Current Status of Experimental Study and Device Modifications in JT-60U

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Abstract—Since tokamak magnetic fusion research has just made a step forward to an international collaborative project ITER, the existing tokamaks including JT-60 are expected to explore more advanced operation scenarios. To test those scenarios in the JT-60 experiment, the discharge pulse length and the duration time of additional NBI/RF heating were extended to 65 s and 30 s /60 s, respectively, in 2003 with modification of the corresponding control systems for power supplies and heating devices. The experimental campaign in 2003-2004 after the above modifications has ended up with the following significant results: (a) The high bootstrap current ratio of 75 % was sustained for 7.4 s in an R/S plasma. (b) Normalized beta value of 2.3 was also done for 22.3 s in a high-beta H-mode plasma. (c) The quasi-steady state beta value was increased to 3.0 with a pulse of 6.2 s with NTM suppression by ECCD. For further exploration toward high performance plasmas, the following modifications will be or has been conducted: (1) To minimize the power loss from a plasma at the region of toroidal field ripple, the 8Cr ferritic steel tiles, having a similar magnetic property to the low activation ferritic material for a DEMO reactor, are being equipped on the first wall of the JT-60 vacuum vessel. (2) Since plasma shape and current profile in the poloidal cross-section are expected to be reproduced in real time to optimize a plasma performance with suppressing the plasma instabilities, a precise reproduction method has been installed in the plasma control system. In this report, the current status of plasma experimental study will be presented together with on-going device modifications in JT-60U.

Keywords—tokamak; JT-60U; advanced operation; plasma experiment; device modifications; ferritic steel; plasma current profile; plasma shape control

INTRODUCTION

Tokamak magnetic fusion research has just made a step forward to construction of an international collaborative device ITER. In such a new phase, the existing tokamaks including JT-60U are expected to focus on challenge of various experimental issues, aiming at exploration of more advanced operation with a high plasma pressure and a high bootstrap current fraction toward a future power reactor.

In order to demonstrate steady state operations toward advanced tokamak for a much longer period than the plasma current relaxation time in JT-60U, the control systems for the coil power supplies, the NBI and RF heating facilities were modified in the shutdown period of July 2002 to Nov. 2003. This led to that the pulse lengths of tokamak discharge and additional heating operation were extended within an allowable limit of the electric and thermal capacities in the power systems [1]. The target performance after the modification are shown in Table 1. To attain these numbers, we have been solving technological issues relating to the thermal limitations for the following heating systems available in JT-60U: The positive- and negative-ion-based NBI (P/N-NBI) [2], the electron cyclotron wave heating (ECH) system [3], and the lower hybrid wave heating (LHH) system [4]. These heating devices were employed to produce advanced plasma operation during the last experimental campaign of Dec. 2003 through Nov. 2004, and the achieved numbers are also described in Table 1.

On the background of facility modifications mentioned above, various important challenges were conducted on the important topics having the following keywords: high beta, (quasi-)steady state, high bootstrap fraction, high density, etc. Furthermore, these experiments have been supported by the sophisticated plasma real-time feedback control systems: (a) plasma equilibrium control system including the magnetic measurements, complete plasma shape reconstruction, identification of plasma current centroid, and toroidal/poloidal field coil power supplies, and (b) particle supply and heating devices control system including plasma diagnostic systems.

For further exploration to the advanced tokamak scenario, a drastic change is installation of ferritic steel plates partly as the first wall of the vacuum vessel from original carbon tiles,

| TABLE 1. Target performance and achievements after modification |
|-------------------|------------|-----------------|
| **Item**            | Normal pulse | Long pulse       |
| Tokamak discharge   |             |                 |
| Max. pulse length   | 15 s        | 65 s            |
| Bt flat top duration| 4 T, 8 s    | 3.3 T, 30 s & 2.7 T, 60 s |
| Total Input Energy to the plasma | 200 MJ (achieved) | 350 MJ (achieved) |
| P-NBI               |             |                 |
| Perp. Inj. 7 units  | 18 MW, 10 s | 18 MW, 10 s     |
| Tang. Inj. 4 units  | 12 MW, 10 s | 8 MW, 30 s      |
|                      | [total 22 MW, 10 s] | [achieved]       |
| N-NBI               |             |                 |
| (with 2 units)      | 10 MW, 10 s | 10 MW, 10 s     |
|                      | [5.8 MW, 10 s (400 keV)] | [achieved]       |
|                      | 2 MW, 30 s  | 2 MW, 30 s      |
| ECH                 | 3 MW, 5 s   | 0.6 MW, 30 s    |
|                      | 3 MW, 2.8 s | [1 MW, 15 s]    |
| LHH                 | 3.5 MW, 10 s| 2.5 MW, 30 s    |

note: [ ] shows the achieved number

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aiming at reduction of the toroidal magnetic field ripple.

For improvement of plasma control capabilities, real-time reconstruction of plasma current profile has been investigated as one of the key issues for control of a tokamak plasma. Since magnetic measurements in a plasma exterior provide the most reliable information necessary for profile reproduction, a method to solve this inverse problem with the measurements was proposed with the eigenfunction expansions [5, 6]. In addition, a new method has been installed and applied to the actual real-time feedback control frame. This method provides approximate plasma q-value profiles along the cord of the MSE measurements without solving the equilibrium equation [7].

**Experimental Study After the Modifications for Long Pulse Discharge**

As described above, modifications of control systems in coil power supplies, heating systems and diagnostics systems of the JT-60U facilities were successfully completed without major hardware upgrade [1]. JT-60U has proceeded to the research of advanced tokamak operation in long pulse discharges. In the coil systems, the available duration and current are limited by capacities in the existing power supplies and thermal limitation of coils. The toroidal field is up to 3.3 T for 30 s and 2.7 T for 60 s, as shown in Table 1. The plasma current is limited by the available flux swing of ohmic heating and equilibrium field coils. Fig. 1 shows a typical discharge waveform with a plasma current \( I_p \) of 0.7 MA and a toroidal magnetic field \( B_t \) of 2.0 T. The divertor configuration was maintained for 60 s during the plasma current flat top. The flux consumption was saved by reducing the loop voltage by 30 s NB and EC heating, and by EC-assisted breakdown. The maximum \( I_p \) maintained for 30 s is 1.4 MA. Careful adjustment of plasma configuration reduced the current in equilibrium field coils, which was necessary for effective increase of \( I_p \).

The positive-ion-based NB (P-NB) system in JT-60U consists of 11 units; 7 units for perpendicular injection, 2 units for co-tangential injection and 2 units for counter-tangential injection. Here, co and counter mean parallel and anti-parallel to the direction of the plasma current, respectively. The typical beam energy is 80-85 keV. The pulse duration of 4 tangential injection units has been extended to 30 s [8], which has been achieved after port conditioning to reduce the re-ionization loss to the port duct. The beam limiters for these units have been modified to increase the heat capacity and to widen the beam-facing area to allow extended pulse length. The pulse duration of other 7 perpendicular injection units is unchanged 10 s, except for a unit for the charge exchange recombination spectroscopy (CXRS) measurements. They can be injected in series for 30 s. As a result, 6 units (12 MW) of P-NB can be injected continuously for 30 s; the injection power per unit is reduced from 2.5 MW to 2 MW in the long pulse operation due to the limitation of power supplies.

The negative-ion-based NB (N-NB) system consists of 2 co-tangential units. The ion sources were modified to increase gas conductance in the accelerator, which reduced the heat load to the grounded grid due to the stripping loss [2]. The achieved pulse length is 25 s with beam energy of 350 keV and the beam power of ~1 MW.

In the RF systems, lower hybrid (LH) and electron cyclotron (EC) range of frequency, the possible maximum duration has been extended up to 60 s. In the LH heating (LHH) system, a grill mouth made of carbon was developed and bolted to the original stainless steel grill to allow long pulse operation [4]. The achieved pulse length is 18 s with ~0.9 MW. In the EC heating (ECH) system, a technique of controlling the anode voltage in the gyrotron has been developed for sustaining the oscillation conditions against the beam current decay during a long pulse operation [3]. A 16-s oscillation has been achieved with gyrotron power of 0.5 MW and a 44.6-s heating pulse has been obtained using four gyrotrons in series. As a result from extension of the heating period, the total input energy into the vacuum vessel reached 350 MJ. Although plasma-facing components were not actively cooled in JT-60U, no carbon bloom was observed even with such high injected energy, which is likely to be the effect of the W-shaped divertor.

### A. High-\( \beta_N \) long pulse Discharge[9]

To increase \( \beta_N \) to the value larger than 3.5 and to increase the fraction of the bootstrap current relative to the total plasma current, \( f_{BS} \), to the value of larger than 70% are key issues to develop an advanced tokamak operation requested for a compact steady-state demo reactor. These issues should be solved in a steady-state time domain. Two characteristic time scales are introduced in the discussion: the current diffusion (relaxation) time, \( \tau_R \), and the wall particle saturation time, \( \tau_{wall} \). Here, \( \tau_R = \mu_e \langle \sigma_n \rangle \tau_{wall} \), where \( \langle \sigma_n \rangle \) is the volume-averaged of plasma neoclassical conductivity., and a is the plasma minor radius [10]. The advanced tokamak operation had not been carried out beyond these time scales in the world. In JT-60U plasmas, the value of \( \tau_R \) is around a few to a few tens of seconds and the value of \( \tau_{wall} \) is the order of several tens of seconds. Before 2002, most of achievements were in the time scale near \( \tau_R \) and shorter than \( \tau_{wall} \).

The beta limit in a long pulse discharge is mainly constrained by neoclassical tearing modes (NTMs). Since the stability of NTM depends on the current profile, the duration of high beta plasma beyond the current diffusion time \( \tau_R \) is crucial for feasibility of long sustainment of high beta. The experiment was performed using high \( \beta_N \), H-mode plasmas. We had achieved 7.4 s sustainment of \( \beta_N \approx 2.7 \) in this regime with the conventional 10 s heating [11]. However, sudden appearance of NTM limited performance in some cases, which suggests that
the duration was insufficient compared to the current profile relaxation time. By optimization of discharge with the newly extended NB heating, $\beta_N$ of 2.3 has been successfully maintained for 22.3 s, as shown in Fig. 2. This duration corresponds to $13\tau_R$. We had another good pulse, where a higher $\beta_N$ of 2.5 was maintained for 15.5 s ($9.5\tau_R$) with $I_p = 0.9$ MA, $B_t = 1.7$ T, and $q_{95} \sim 3.4$ [7]. The initial rise of $\beta_N$ was carefully optimized in order to reach as a high value as possible avoiding NTM appearance. In fact, NTMs appeared in some discharge pulses with faster $\beta_N$ rise. The key point is to form a pressure profile with moderate pressure gradients at $q = 1.5$ and 2 surfaces at which NTMs are likely to occur. The $\beta_N$ value after $t = 5.0$ s was kept constant, which was achieved by real-time feedback control of stored energy using P-NB. This discharge also had high confinement $H_{99} = 1.9-2.3$ ($H_{99}$ is the confinement enhancement factor over the L-mode scaling [12]) at a normalized density of $n/e_n_{LH} \sim 0.4-0.5$, and a factor $G = \beta_N H_{99}/q_{95}^2$, a figure of merit of fusion performance (fusion gain and fusion power), was kept 0.4-0.5. The bootstrap current fraction $f_{BS}$ was $\sim 30-35\%$. Considering higher G value at a higher $q_{95}$ than ITER baseline scenario ($G = 0.4$, $q_{95} \sim 3$), this discharge can be categorized as a demonstration discharge for the ITER hybrid operation scenario that aims at extended duration of high fusion power at a reduced plasma current [13].

In this discharge, the current profile reached a stationary state without appearance of sawteeth and NTMs. The profile of plasma current reached a stationary state. The $q$ profile is nearly flat in the central region ($r/a < 0.4$) with $q_{95} \sim 1$.

The progress of high $\beta_N$ sustainment is summarized in Fig. 3. The duration with $\beta_N$ of 2.5 is extended to 16.5 s while that with $\beta_N = 1.9$, corresponding to the ITER standard operation, is remarkably extended to 24 s. A higher $\beta_N$ value of 3 was maintained for 6.2 s with a larger heating power of 23-25 MW. In this discharge, lowering $q_{95}$ down to 2.2-2.8 during NB heating seems to be effective to maintain high $\beta_N$ by shifting $q = 1.5$ and 2 surfaces outward [7]. Sustainment of high $\beta_N$ of $\sim 2.9$ at $q_{95} \sim 3.5$ for 5 s was also achieved with NTM stabilization by the ECH [14, 15].

### B. Nearly full Current Driven Plasmas[9]

Steady state operation of tokamaks requires full non-inductive current drive with a large fraction of bootstrap current. The required $f_{BS}$ is $\sim 50\%$ in ITER steady-state operation [16] with $Q = 5$ and $>70\%$ in a reactor [17]. In this regime, full and close to full non-inductive current drive with $f_{BS} \sim 50\%$ had been achieved with the short duration ($\sim 1$ s) in other tokamaks [18, 19]. A quasi-stationary ITB discharge in JET was maintained for several seconds under highly non-inductive current drive with LHCD [20]. The real-time control of pressure profile was performed with moderate $\beta_N$ of 1.7 and $f_{BS} < 50\%$. We have attempted to extend the duration of nearly full non-inductive current drive with a large $f_{BS}$ in a higher $\beta_N$ regime by the high $\beta_v$ H-mode and in a larger $f_{BS}$ regime by the reversed shear plasma. Experiments were performed in conventional 10 s heating discharges with higher available power.

The reversed shear plasma is characterized by capability of a very high bootstrap current fraction. In JT-60U, $f_{BS}$ of 80% was maintained under full non-inductive drive conditions previously [24], but the duration was limited to 2.7 s ($<\tau_R$). Though $q_{95}$ was nearly constant during this period, evolution of current profile in a longer time scale had been our concern. By optimizing a similar discharge ($I_p = 0.8$ MA, $B_t = 3.4$ T, $q_{95} \sim 8.3$), $f_{BS}$ of 75% and $\beta_N \sim 1.7$ ($\beta_v \sim 2.4$) were maintained for 7.4 s ($2.7\tau_R$) until the end of NB heating as shown in Fig. 4 [22]. The control of toroidal rotation was employed to reduce the pressure gradient at the ITB foot when passing $q_{95} = 4$ around $t = 7$ s. The radiation power remained constant and no strong impurity accumulation was observed; the fraction of radiated power to the heating power remained $\sim 5\%$ in the core region ($r/a < 0.4$). Due to large radii of $q_{95}$ and ITB foot as shown in Fig. 5, together with the H-mode pedestal at high triangularity ($\delta \sim 0.42$), very high confinement of $H_{99} \sim 3$ and $H_{12} \sim 1.7$ was maintained. The calculated non-inductively driven current amounts to 95% of plasma current (75% bootstrap and 20% beam-driven), and its profile is close to the measured total current density as shown in Fig. 4. The pressure and current profiles reached stationary conditions in a later phase of this duration. This is the first experiment in which stationary conditions in the current and pressure profiles have
been obtained in a plasma with a bootstrap current fraction relevant to the steady-state tokamak reactor.

**CONTROL TOOLS TOWARD ADVANCED TOKAMAK OPERATION**

Experimental optimization toward advanced tokamak operation described above has been conducted with the sophisticated plasma control system. Generally speaking, it should be noted that real-time feedback control could provide a new dynamics on the controlled parameters to resolve the issues. Until now, many control functions developed for the JT-60 experiment have contributed to produce new results. The following two tools have already been working for JT-60U in software as well as in hardware: (a) A complete plasma shape is precisely reproduced by the method of CCS (Cauchy condition surface) as an inverse problem in real time, and plasma current centroid (near magnetic axis) of the current is calculated also in real time. And (b) Plasma $j$ or $q$ profile is precisely reproduced using the shape, magnetic data and MSE diagnostics in real time. Employing the LHRF system, the $q$-profile feedback control have been successfully demonstrated. Now in this section, we explain these tools in detail.

A. **Plasma Shape Reconstruction by the CCS method**

Plasma shape in a tokamak often determines energy confinement performance, stability, and safe operation. Recent 10 years, remarkable advances in understanding of shape reproducibility have been made [25, 26]. Since the shape can be defined as an outermost magnetic surface, the problem of shape reproduction is to find magnetic field distribution from the measurements. From the mathematical point of view, this is a boundary value problem for a static Maxwell equation that is classified into a Dirichlet or Cauchy problem of the second-order partial differential equation (PDE). In the axisymmetric tokamak geometry, there exists an integral formula of the exact analytical solution with a Green function [25]. On the basis of this consideration, we have developed the method of CCS (Cauchy condition surface) as an inverse problem [26]. The concept of this method is summarized below. We start with the static Maxwell’s equation, $\nabla \times B = \mu_0 j$, in a toroidal axisymmetric region, that analytically yields the integral relation by the introduction of Green function. The choices of the integral region with boundaries and the observation point yield the different sets of equations. The Cauchy condition (=both Dirichlet and Neumann conditions) on the innermost closed boundary (we call “Cauchy condition surface (CCS)”) is calculated using equipped magnetic sensors in a sense of least-squares with neglecting eddy currents in the nearby conductors. Once the Cauchy condition is determined, the flux distribution in the CCS exterior provides the precise shape reconstruction.

Calculations of special functions and matrix inversion, that are unavoidable, could not be done within a short period enough for real-time application. Magnetic flux at a certain point is provided by linear combination of sensor signals, whose coefficients depend on the point location. The coefficient vectors are stored in a tabular form (occupied in the core memory of approximately 0.5 Gbyte for the case of 1-cm mesh) prior to the discharge. In real-time, necessary number of scalar products of signals and coefficients vectors provide flux distribution and plasma shape, as shown in Fig. 5. It takes 1.0 ms for the 4-parallel computing system with 0.5-GHz CPUs to reproduce one shape [6]. The reproduced position and shape are directly utilized for their feedback control. In addition, the full plasma shape is visualized on the screen as a real-time monitor. Since magnetic field surrounding a plasma is determined through shape reproduction, plasma total current is calculated by line integral of vacuum magnetic field along a fixed closed line enclosing a plasma at the same time of shape identification. This process can be executed after positioning the CCS at the plasma current centroid inside the plasma. Other quantities related to plasma shape such as the Shafranov lambda, the safety factor, etc., are also obtained.

B. **Plasma Current Profile Control**

Although extension of the pulse duration mentioned in the above section has been achieved without feedback control of current profile, it will be necessary for longer sustainment of advanced tokamak plasmas with high $\beta_s$ and high $f_{BS}$. The real-time $q$-profile was demonstrated in the JET tokamak for the first time using the interferopolarimetry for the $q$ profile evaluation and LH for the current driver. In JT-60U, a real-time $q$-profile control system has been developed using the motional Stark effect (MSE) diagnostics as a detector and the LHCD as an actuator [9]. The $q$ profile was successfully obtained in real time every 10 ms by using local pitch angle measurement with MSE for the first time [27]. Here, the equilibrium is not reconstructed in real time, but $q$ profile was estimated under an assumption that the shape of internal flux surfaces are similar to the shape of the last closed flux surface determined by the CCS method. The location of magnetic axis is also given as a weighted center of current density by the CCS method. The agreement between $q$ values by the simplified real-time calculation and those by full equilibrium reconstruction was confirmed in a wide range of plasma parameters. The $q$ profile was controlled towards the reference profile with the control of CD location by changing the parallel refractive index $N_p$ of LH waves. The phase difference of LH waves between multi-junction launcher modules was changed in real time to change $N_p$ and hence the CD location. The LH power can be also controlled to keep the amount of LH-driven current constant considering the $N_p$ dependence of the current drive efficiency. This logic seems to be relevant to the current profile control in full non-inductive current drive plasmas.
The LH was injected into a discharge with \( I_p \) of 0.6 MA, \( B_t \) of 2.3 T, \( \psi_{50} \approx 6.3 \), \( q(0) \approx 1 \) (sawtoothing) and a line-averaged density of \( 0.5 \times 10^{19} \text{ m}^{-3} \). The reference \( q \) profile was set to a monotonic increasing profile with \( q(0) = 1.3 \) as shown by a solid line with small diamonds in Fig. 6 (b). After \( t = 10 \text{ s} \), a power of about 1 MW was almost continuously injected with a good wave coupling and the loop voltage decreased to zero level. During this period, the \( q = 1 \) surface vanished and the radius of \( q = 1.25 \) surface decreased as shown in the bottom panel of Fig. 6 (a). The \( q \) profile approached to a reference profile at \( t \approx 13 \text{ s} \) and was sustained for 3 s. The plasma conditions of low beta (\( \beta_p \approx 0.5, \beta_N \approx 0.6 \)) and low density were selected as a first target of demonstration. The \( q \) profile control in higher \( f_{\psi 50} \) and/or higher \( \beta_N \) plasmas is a next target.

**FERRITIC STEEL PLATES INSERTION FOR REDUCTION OF THE TOROIDAL MAGNETIC FIELD RIPPLE [28]**

Further investigation for advanced tokamak operation requires exploration of the following ideas; i) a wall stabilization without losing heating power, ii) a better RF availability to control a current profile with avoiding large heat load on the RF antennas from escaping energetic ions, iii) source of rotation control to stabilize MHDs such as a resistive wall mode (RWM), and iv) more absorbed heating power and more efficient current drive by increasing the confinement of NB ions. For the first two, we need to use a large volume plasma, which suffers from large ripple induced loss in JT-60U. This loss reduces the confinement of energetic particles, which could have a great influence on the heating, current drive, and rotation source. The loss also causes large heat load on RF antennas, which could limit the pulse length and the coupling with a plasma. Thus, the available heating or current drive source by the RF system is limited. In addition, due to the ripple induced loss, a counter rotation is produced even during perpendicular NBI alone. The counter rotation prevents the peripheral region from the core rotation, which is expected to be a key of the MHD stability and transport in the pedestal region in H-mode plasmas. It is relatively difficult to control rotation profile in a plasma with a large ripple induced loss. Because of these limitations, the plasmas with a large volume have not been used.

The ripple reduction is expected to bring about 4 benefits, previously mentioned. The wall stabilization and better RF efficiency are mainly obtained in a large volume plasma. Thus, our assessment has been carried out intensively for the large volume plasma configuration. As a basic configuration in the discussion, a plasma with \( 1.1 \text{ MA}, \beta_N = 1.9, n_e \approx 1.5 \times 10^{19} \text{ m}^{-3}, B_t = 1.86 \text{ T} \) was employed, as shown in Fig. 7 [29]. At first, the reference results were made without ferritic steel plates. The loss of perpendicularly injected beams is large and 63% of the deposited power. The loss of tangentially injected beams is also not negligible small. In the distribution of escaping energetic ions, most of the escaping ions hit on the first wall on the mid-plane of plasma as an orbit loss in this configuration. The enhanced confinement of energetic ions has been assessed for the NB ions by using the Full 3-D OFMC code.

We chose the ferritic steel with the ingredient of 8Cr-2W-0.2V. This has a similar saturation magnetization (1.7 T at 573 K) to F82H ferritic steel (8Cr-2W-0.2V-0.04Ta), which is a low-activation ferritic steel developed by JAERI [29].

As a result from the analysis, the 18-fold toroidal symmetry was lost in the magnetic field. The bird’s eye view of the ferritic inserts is depicted in Fig. 8. In comparison of with and
Discharges with high \( f_{\text{BS}} \) confinement for ITER hybrid and informative discussions, suggestions and comments. to the research and engineering staffs in JAERI for their operation toward ITER and a fusion DEMO reactor.

Remarkable results were attained: (a) High \( \beta_N \) sustainment. In \( 60U \) control system, various significant experimental results have been produced. 65-s pulse discharge has been completed. We will restart experiments in Nov, 2005. Now we are ready to make further explorations into AT plasma operation toward ITER and a fusion DEMO reactor.

CONCLUDING REMARKS

With drastic modifications and improvements on the JT-60U control system, various significant experimental results have been produced. 65-s pulse discharge has been successfully achieved with 30-s H-mode sustainment. In extending advanced tokamak (AT) operation, the following remarkable results were attained: (a) High \( \beta_N \) H-mode plasma with \( \beta_N \approx 2.3 \) for 22 s. (b) RS ELMy H-mode plasma with \( f_{\text{BS}} \approx 75\% \) and \( \beta_N \approx 1.7 \) for 7.4 s under nearly full (95\%) CD. (c) \( \beta_N \approx 3 \) for 6.2 s with suppressing NTM by ECCD or by flattening q profile. For AT operation with a higher \( \beta_N \) and \( f_{\text{BS}} \) in longer pulse discharge, plasma precise equilibrium observer has been developed. Current profile control has been demonstrated. Ferritic steel tile insertion for TF ripple reduction has been just developed. Current profile control has been demonstrated.

Ferritic plates (23 mm thick), the ripple well area for trapped particles is reduced to a large extent, as shown in Fig. 9. In the same comparison, the absorbed power is increased by about 1.3 times (53% to 68% of the full NB injection power), and the absorbed power of perpendicularly injected beams is increased by 1.5 times (37% to 57% of the full NB injection power). We also estimated the heat load on the LH antenna region. The heat load was reduced from 0.27 to 0.1 MW/m² in this particular discharge.

With the increases of the absorbed heating power, the MHD stability, the controllability of the current profile, and the rotation profile, we expect 1) longer discharges at high \( \beta_N \), 2) higher \( \beta_N \) beyond an ideal limit of free boundary, 3) longer discharges with high \( f_{\text{BS}} \) confinement for ITER hybrid and steady-state scenarios.

Fig. 10 shows the newest inside view of the JT-60U vacuum vessel, where the first wall of ferritic plates shines differently from the carbon tiles.

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