coupling of the surface MHD instability driven by peripheral large plasma current and the internal mode in the RS region at the rational surface whose safety factor correspond to the mode number of the surface mode [16]. Both a surface mode and a internal mode can trigger to interact each other.

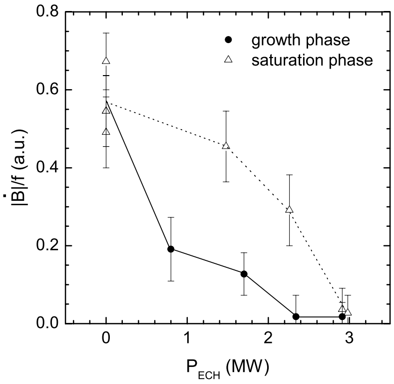

FIG. 9: Dependence of the m/n = 3/2 mode stabilization on the EC power, magnetic fluctuation amplitude. Close circles indicate a case where ECRF injected during the mode is growing, while ope triangles indicate a case ECRF injected after the mode saturated.

3.4 Divertor compatibility of AT plasmas

In ITER or a reactor, it is required to raise the electron density around or above the Greenwald density (n_{GW}), in order to attain enough fusion reaction. High radiation loss is also required in order to reduce the heat flux to the divertor. Compatibility of AT plasmas with high density and high radiation loss has been investigated in both RS plasma and high β_{p} H-mode plasma with a weak positive shear [17]. In the RS plasmas, high confinement of $HH_{98(\mathrm{y},2)} = 1.3$ is achieved at the high density above n_{GW}, that is f_{GW} ($= \bar{n}_{\mathrm{e}}/n_{\mathrm{GW}}$) $= 1.1$, even with NB fuelling only. The total radiation loss is enhanced to the level greater than 90% of the net heating power with high confinement ($HH_{98(\mathrm{y},2)} = 1.1$) at high density ($f_{\mathrm{GW}} = 1.1$) by injecting seed impurity Ne together with D_2 gas into the RS plasmas. In the high β_{p} H-mode plasmas, high confinement ($HH_{98(\mathrm{y},2)} = 0.96$) is maintained at high density ($HH_{98(\mathrm{y},2)} = 0.92$) with high radiation loss fraction ($f_{\mathrm{rad}} > 0.9$) by utilizing high-field-side pellets and Ar injections. These results are summarized in Fig. 10. In these plasmas, the high f_{GW} is obtained due to a peaked density profile inside the internal transport barrier (ITB). The strong core-edge parameter linkage is observed in the high β_{p} H-mode plasmas with pellets and Ar injections, as well as without Ar injection, where the pedestal β_{p} is enhanced with the total β_{p}. On the other hand, the pedestal β_{p} is kept at small value in the RS plasma, indicating that confinement improvement is mainly attributed to the strong ITB. The radiation loss profile in the main plasma is peaked due to the impurity accumulation in both plasmas.

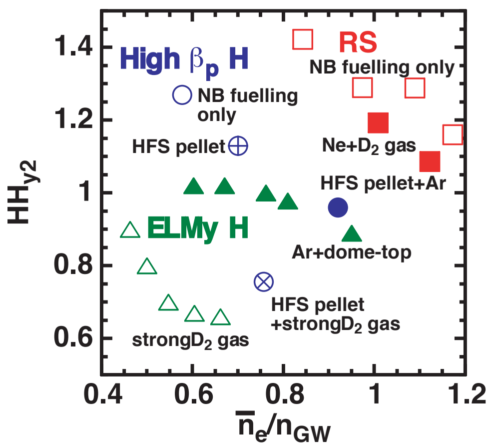

FIG. 10: $HH_{98(\mathrm{y},2)}$ as a function of f_{GW}. Squares : RS plasma. Circles : high β_{p} H-mode plasma. Triangles : ELMy H-mode plasma. Closed symbols show the data with impurity seeding.

3.5 Development of real-time current profile control

Control of safety factor profile ($q(\rho)$) is essential for stable sustainment of AT plasmas. Therefore, a real-time $q(\rho)$ control system has been developed in JT-60U. This system enables real time evaluation of $q(\rho)$ by MSE diagnostic and control of CD location by adjusting the parallel refractive index $N_\parallel$ of LH waves through the change of phase difference ($\Delta\phi$) of LH waves between multi-junction launcher modules. Real time estimation of $q(\rho)$ that is evaluated from the MSE measurement has been demonstrated for the first time. Real time feed back control with the LHRF system is under going.

4 Towards understanding of confinement and transport in AT plasmas and more general issues

As described at the beginning, recent JT-60U experiments have focused largely on extension of AT relevant issues towards time-axis. However, at the same time we have continuously kept investigation on other issues that are related to AT development as well and on more general plasma physics.

4.1 Current hole studies

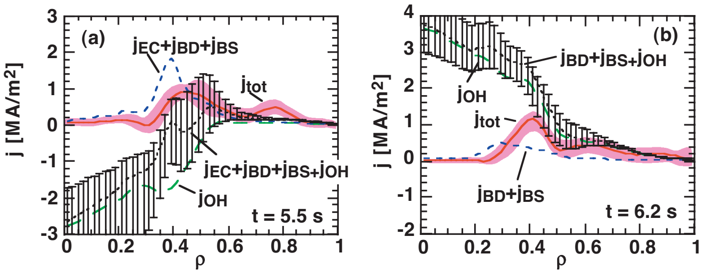

FIG. 11: Radial profiles of current density in the current hole experiments. The inductive toroidal field is positive in (a) and negative in (b). Here, jtot (solid line with a shaded belt) denotes measure current density, $j_{\rm OH}$ is calculated inductive current density, $j_{\rm EC}$ is calculated EC-driven current density, $j_{\rm BD}$ is calculated beam-driven current density and $j_{\rm BS}$ is calculated bootstrap current density. The sum of $j_{\rm OH}$ + $j_{\rm BD}$ + $j_{\rm BS}$ + $j_{\rm EC}$ is shown by a dotted line with error bars.

It has been investigated if there exists some mechanism to clamp the current density at zero level in the current hole. Though the nearly zero toroidal current in the central region (a 'current hole') is sustained for several seconds in the JT-60U tokamak [10], it has not been clear whether the current drive source such as inductive toroidal electric field (E_ϕ) and non-inductively driven current ($j_{\rm NI}$) remains at zero level or some mechanism works to clamp the current density at zero level against the current drive source. Two kinds of experiments were performed to investigate responses to E_ϕ and $j_{\rm NI}$ separately [18]. In the first experiment, E_ϕ was changed transiently, with keeping $j_{\rm NI}$ inside the current hole as small as possible, by changing non-inductive current (bootstrap and EC-driven currents) outside the current hole. In Fig. 11, the sum of calculated $j_{\rm OH}$, $j_{\rm EC}$, $j_{\rm BD}$ and $j_{\rm BS}$ and measured $j_{\rm tot}$ are compared for (a) positive E_ϕ and (b) negative E_ϕ cases. In both cases, the calculated current density is dominated by inductive current and is largely negative in (a) and is largely positive in (b). The measured current density, however, remained nearly zero. In the second experiment, EC current drive inside the current hole was attempted in the co- and counter-directions to the plasma current during the quasi-stationary period with sufficiently small E_ϕ. In both directions, the EC current drive did not change the current inside the current hole and the current hole was maintained. From these results, it has been shown experimentally for the first time that the current hole is maintained by some mechanism to clamp the current density at zero level once when the current density becomes at zero level in the central region. Though appearance of MHD instabilities and reconnection of field lines are shown by MHD simulations [19], no global MHD instabilities were observed during these experiments. Hence some mechanism other than MHD instabilities seems to work. Model simulation on current hole formation and sustainment has been carried out and consistent results are obtained [20].

4.2 Confinement of energetic ions

Anomalous transport of α particles by collective instabilities induced by the energetic ions and/or MHD instabilities can cause serious problems in a fusion reactor. In JT-60U, behavior of high energy ions and their inducing instabilities have been investigated by utilizing N-NB [21, 22]. Newly installed energy analyzer using natural diamond detector enables measurement on the energy distribution of the energetic ions [23]. Multi channel neutron emission detector brings information of spatial distribution of the energetic ions. It is found that in a weak shear plasma with higher $\beta_{\rm h}$ (the beta of the high energy ions) a bursting mode called abrupt large-amplitude event (ALE) redistributes energetic ions from the core region to the outer region of the plasma (Fig. 12 (a)). Neutral particle flux in limited energy range (100 – 370 keV) is found to be enhanced by the ALEs (Fig. 12 (b)). This indicates that energetic ions in this energy range are redistributed. The energy corresponds to the N-NB ions that induce the modes.

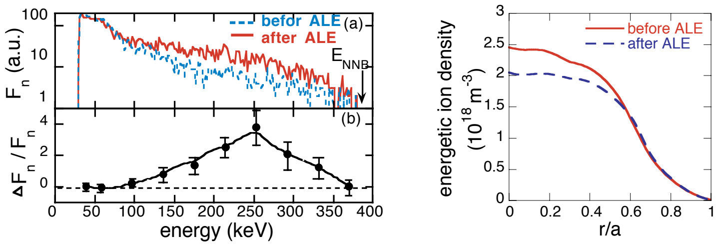

FIG. 12: Left: Energy spectrum of (a) neutral particle fluxes before and after ALE, (b) the fraction of enhanced neutral particle fluxes due to ALEs. Right: Energetic ion density profiles before ALE at t = 4.642 s and after ALE at t = 4.654 s, which are estimated from the neutron emission profile measurement. The inversion of the energetic ion profiles by ALE can be seen at $\rho \sim 0.45$.

4.3 Electron transport

Transient transport experiments are performed [24]. This research is carried out in collaboration with the National Institute for Fusion Science (NIFS), Toki, Japan, in order to compare experimental results between a tokamak (JT-60U) and a helical device (Large Helical Device: LHD) to find out common physics in a torus plasma or in more general sense. The dependence of $\chi_{\rm e}$ on the electron temperature ($T_{\rm e}$) and the electron temperature gradient ($\nabla T_{\rm e}$) is analyzed by an empirical non-linear heat transport model. In an OH plasma and low power NB heated L- and H-mode plasmas, two different types of non-linearity of the electron heat transport are observed from cold/heat pulse propagation utilizing pellet injection and ECRF injection. It is found that $\chi_{\rm e}$ depends not only on $T_{\rm e}$ but also on $\nabla T_{\rm e}$ in JT-60U, while $\chi_{\rm e}$ depends only on $T_{\rm e}$ in LHD.

4.4 Disruption and runaway electron generation mitigation

Fast plasma shut-down and mitigation of disruption are key issues for a safety operation. Puffing noble gas and intense hydrogenic gas is found to be effective to reduce divertor heat load and suppress runaway electrons generation [25].

4.5 Innovative operational concept development

In order to reduce the construction cost and the weight of a tokamak fusion reactor, it is beneficial to remove a massive center solenoid (CS). In such a case, it is necessary to develop a scheme to initiate and ramp up $I_{\rm p}$ without CS. In the last IAEA conference, we reported that Ip could be initiated and ramped up without the CS (F-coil in JT-60U), by RF (ECRF and LHRF) and induction from poloidal coils only, and high confinement high $f_{\rm BS}$ plasma was obtained by injecting NB in such a plasma [26,27]. However, since some of the poloidal coils ('VT-coil' that is used to control the triangularity) have turns inside the torus the situation was not complete CS-free. Further optimization with the inside turns of VT-coil disconnected, it is demonstrated that $I_{\rm p}$ of about 100 kA can be formed with ECRF and flux injection by the outboard coils only (genuine CS-less condition) [28].

5 Pedestal, SOL and divertor studies towards particle/heat handling

The extended heating phase of a discharge has unveiled phenomenon that had not been experienced when the heating duration had been limited within 10 s. One of the most distinct ones is saturation in the wall retention. That is the wall recycling rate keeps increasing to unity. The increase in the recycling can affect plasma performance. The divertor pumping plays an important role in preventing wall saturation to degrade performance. Understanding of ELMs is a key to mitigate particle/heat load to the divertor for a steady-state operation. Characteristics of ELMs have been investigated with emphasis of the plasm toroidal rigid rotation.

5.1 Extension of high recycling ELMy H-mode

One of other important issue in long pulse operation is change in wall recycling that has longer time scale than the current diffusion. The term 'wall' in this paper mainly means the divertor tiles that are mede of carbon in JT-60U. A typical effect of increase in wall recycling can be seen, for example, in Fig. 3. The H factor gradually decreases (Fig. 3 (c)). At the same time the intensity of the D_α brightness gradually increases (Fig. 3 (b)). This link between the confinement and D_α intensity is often observed in a JT-60U long pulse discharges. Therefore, the degradation of confinement is though to be attributed to the increase in recycling. After sufficient wall conditionig, typically glow discharge of several to more than ten hours, base level of the D_α intensity becomes much lower and does not increase much. However, sooner or later the D_α intensity, that is the wall recycling, is expected to become large enough to disturb desired confinement. The uncontrollable continuous increase in the wall recycling in a discharge like one show in Fig. 3 can be attributed to a fact that such a plasma configuration is not suitable for divertor pumping in JT-60U. The divertor section of JT-60U is so-called 'W-shaped' divertor as shown in Fig. 13 Ref. [29]. The pumping slot are located at the end of the both wing of the dome. It is necessary to set divertor legs closer to the slots in order to make pumping efficient, since neutral pressure at the slot that should be high enough for pumping does not increase if the legs are far off even by several centimeters. For a discharge like E43903 in which higher $\beta_{\rm N}$ is intended, it is necessary to raise the triangularity for higher pedestal height. However, in such a plasma the hit point of the inner divertor leg tends to move upwards in JT-60U. Furthermore,

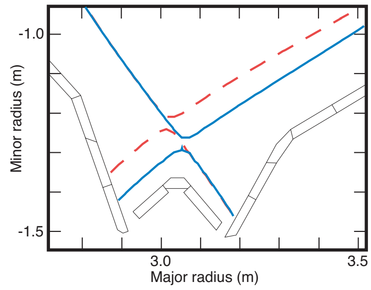

FIG. 13: Magnetic configurations around the divertor of a high (0.36) and a low (0.27) trianglerity plasma, shown by a solid and a broken line, respectively.

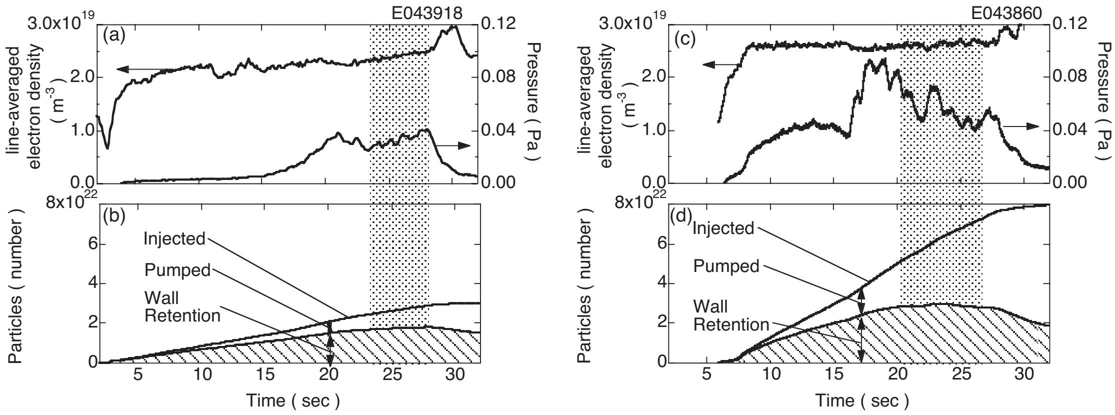

FIG. 14: (a) line-averaged electron density, neutral pressure in the main chamber, (b) number of injected particles (neutral beam + gas), pumped (divertor-pumping), and retained in the wall for a high triangularity ($\sim$ 0.36) plasma. (c) and (d) for a low triangularity ($\sim$ 0.27) plasma. Shaded regions indicate periods of wall-pumping rate of $\sim$ 0.

even if the legs are close enough to the slots, pumping is not efficient for plasmas with low fueling such like E43903. Due to these reasons, it is very difficult to make divertor pumping efficient in a discharge like E43903 and to demonstrate if divertor pumping can suppress recycling low enough to maintain good confinement. In order to make the effectiveness of divertor pumping clearer, experiments are carried out in a high recycling ELMy H-mode plasma [30]. In Fig. 14 shown is a comparison between a discharge without (left hand side) and with (right hand side) divertor pumping. When the wall is almost saturated, in other words when the wall retention is almost saturated (shown with a shade belt), without divertor pumping the electron density and the neutral pressure in the main chamber keep increasing, while with pumping the electron density stays almost constant and the pressure decreases. It shoud be noted here that the saturation in wall retention becomes observable in JT-60U after the extension of the pulse length. As shown in Fig. 14, the saturation does not occur within 15 s which was conventional pulse length in JT-60U. As plotted in Fig. 15, as the divertor pumping rate increases the incremental rate of the total electrons (dN_e/dt, whiere N_e is the number of the total electrons in the plasma) monotonously decreases. This means that with sufficient pumping, the electron density can be controlled even if the wall is saturated.

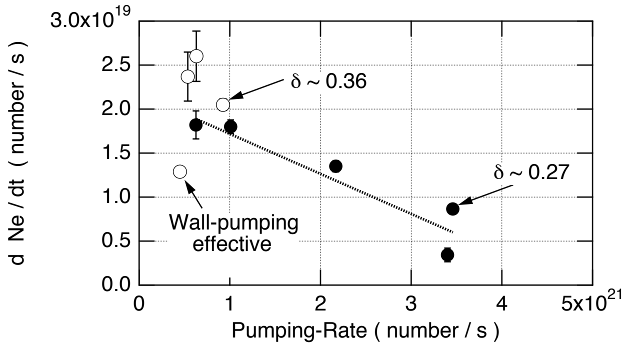

FIG. 15: Increase of number of electrons in the core plasma as a function of divertor-pumping rate under the condition of wall-pumping rate of ~ 0.

5.2 H-mode pedestal, ELMs and divertor compatibility

Mitigation of particle/heat load to the divertor is a critical issue for a steady-state operation. Towards mitigation of ELM impact, characteristics of various ELMs have been investigated, especially on energy lost at each ELM pulse and impact of toroidal plasma rotation on ELMs by utilizing co and counter parallel NBs. The energy loss for the grassy ELM has been studied to investigate the applicability of the grassy ELM regime to ITER [31]. The grassy ELM is characterized by the high frequency periodic collapse up to $\sim$ kHz, which is $\sim$ 15 times faster than that for type I ELM. A divertor peak heat flux due to grassy ELMs is less than 10% of that for type I ELMs. This smaller ELM heat flux is caused by narrower radial extent of the collapse of temperature pedestal. The dominant ELM energy loss for grassy ELMs seems to be conductive loss, and its ratio to the pedestal stored energy was 0.4–1%, which is smaller by a factor of about 10 than that for type I ELMs of 2-10%. ELM size and type can be changed from type I ELM to high frequency grassy ELM as increased CTR plasma rotation (Fig. 16).

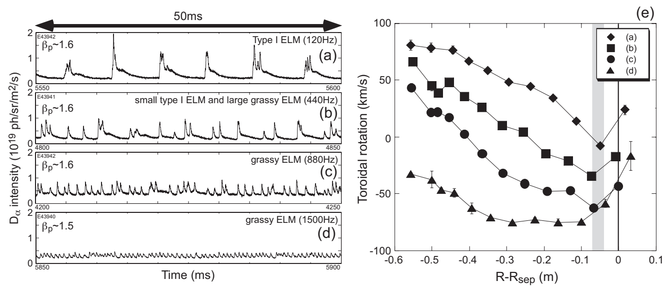

FIG. 16: (a)-(d) Time evolution of D_α signal during plasma toroidal rotation scan (q_{95} $\sim$ 4.9 and δ $\sim$ 0.59). (e) Toroidal rotation profiles mapped into outer midplane measured with charge-exchange recombination spectroscopy. Shaded region shows area of top of T_i pedestal. Plasma rotation profiles were changed by using different combination of NBs: (a) 2CO+2perp+2NNB, (b) 2CO+3perp+1NNB, (c) 2CO+5perp and (d) 1CO+1CTR+5perp.

The complete ELM suppression (QH-mode) has been achieved using counter and perpendicular NBIs, when the plasma position was optimized. The existence of the edge fluctuations localized pedestal region may reduce the pedestal pressure, and therefore the QH-mode can be sustained for long time up to 3.4s ($\sim$ 18 $\tau_{\rm E}$). A transient QH phase was also observed during the CO-NB injection phase with almost no edge toroidal rotation, which is a similar condition in ITER.

A series of collaborative experiments between JET and JT-60U has been performed for the first time to compare the H-mode pedestal and ELM behaviour in the two devices [32]. Given their similar size, dimensionless matched plasmas are also similar in their dimensional parameters such as minor radius a, except for the aspect ratio by $\sim$ 15%. Contrary to expectations, a dimensionless match between the two devices has been quite difficult to achieve. The toroidal rotation profile and the magnitude of the toroidal ripple field in the large bore plasma used in JT-60U for the similarity experiments are identified as the main difference with JET. When ripple induced fast ion losses were reduced by factor of 2 by replacing Positive NBIs with Negative-ion based NBIs, higher p_e^{ped} providing a good match to full power JET H-modes was achieved in JT-60U.

5.3 SOL plasma and plasma wall interaction

Understanding of parallel and perpendicular transport during an ELM in the SOL region is important. It is found that radial velocity of the SOL plasma expansion is 1-3 km/s at the low-field-side, which is larger than that at the high-field-side. Analysis of the first wall tiles shows that hydrogen and deuterium retention in divertor region is mostly $<$ 0.05 in (H+D)/C ratio that is less than most of observation in other tokamaks probably due to high wall temperature (573 K) in JT-60U [33].

6 Summary

With extension of JT-60U pulse length in addition to the continuous effort in conventional pulse length, successful progress has been made on development and understanding of AT relevant plasmas and their issues towards steady state operation. Sustainable duration of high $\beta_{\rm N}$ that is comparable to the ITER advanced operation domain has been extended to 15.5 s with $\beta_{\rm N}$ = 2.5. Duration of higher $\beta_{\rm N}$ (= 3) sustainment which is rather closer to the reactor domain has been extended to 6.2 s. High non-inductive current drive duration with high $f_{\rm BS} \sim$ 50% is extended to 2.3 s. High bootstrap current fraction RS plasma was demonstrated to be maintained for 7.4 s. Long pulse ($<$ 30 s) high recycling ELMy H-mode plasma was obtained. Effectiveness of divertor pumping was studied on that plasma and demonstrated. Compatibility of AT relevant plasmas was investigated. High confinement and high radiation were realized at high density region in both weak and reversed magnetic shear plasmas. Injecting ECRF in prior to an NTM became saturated was found to be more effective to stabilize the mode than injecting ECRF after the mode got saturated. Genuine CS-less tokamak plasma start-up was demonstrated. It was shown that no current could be driven by any scheme (inductively by toroidal electric field, non-inductively by external current drivers such as ECCD and N-NBCD) in the current hole region. Detailed measurement of energetic ions in both real and energy spaces showed spatial redistribution of energetic ions resonating with the collective mode that is induced by these energetic ions themselves. Energy lost at each ELM was found to be smaller in Grassy ELM domain than that in Type I ELM domain. Impact of the toroidal rotation on both Type I and Grassy ELMs were found.

Acknowledgement

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

Reference

[1] M. Kikuchi, Nucl. Fusion **30**, 265 (1990).
[2] T. Fujita and the JT-60 Team, Nucl. Fusion **43**, 1527 (2003).
[3] *ITER Technical Basis*, no. 24 in ITER EDA DOCUMENTATION SERIES (IAEA, Vienna, 2002).
[4] A. Isayama, *et al.*, Nucl. Fusion (2003).
[5] T. Suzuki *et al.*, this conference, EX/1-3 .
[6] A. Isayama, *et al.*, Plasma Phys. & Controlled Fusion **42**, L37 (2000).
[7] ITER Physics Expert groups on Confinement and Transport, Confinement Modelling and Database, and ITER Physics Basis Editors, Nucl. Fusion **39**, 2137 (1999).
[8] Y. Kamada and JT-60 team, Nucl. Fusion **41**, 1311 (2001).
[9] Y. Sakamoto *et al.*, this conference, EX/4-3 .
[10] T. Fujita, *et al.*, Phys. Rev. Lett. **87**, 085001 (2001), 5+10.
[11] K. Nagasaki, A. Isayama, S. Ide, and JT-60 Team, Nucl. Fusion **43**, L7 (2003).
[12] K. Nagasaki *et al.*, this conference, EX/7-4 .
[13] T. Oikawa *et al.*, this conference, EX/P5-15 .
[14] S. Takeji, *et al.*, Nucl. Fusion **42**, 5 (2002).
[15] Y. Ishii, M. Azumi, and Y. Kishimoto, Phys. Rev. Lett. **89**, 205002 (2002).
[16] M. Takechi *et al.*, this conference, EX/P2-32 .
[17] H. Takenaga *et al.*, this conference, EX/6-1 .
[18] T. Fujita *et al.*, this conference, EX/P4-3 .
[19] J. A. Breslau *et al.*, Phys. Plasmas **10**, 1665 (2003).
[20] N. Hayashi *et al.*, this conference, TH/1-6 .
[21] K. Shinohara, *et al.*, Nucl. Fusion **41**, 603 (2001).
[22] M. Takechi, *et al.*, in *19th IAEA Fusion Energy Conference Lyon, 2002* (IAEA, Vienna, 2003), pp. EX/W–6.
[23] M. Ishikawa *et al.*, this conference, EX/5-2Rb .
[24] S. Inagaki *et al.*, this conference, EX/P2-12 .
[25] M. Bakthiari *et al.*, this conference, EX/10-6Rb .
[26] Y. Takase, *et al.*, in *19th IAEA Fusion Energy Conference Lyon, 2002* (IAEA, Vienna, 2003), pp. PD/T–2.
[27] S. Shiraiwa, *et al.*, Phys. Rev. Lett. **92**, 035001 (2004).
[28] Y. Takase *et al.*, this conference, EX/P4-34 .
[29] N. Hosogane, *et al.*, J. Nucl. Mat. **266-269**, 296 (1999).
[30] T. Nakano *et al.*, this conference, EX/10-3 .
[31] N. Oyama *et al.*, this conference, EX/2-1 .
[32] G. Saibene *et al.*, this conference, IT/1-2 .
[33] T. Tanabe *et al.*, this conference, EX/P5-32 .