Physics and Technological Issues for Steady-State Tokamak Operation on TRIAM-1M

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Abstract

Subjects on TRIAM-1M are presented from the viewpoint of plasma physics and reactor technology associated with ‘steady state operation’ (SSO). For a future fusion power plant, for burning plasma, complete steady state operation is required. Control of the density under the complicated plasma-wall interaction, non-inductive current start-up, and sustainment of high performance are key areas in the present and future investigations.

The following experimental results are reviewed. First, it is shown that the co-deposition of the metallic impurity and oxygen plays an important role in the temporary change in the wall pumping rate, and a model of the co-deposition probability agreed with the observation. It was also noticed that the thermal release of the hydrogen from the plasma-facing components affects the steady state density operation in the ultra long discharge. It was found that enhanced influx of metal impurities from the hot spot affect the steady state operation of the high performance plasma. Second, helium effects on microscopic damage on metals were studied in helium/hydrogen mixture discharges. A large quantity of dislocation loops and dense fine bubbles were observed by means of TEM even for exposure only 125 seconds in duration. From TDS for the specimens, the amount of retained helium was evaluated to be $3.9 \times 10^{20}$ He/m$^2$. Third, the physics understanding for the enhanced current drive (ECD) mode around the threshold power level progressed from the viewpoint of transition probability. The forward transition frequency from a non-ECD plasma state to the ECD state was precisely determined under fixed LHCD power. Thus, a statistical probability for ECD transition was determined; that is, the transition behavior around the threshold power could be described in a statistical manner. Transition frequency showed a strong power dependence. Fourth, the current ramp-up scenario without using centre solenoid coils was reinvestigated at higher density, and controllability of the current ramp-up rate was studied. The plasma was initiated by ECH (fundamental O-mode at 170 GHz with 200 kW) at $B = 6.7$ T, and a ramp-up rate below the technical limit of 150 kA/s for ITER could be achieved by choosing LH power. A model to describe the ramp-up is proposed.

Keywords: tokamak, steady state operation, LHCD, transition, current ramp-up, He irradiation

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1. Introduction

Although integrated experiments on issues associated with burning plasma physics and fusion reactor technology in ITER (International Thermonuclear Experimental Reactor) are planned, there are still unresolved issues for the demonstration of steady state operation of a fusion power plant. In ultimate steady state burning plasma, which is assumed to be stable against MHD instabilities, how to control plasma density while keeping the exhaust criterion below a threshold under complicated, temporally varying wall and divertor plate conditions will become a crucial issue.

Concerning these issues, it has been pointed out in ref. [1] that the temporal behavior of wall (Plasma Facing Components) pumping plays an essential role in particle sink or as a source affecting in-vessel global particle balance. The temperature evolution of PFCs and energy deposition on them should be monitored to understand the global particle balance. In order to solve uncontrolled density evolution in a long pulse, in which the particle source from PFCs far from the plasma changes over time, the active cooling of all PFCs has been proposed [2]. This was tested recently in Tore Supra [3-5], and the global density of $1 \times 10^{19}$ m$^{-3}$ could be sustained for 4 minutes. Although the expected saturation time $\tau_{sat}$ for the toroidal pump limiter (TPL) made of carbon head is about 60 sec, the in-vessel gas inventory does not show any sign of saturation for 4 minutes ($\sim 4\tau_{sat}$). Although the temperature is well controlled at 300°C, unexpected higher temperatures were observed on some parts of PFCs, which are considered as new particle sink regions. Thus, even under normal operating conditions in the steady state an essentially difficult problem is that there are no controlling tools to prevent the energetic neutrals from escaping from the plasma region and bombarding the first wall. For off-normal events in tokamaks, such as disruption, ELMs, loss of runaway electrons, and ripple loss for energetic particles, severe damages are expected on very local parts of PPCs and will lead to high temperatures at these sites. Generally, those sites become sources of impurity atoms via evaporation [6] and of new particle sinks through co-deposition processes [7].

Another issue is the radiation effect on PPCs due to bombardment of helium ash escaping from the hot core. Recently, it has been pointed out (see ref. [8]) that the radiation-induced defects on the materials (tungsten and molybdenum) caused by helium atoms become important from the viewpoint of creation of a new particle (hydrogen isotope) sink region. As shown in ref. [9], in a linear divertor plasma simulator device, it should be noted that bombarding helium atoms could create bubbles on the surface of tungsten even if their energy (< 50 eV) was much below the sputtering threshold energy (~ 0.4 keV). It was also found that the bubbles could be formed even at high temperature (~ 1,000 K) [10]. The fact that helium bubbles are remarkably formed at high temperature in tungsten even if the energy of the helium is less than or near the threshold energy for displacement damage suggests the enhancement of hydrogen trapping and difficulty controlling the recycling process for the hydrogen isotope during the burning plasma discharge.

From the plasma physics point of view, on the other hand, in addition to the steady state sustainment of the high performance plasma, new physics study of the plasma, especially, transient phenomena occurring with less probability or relaxation oscillations with very low frequency will be expected in a steady state long pulse discharge, based on the fact that such phenomena are often observed in a cw laboratory plasma [11]. In TRIAM-1M, it was observed that a hot ion temperature mode (HIT) with an ion temperature barrier formed at half radius could be sustained for more than one minute in a low density lower hybrid current drive plasma [12]. Recently, the enhanced current drive mode (ECD) was discovered in the higher density plasma at high power conditions [13-16]. This mode shows a clear $H_\alpha$ drop at the transition and is characterized by simultaneous improvement of the energy confinement $H_{90} \sim 1.4$ and current drive efficiency from $3-4 \times 10^{18}$ A/m$^2$ MW to $6-10 \times 10^{18}$ A/m$^2$ MW. The obtained transition frequency of this mode is about 0.1 Hz near the threshold power, which means that no observation of the transition is expected in a short pulse discharge with duration time of less than 10 sec. This point will be discussed later.

Furthermore, in a future fusion power plant continuously working over several months, the role of inductive effects on the plasma performance will be neglected. In order to establish this full non-inductive scenario, pure rf-driven current start-up experiments are being conducted on TRIAM-1M under ITER relevant conditions ($B \sim 6.7$ T and $n_e \sim 2 \times 10^{19}$ m$^{-3}$) [17,18]. In JT-6U, a plasma is initiated by ECRH, but current start-up is assisted by the inducted electric field by swing the coil current in the outer poloidal coils [19]. The target density is much less than $10^{19}$ m$^{-3}$. Although the toroidal electric effect becomes small in a large toroidal device, the effect on the density peaking is still observed [3]. In addition to the effect on the density...
profile, the current ramp-up process is affected by the induced electric field, which reduces the current drive efficiency. The characteristic time and ramp-up process for current ramp-up in the start-up phase and steady state are examined. From a theoretical point of view, a small electric toroidal field, much below the conventional runway threshold, has been proposed to be sufficient to accelerate electrons via large angle Coulomb scattering [20,21]. This is taken into account in the current ramp-up experiment.

In this paper, the results for the above mentioned investigations on the TRIAM-1M are reviewed [22]. In Sec. 2, the experimental conditions for a long pulse operation are briefly described. The global energy distribution on in-vessel PFCs and global particle balance will be described in Sec. 3. Impurity problems arising from local heat deposition will also be discussed. In Sec. 4, the helium radiation damage effects will be presented for helium discharges. In Sec. 5, results of a statistical investigation of the ECD transition phenomenon and power threshold are shown. The research results of current ramp-up analysis in full current drive plasma are given in Sec. 6. Finally, a summary is given in Sec. 7.

2. Steady State Operation and Related Diagnostics

TRIAM-1M ($R_0 = 0.8$ m, $a \times b = 0.12$ m $\times$ 0.18 m and $B = 8$ T) is a high toroidal magnetic field superconducting tokamak [23]. Lower hybrid waves and electron cyclotron waves are used for current drive and plasma heating. Two kinds of lower hybrid current drive systems have been installed. One consists of a 2.45 GHz 50 kW cw klystron tube, oversized transmission waveguides, phase shifters, and a grill antenna with four waveguides. The launched peak refractive index $N_i$ is ~ 1.6, and $\Delta N_i/N_i$ ~ 1. This system is usually used for long pulse discharge operation. The expected resonant energy is ~ 100 keV. Other systems include two sets of the cw 8.2 GHz system, with power of 200 kW each and an antenna comprised of a grill with $2 \times 8$ waveguides. The peak of $N_i$ can be varied from 1.6 to 2.5 during a single discharge for one system. These systems are located around the torus separated by 180°, but are operated by a single oscillator. A gyrotron (200 kW and five seconds pulse-width at 170 GHz) is used for ECRH and plasma production [24]. The waves in the ordinary mode are injected perpendicular to the toroidal magnetic field.

The main chamber consists of 304SS, and the limiters at three toroidal locations are molybdenum, as are the divertor plates that cover an inner area ~ 1/8 of the chamber surface. Because these molybdenum PFCs interact with plasma strongly, the chamber surface is considered to be the deposition-dominant surface [25] and therefore to play an important role in particle balance. The temperatures of in-vessel PFCs, also a key parameter to understanding the global particle balance, are monitored by thermocouples, and the surface temperature of the limiter that contacts the plasma at the shorter major radius is measured by IRTV. For a low power long pulse discharge, the maximum wall temperature increases up to ~ 120°C with a time constant of 10 to 35 minutes. The temperature of the limiter-contacting part goes up to more than 1,000°C and saturates quickly, within 5 minutes. These different time evolutions of in-vessel PFCs lead to complex global particle balance. In order to understand the energy deposition on PFCs and extracted capability, the calorimetric technique is applied by thermistors. The temperature difference $\Delta T$ of inlet and outlet cooling water is monitored, and the injected energy for the ultra long discharge is determined by $\Delta T$ at the steady state and by the flow rate. The total balance between injected and extracted energy agrees within 5%. Thus the distribution of energy deposition on the limiters, wall, and divertor plates is determined. Energy deposition is distributed in the following fraction: 64% on the vacuum chamber wall, 12% on the divertor plates, and 21% on the limiters [26]. For the short pulse discharges at high power, the time integration method of the temperature rise is used to evaluate energy deposition. The evaluated fraction is similar to that in the steady state measurement.

Neutral pressure is recorded by a B-A gauge at the head of the turbo-molecular pump and is used to evaluate the external pumped particle rate. The Balmar $\alpha$ emission $H_\alpha$ is used in a feedback system for steady state particle control, in which the particle influx is kept constant during the discharge [27,28]. Particle fueling is adjusted by piezoelectric valve in this feedback system, and fueled particles are therefore determined. Toroidal symmetry of the particle influx is monitored at four different toroidal positions, including that near the limiter port. The poloidal distribution of $H_\alpha$ is also measured by a 25-channel array. In order to evaluate the amount of hydrogen retained on the PFCs, specimens mounted on a movable surface probe system are inserted into the shadow of the limiter and exposed for particular discharges. The retention of hydrogen is subsequently
measured by means of thermal desorption spectroscopy TDS.

3. Global In-Vessel Particle Balance and Local Impurity Influx Effects

Two problems concerned with particle control affecting steady state operation have been studied; they are considered to relate to the normal conditions and off-normal conditions, respectively. The first problem involves global particle balance relative to wall pumping, wall saturation, wall fuelling, external pumping, and external fuelling [1,29]. This was investigated in a long duration discharge at a low density and low power. The second one involves enhanced impurity influx from the local parts of PFCs composed of metallic materials (iron Fe, chromium Cr, and nickel Ni of the vacuum chamber, and molybdenum Mo of the limiter and divertor plates), and its effect on the sustainment of high performance of the plasma. This problem has been studied in a high power medium density plasma, focusing on termination of the enhanced current drive (ECD) mode.

3.1 Global particle balance

First, a key phenomenon that should be taken into account for the steady state global particle balance will be discussed. Figure 1 shows the wall pumping rate (A) (defined as external gas fueling rate (solid line)-external pumping rate (B by dotted line)) in an ultra long discharges at low power (< 20 kW) and low density (0.1–0.2 × 10¹⁹ m⁻³). It should be noted that in Fig. 1 (d), the wall pumping rate changes over time; that is, the PFCs alternate between particle sink and source during a single discharge under the fixed particle influx condition. The averaged wall pumping rate was evaluated to be 1.5 × 10¹⁶ atoms/s/m² at t ~ 15 minutes, and total wall inventory was ~ 3 × 10²⁰ hydrogen atoms at 70 minutes. Based on a comparison with the inventory of a plasma at higher density (~ 1 × 10¹⁹ m⁻³) and higher wall pumping rate of 4 × 10¹⁷ atoms/s/m² but short pulse (~ 30 sec), this phenomenon occurs when that the whole surface of the PFCs is not saturated but, rather, the wall is apparently saturated [30]. It has been also pointed out, in ref. [1], that the charge exchange CX neutrals play an important role in wall pumping because of the longer implantation length and re-deposition of impurities by sputtering.

As mentioned earlier, the entire wall surface is coated with molybdenum regardless of substrate materials, and this was experimentally confirmed by spectroscopic measurement. No metal influx (Fe, Ni, Cr) except Mo I and II is observed in low density long duration discharges. Generally, one can neglect the hydrogen retention on the bulk metal (Mo or W) even at a high fluence value of 10²⁷ atoms/m², which is mainly obtained in laboratory experiments [7]. However, in the real circumstance of the plasma-wall interaction, the re-deposition of impurities on specimens exposed to the low density plasma showed a quite different aspect with respect to hydrogen retention [30-33]. The deposition was composed mainly of Mo and O, and contained C and small amounts of Fe, Ni, and Cr. It was observed that the chemical state of deposited Mo was a mixture of metal and oxide, and that the crystal structure of Mo is not bcc but fcc. Thus, recent materials analysis data obtained in the real plasma condition have indicated that
in the presence of oxygen impurities, hydrogen co-deposition in metallic co-deposits can be as significant as in carbon co-deposits, as shown in Fig. 2. It is suggested that the retention of hydrogen is enhanced even if the least amount of deposition is formed on PFCs. Thus, this complex process is considered to be a candidate for use in interpreting the temporary variation in wall pumping rate.

To evaluate the co-deposition quantitatively, a model for the global balance is proposed [34]. This model is a zero-dimension and four-reservoir (core plasma, SOL region, gas phase, wall materials) particle balance model, but the sputtering yield and co-deposition probability is taken into account to interpret the wall pumping effects. The particle flows among reservoirs are hypothesized in this model. Due to charge exchange neutrals with high energies above the threshold energy of 275 eV for molybdenum, the entire in-vessel wall is assumed to be subjected to rather high-yield sputtering. In the present model, the hydrogen co-deposition characteristics are assumed to be such that the co-deposition probability is 0.3. Temporal variation of the recycling coefficient is taken from the measured data. The result of the wall pumping rate is shown in Fig. 3. The obtained time-averaged wall pumping rate calculated from these data is $1.8 \times 10^{16}$ atoms/s/m$^2$, which shows good agreement with our observation of $1.5 \times 10^{16}$ atoms/s/m$^2$. This model predicts that the steady state core plasma density decreases with increasing the co-deposition probability.

The global particle balance is also investigated in a record discharge plasma of 3 hours and 10 minutes, as shown in Fig. 4. In this low $\bar{n}_e \sim 1 \times 10^{19}$ m$^{-3}$ and low power (< 10 kW) discharge, the fuelling was stopped at $t \sim 30$ minutes, after which $\bar{n}_e$ could be sustained without gas fuelling up to the end. The temporal change in wall pumping rate was measured. A wall pumping rate of $2.4 \times 10^{16}$ atoms/m$^2$/s was observed at $t = 10$–20 min, but it became negative ($-8 \times 10^{15}$ atoms/m$^2$/s) for $t > 30$ min. This means the wall acted as a particle fuelling source after the gas puffing was stopped. As a consequence of no fuelling and recycling, dominant H$_\alpha$ increased gradually by a factor of 1.6 with a time constant of ~ 50 minutes up to the end of discharge. Given that the wall temperature increases (up to ~ 120°C) with a longer time constant of 10 min to 35 min, this slower temperature rise might enhance gas release from the wall [1]. Thus, the temperature control of PFCs and the development of tools for controlling the co-deposition process become important to achieve steady state
3.2 Localized impurity influx from a hot spot

A second key phenomenon that should be taken into account for the steady state of high performance plasma will be discussed. This phenomenon, the enhanced impurity influx, is considered to be related to off-normal conditions as a consequence of the localized temperature rise of the PFCs. In this case the loss of ripple trapped energetic electrons is considered to lead a localized thermal load.

This becomes a problem particularly in the higher power (> 70 kW) higher density (> $1 \times 10^{19}$ m$^{-3}$) 8.2 GHz LHCD plasma. Fig. 5(a) shows a typical hot spot on the limiter in the non-ECD 60 sec discharge at ~ 70 kW. It appears at $t \sim 30$ s. In the case of $P_{\text{CD}} > P_{\text{th}}$ when the ECD transition occurs, the hot spot appears at the same position 1–2 sec after the transition, but its brightness is enhanced. It is also found that Fe, Cr, and Ni influx relevant to the hot spot from the topside (electron toroidal drift side) is significantly enhanced when ECD occurs, and oxygen flux does not. In Fig. 5(b), (c) the two chordal profiles are shown, respectively. In the case of non-ECD discharge, this kind of enhanced metal ion influx is not observed, although a similar hot spot is formed. Although the CX energetic neutals play an important role in co-deposition and related hydrogen retention, the evaporation process from the hot spot is considered to be a cause for the localized impurity influx.

The evaporation rate, that is, the metal atom influx, can be evaluated from the relation of the surface temperature $T_s$ to the evaporation enthalpy $\Delta H$ [6] shown in the following equations:

$$\Gamma_{\text{eva}} = 3.5 \times 10^{26} p_0 \exp(-\Delta H/k_B T_s)\sqrt{M T_s} \left[\text{torr K; atoms/m}^2 \text{s}\right].$$  (1)

For a semi-infinite plane, $T_s(t)$ is obtained analytically by

$$T_s(t) = T_s(0) + 2q_0\sqrt{t}/\sqrt{\pi \rho c k_B} \left[\text{sec; K}\right],$$

where $p_0$ is the equilibrium vapor pressure at $T_s$, $k_B$ the Boltzmann constant, $q_0$ the heat flux, $\rho$ the density, and $c$ the specific heat. In this case, the numerical calculation based on a finite volume method was performed to obtain the distribution on surface temperature on part of the vacuum chamber wall. The total evaporation rate $\Gamma_{\text{eva total}}$ [atoms/s] is dominated by the highest $T_s$ region. The results are shown in Fig. 6. The hot spot size and the heat flux are assumed to be 1 cm$^2$ and 5 MW/m$^2$, respectively. These assumptions are based on a change in the heat load measurement (1 ~ 2 kW) on the wall for ECD plasma and the size of the damage on the top of the vacuum chamber. Further, no surface cooling by radiation is assumed. In order to compare $\Gamma_{\text{eva}}(t)$ with the evolution of metal influx, an additional assumption regarding the surface temperatures; which are the same as those at which the hot spot appears, is introduced. Based on observation, the required times for the appearance of the hot spot are 10 sec for 120 kW and 30 sec for 80 kW. Thus, heat flux is determined. Finally, the time evolution of $\Gamma_{\text{eva}}$ integrated over the hot spot is compared with the observed metal impurity evolution, as shown in Fig. 7. Although the surface temperature changes slowly even when the power is turned off, the $\Gamma_{\text{eva}}(t)$ shows quick decay according to the nonlinear temperature dependence of $\Gamma_{\text{eva}}$. During the rising phase, although there is a discrepancy between evolutions, the
waveform of the heat flux and the resulting evolution of the surface temperature may be the cause of the enhanced impurity influx. In order to evaluate the absolute value of $\Gamma_{eva}$, the infrared camera and visible measurement system will be set up at the same measuring port.

In order to investigate how the metal influx affects the lifetime of the discharge, a time $\tau_{IP=90\%}$ for steady state operation of the ECD mode, is introduced, where $\tau_{IP=90\%}$ is defined as the time period for the plasma current to be sustained within 90% of the maximum value. Given that no significant drop in $I_p$ was observed in non-ECD discharges, $\tau_{IP=90\%}$ corresponds to the discharge pulse width. In the ECD discharge, however, $I_p$ is increased at the transition and then sustained for some time period, and finally begins to decay quickly. The metal ion influx from the topside well correlates with this sequence. With increasing power, the metal ion intensity is greatly enhanced, and $I_p$ drops drastically. An example is shown in Fig. 8(a), and in Fig. 8 (b)-(c) these results are summarized. The longest SSO period in ECD mode ($\tau_{IP=90\%} \sim 18$ sec) is achieved at $P_{CD} \sim P_{th}$. With increasing power ($\sim P_{th}$), it becomes difficult to enlarge $\tau_{IP=90\%}$. Further optimization for quenching the hot spot and minimizing the local PWI effects remains to be a subject for SSO of high performance plasma.

4. Helium Irradiation Effects on the Surface Materials

High-energy CX particles bombarding the PFCs cause not only physical sputtering but also damage the inside of the materials due to their high energy [35-37]. The depth distribution of accumulated radiation damage (dislocation loops) shows a significant contribution from the high energy (several keV) CX neutrals. The minimum energy for damage by incident particles with mass $M_1$ to the target atom with mass $M_2$ can be estimated from the following equation:

$$E_{\text{min}} > \frac{\left( M_1 + M_2 \right)^2}{4M_1 M_2} E_d,$$

where $M_1 = 1$ for H or 4 for He, $M_2 = 96$ for Mo, and $E_d = 35$ eV for Mo. $E_{\text{min}}$ is 850 eV for hydrogen atom and 228 eV for helium atom.

In the case of burning plasma, we should take into
account the effects of helium because it is well known that helium atoms have much stronger effects of material damage than do hydrogen atoms. Thus, effects of helium ions and charge-exchanged neutral helium (He$^+$ + He$^0$ = > He$^0$ + He$^+$) in terms of material damage were investigated in helium LHCD plasma at 8.2 GHz [8]. Typical discharge duration was ~ 7 s and the total exposure time was ~ 125 s (~ 18 shots). Experimental conditions were as follows: $n_e \sim 1 \times 10^{19}$ m$^{-3}$, $I_p > 20$ kA, $T_i$ (H) < 0.2 keV, $P_{CD} \sim 150$ kW, and $B = 7$ T. A typical discharge is shown in Fig. 9. Although the working gas is helium, this plasma is not a pure helium plasma; it is contaminated by hydrogen, which is

Fig. 8 (a) The effects of the enhanced metal influx causing to the current drop, (b) plasma current in non-ECD ($P < P_{th}$) and ECD transition plasma ($P > P_{th}$). $P_{th}$ is about 100 kW. The $\tau_{0.9}$ is determined from these series. (c) $\tau_{IP=90\%}$ v.s. Power. $\tau_{IP=90\%}$ decays with $P_{CD}$.

Fig. 9 A typical discharge of He plasma, $I_p$ (a), $n_e$ (b), and charge exchange flux at ~0.3 keV (c). 8.2 GHz LHCD (hatched area) is used at B = 7 T. The power is 150 kW.
trapped in, and released and recycled from the PFCs. $T_i$ is measured from the hydrogen CX energy spectrum. Based on the fact that the temperature equilibrium time is $\sim 0.1$ ms for $\text{He}^{2+}$ and protons under this condition, the ion temperature of He ions is considered to be the same as that of $T_i$ (H). Because that of $T_i$ (He) is lower than those in ref. [37], relatively strong effects of helium, compared with those of hydrogen, are expected.

A water-cooled collector probe system has been installed on the TRIAM-1M, as shown in Fig. 10. Pre-thinned vacuum-annealed molybdenum and tungsten disks of 3 mm$\phi$ were used for the damage analysis and bulk copper plates for the detection of implanted helium. These specimens were attached on the head, and this collector probe system could be inserted into the SOL through a horizontal port situated 11 mm behind the poloidal limiter surface. The toroidal location of the limiter is on the right-hand side with respect to the probe. The temperature of the probe head during discharges was kept at about 24°C by circulating water. The actual temperature of the specimens was expected to be comparable with that of the probe head due to the tight thermal contact. In order to collimate the particle incident directions and to avoid the effects of charged particles, the probe-specimens were placed in deep holes on the plasma facing side (P-side). The holes lead in different directions, from the bottom to the top or from the left to the right of the plasma, at a semiangle of 14 degrees. Some specimens were directly mounted on the surface of the P-side, and some on that of the electron drift side (E-side) of the probe head.

After the discharge experiments, transmission electron microscopy TEM and TDS were performed. First, the microstructure of the pre-thinned molybdenum specimens directly placed on the probe surface at the P-side was studied by means of TEM. Figure 11 shows radiation-induced defects in the molybdenum specimens placed on the surface at the P-side, where bombardment by both CX-neutrals and ions is expected. It is remarkable that a large quantity of dislocation loops (black images in (a)) and very dense fine bubbles of about 1–2 nm in diameter (white image in (b)) were formed by exposure to helium plasma discharges for 125 s. Such heavy damage has not been observed in hydrogen plasma discharge with a duration of more than one hour, as discussed in Sec. 3-1. For Mo specimens placed

Fig. 10 A schematic view of the collector probe system (a) and the probe head (b).

Fig. 11 TEM image of radiation damage in molybdenum placed on the surface at the P-side. (a) Image at small s condition. (b) Image at large s condition.
inside the hole, only dislocation loops were formed; the fluence is much smaller than that for directly placed specimens due to smaller solid angle opened in the plasma.

Second, by TDS (from room temperature to 1,100 K with a ramping rate of 1 K/s) of the P-side bulk copper plates, the amount of retained helium was estimated to be $3.9 \times 10^{20}$ He/m$^2$. Based on the fluence dependence of the total retention of helium, the incident fluence was considered to be $6.1 \times 10^{20}$ He/m$^2$. This means that the average flux of the helium at the plasma facing surface was at least $3.1 \times 10^{18}$ He/m$^2$.s. A clear deposition of Mo on the E-side of bulk copper specimens (< 16 mm from the limiter surface) was observed. TDS results for the specimens on the E-side showed that the retention of He was $> 10^{21}$ He/m$^2$. This value indicates that the He trapping site is formed by Mo deposition. These results indicate that the hydrogen recycling phenomenon during the burning plasma discharge must be quite different from that in the hydrogen plasma discharge experiments. The study of hydrogen recycling in steady state operation under these circumstances is subject of some urgency.

5. Threshold Power for the ECD Transition

Threshold power for the H-mode has been investigated in ITER physics R&D because assessment of the necessary power has been a crucial issue for ITER design study [38]. Two scaling laws are proposed based on the various tokamak databases. On TRIAM-1M, we approached this subject in a different way. Using the advantage of the steady state operation and introducing a transition frequency for a transient event as a function of power, we investigated the threshold power for the ECD mode from a statistical viewpoint.

Thus, the transition behavior of the LHCD plasma into the ECD mode has been intensively studied below and near the threshold power $P_{th}$. The ECD mode is characterized by simultaneous enhancement of energy confinement and current drive efficiency $\eta_{CD} = \eta_{IECD} R_0 / P_{CD}$, and by a clear power threshold for the transition [13-16].

In Fig. 12, time trajectories of a measure of the ion confinement time ($\tau_i^n = n_e T_i / P_{CD}$) and $\eta_{CD}$ are plotted as a function of power. A clear hysteresis curve is seen during the step-like power change cycle. In phase 1, rf power is 100 kW and the non-ECD plasma is sustained for more than 1.5 sec; in phase 2, rf power is raised up to 150 kW and kept constant for 2 sec. The plasma changes to the ECD mode at 0.2 sec after the power is raised. Then, in phase 3 and phase 4, rf power is reduced to the previous level of 100 kW. During phase 3, the plasma varies slowly in time from the ECD to the
non-ECD mode for ~ 0.3 sec, and in phase 4 the transition occurs again, even at the low power of 100 kW. In order to confirm that the power hysteresis is not the ‘apparent hysteresis’ driven by a difference in characteristic time constants between power and plasma parameters, this power hysteresis was checked by varying the power with a changing rate of ±10 kW/s in a single shot. This changing rate means that only 100 W is changed during the energy confinement time scale of 10 ms, and the power fraction δP/P is less than 0.1%. Usually, no effect is considered on the power balance. The power ramp-up and ramp-down results show a clear hysteresis. Thus we conclude that the hysteresis between τ* (ηCD) and power certainly exists, even in this limit.

In the transient phenomena including the LH transition, a similar hysteresis relation has been observed, although both the power changing rate and power changing fraction are higher by an order of magnitude than those in our case [39,40].

Another characteristic is the transition at various power levels; for example, the transition in phase 2 at 150 kW and that in phase 4 at 100 kW in Fig. 12. Recently, transition phenomena and the hysteresis relation were analysed in conjunction with a statistical theory, in which the transition process is described statistically by a transition probability [41,42]. In order to study the statistical nature of the ECD transition at around the threshold power, a measure of the transition frequency, which is expected to be associated with the transition probability, (i.e., a time τtransition is defined as a time difference tECD − trf with which the transition occurs after the power is applied at t = trf, where tECD is the time of ECD onset is introduced. In Fig. 13, the transition frequency (1/τtransition) is plotted as a function of PCD for various discharges, with the longest duration being up to 60 sec. At ~ 70 kW, no transition occurs for discharge with a duration of at least 60 seconds, meaning almost zero transition probability. For non-ECD discharges, only the upper limit of 1/τtransition is given. With increasing PCD > Pth, the transition tends to occur with increasing high probability; that is, larger value of 1/τtransition. The frequency increases logarithmically within the hysteresis power window (90–140 kW), as shown in Fig. 13. The spontaneous transition will occur with the frequency of an inverse energy confinement time of 100–200 Hz. This power level is the minimum power required to trigger the transition with high probability. On the other hand, the absolute power threshold with minimum probability is considered to be ~ 80 kW, at which the transition frequency is ~ 0.1 Hz. At around this power, the transition probability is not only reduced; the plasma also shows oscillatory evolution between non-ECD and ECD states.

6. Non-Inductive Current Ramp-Up and Controllability of Ramp-Up Rate

The non-inductive current ramp-up scenario and controllability of the current ramp-up rate are investigated under the higher density regime. Although so far many current ramp-up experiments by non-inductive methods have been performed [43-46], relatively low density plasma (0.05 ~ 0.5 × 10¹⁹ m⁻³) has been used, as shown in Fig. 14. A recent JT-60U result was obtained at ~ 0.3 × 10¹⁹ m⁻³ [19]. Here the characteristic ramp-up time is derived by fitting Ip (t) by the equation Ip (t) = Ip∞ (1 − exp (−t/(τrise))), where Ip∞ is a steady state value and ⟨τrise⟩ is a global characteristic ramp-up time. For Refs. [19,45,46], the doubling time, for which the initial current becomes double, is used as a characteristic ramp-up time because Ip increases linearly with time. In ITER, an ECRH initiated target plasma with higher density will be expected in the non-inductive ramp-up scenario [38]. We investigated such a situation on TRIAM-1M; as follows. A target plasma is initiated by ECH of 170 GHz at B = 6–6.7 T. Then LHWs are injected to drive the plasma current. The target density ̃n of 1–2 × 10¹⁹ m⁻³ is obtained under the conditions of ECRH 70–150 kW and gas pressure from 1.2 × 10⁻⁶ torr.

![Fig. 13 Transition probability (frequency) is plotted as a function of PCD. Open squares correspond to non-ECD discharge with duration of τtransition. At 70 kW probability is zero for a 60 s discharge with no transition. At 80-90 kW the probability is marginal for ECD.](image-url)
to $2 \times 10^{-5}$ torr. The electron temperature is not measured, but is deduced to be $> 20–50$ eV based on spectroscopic O V (278 nm) measurement. There is no evidence indicating the presence of energetic electrons having more than a few tens of keV, but electrons having at least $\sim 1$ keV are observed by soft X-ray measurement. The resonance position of ECRH was surveyed from a better coupling point of view, and the best condition was obtained at $B_t = 6.7$ T where ECW breakdown occurred 4 cm from the plasma centre of $R_p = 0.84$ m. Given an initial density during the ECH phase of $\sim 1 \times 10^{18}$ cm$^{-3}$ at 7 mm behind the limiter or near the LH antenna mouth, the coupling efficiency is about 90% even in the initial phase.

A typical current ramp-up result is shown in Fig. 15. In these experiments, for $t < 0.1$ s, a pre-programmed vertical field $B_v$ of $\sim 60$ G is applied, and $I_p$ grows with shorter $\tau_{\text{rise}}$ of $\sim 0.1$ s. For $t > 0.1$ s, the plasma position is controlled by a feedback loop, and $I_p$ increases with longer $\langle \tau_{\text{rise}} \rangle$ of $\sim 0.2$ s. In Fig. 16 the current ramp-up rate $I_{\text{rms}}(\tau_{\text{rise}})$ and $\langle \tau_{\text{rise}} \rangle$ are plotted as a function of injected power. Data are taken from current ramp-up experiments with $I_{\text{rms}} > 20$ kA and at least 2 sec duration ($\sim 10$ times $\langle \tau_{\text{rise}} \rangle$), and also include discharges with an ECD transition occurring at $t = 0.5$ sec ($\sim 2 \langle \tau_{\text{rise}} \rangle$). The current ramp-up rate $I_{\text{rms}}(\tau_{\text{rise}})$ seems to increase with increasing $P_{\text{CD}}$, and can be controlled below the technical limit of $\sim 150$ kA/s in ITER when $P_{\text{CD}}$ is below 100 kW. The obtained $\langle \tau_{\text{rise}} \rangle$ is 0.15–0.3 sec. In the non-inductive current ramp-up scenario, we have to obtain the required $I_{\text{rms}}$ and keep the ramp-up rate below the limit because of the AC loss of the superconductors in the poloidal coils. Using current drive efficiency $\eta_{\text{CD}} (= n_eI_{\text{rms}}R_0P_{\text{CD}})$ and the empirical temperature scaling $\eta_{\text{CD}} = \langle \eta \rangle \langle T_e \rangle$ [37], getting the required $I_{\text{rms}}$ means increasing both $P_{\text{CD}}$ and $\langle T_e \rangle$ for fixed $n_e$, $R_0$, and $\langle \eta \rangle$. Here $\langle T_e \rangle$ is the volume-averaged electron temperature. Two independent tools will be required. If $\langle \tau_{\text{rise}} \rangle$ can be given by a so-called L/R (inductance/resistance) time, $P_{\text{CD}}/\sqrt{T_e0}$ should be kept below a certain value for given $\langle T_e \rangle$ and fixed temperature profile ($\langle T_e \rangle$ is replaced by $T_{e0}$ profile factor) in order to control the ramp-up rate below the limit. This means that electron heating is required during the ramp-up phase.

The ramp-up process is investigated in detail. Given that the fundamental problem concerning the spectral gap should be highlighted, especially under the condition of no inductive electric field and cold plasma condition at the beginning of the start-up phase, a ramp-up model with two kinds of characteristic ramp-up time is considered [17]. According to Fisch’s current drive theory [47], the power stored as the poloidal magnetic field can be approximated at the condition of $|E_i| = 0$, $E_i > 0$, as
\[ P_{\text{el}} = V_L \left( I_p - \frac{V_L}{R_{sp}} \right) \]

\[ V_i = \left( n_e e^3 \log \Lambda / 4 \pi e^3 \right) \left( \frac{V_{\text{ph}}}{v_r} \right)^2 \]

(1)

where \( V_L \) is the loop voltage on the last closed flux surface, \( R_{sp} \) the Spitzer resistivity, \( \xi \) the absorption efficiency, \( v_{ph} \) (= light velocity/parallel refractive index \( N_i \)) the phase velocity of the excited LHW in the plasma, \( v_r \) the runaway velocity, and \( E_i \) the electric field in the toroidal direction. Using a relation of \( V_L = E_i / 2\pi R \) and \( V_i = -L \partial I_p / \partial t \), the following differential equation for \( I_p \) is derived:

\[ \frac{\partial I_p}{\partial t} = -\frac{R_{sp}}{L} \left( I_p - I_M \right) \]

(2)

where \( I_M = P_{\text{CD}} \xi (V_{\text{ph}})^2 2e_0^2 m_e / R_{sp} n_e e^3 \log \Lambda \) and \( L \) is the inductance. If \( I_M \) is constant, the solution of this equation, \( I_p(t) = I_M / \tau_{\text{L/R}} \left[ 1 - \exp \left( -t / \tau_{\text{L/R}} \right) \right] \), is obtained, and can be used to evaluate \( \langle \tau_{\text{rise}} \rangle \). However, there is still a discrepancy, \( \langle \tau_{\text{rise}} \rangle \) is longer than \( \tau_{\text{L/R}} \). Here, we assumed that \( I_M \) varies as \( I_M = I_M(1 - \exp \left( -t / \tau \right)) \) [17]. Namely, it is assumed that \( \xi \) or \( V_{\text{ph}} \) increases exponentially in time, and that \( \tau \) is introduced as a characteristics time which is different from \( \tau_{\text{L/R}} \). The solution is described in terms of two characteristic times, as follows:

\[ I_p(t) = I_M \left\{ \frac{\tau_{\text{L/R}} \left[ 1 - \exp \left( -t / \tau_{\text{L/R}} \right) \right] - \tau \left[ 1 - \exp \left( -t / \tau \right) \right]}{\tau_{\text{L/R}} - \tau} \right\} \]

(3)

Thus, the tangent-hypobaric evolution of ramping-up current can be reproduced, for example, by \( \tau_{\text{L/R}} \) and \( \tau \) of 20 ms and 100 ms, respectively. Because there is a very large spectrum gap between the injected wave phase velocity and the thermal velocity of electrons, up-shift of the injected \( N_i \) is required, especially in the early phase. The physics for time-varying absorption efficiency and time-varying up-shift physics are subjects for future study.

**Summary**

The experimental results associated with physics and technology issues of TRIAM-1M are reviewed. Global particle balance including wall pumping effects is investigated in long duration discharges. Based on observation of specimens exposed to long duration plasma, it is proposed that the co-deposition of Mo + O affects hydrogen retention. A model taking this effect into account is proposed, and showed good agreement between calculated and experimental values. In the high power and high density experiments, it is pointed out that the local PWI strongly affects the lifetime of the high performance plasma. In particular, it is found that power-dependent local metal impurity influx from the hot spot terminates the high performance ECD plasma.

In order to evaluate the effects of helium bombarding on PFCs and hydrogen retention in burning plasma, microscopic damage of metals and He retention for specimens exposed to helium discharge of \( \sim 125 \) s are studied. A large quantity of dislocation loops and dense fine bubbles are observed by means of TEM. Based on TDS for the specimens, the amount of retained

![Fig. 16 Current ramp-up time (a) and ramp-up rate (b) are plotted as a function of P_cd.](image-url)
helium is evaluated to be $3.9 \times 10^{20}$ He/m$^2$.

The statistical nature of the ECD mode transition is studied around the threshold power level. The transition frequency for the ECD transition is introduced, and is determined as a function of $P_{CD}$. For non-ECD discharges, only the upper limit of $1/\tau_{transition}$ is given. With increasing $P_{CD} > P_{th}$, the transition tends to occur with increasing high probability; that is, with larger value of $1/\tau_{transition}$. The frequency increases log arithmically within the hysteresis power window (90–140 kW).

Experimental results of non-inductive current ramp-up and controllability of the ramp-up rate are studied by combination of ECRH + LHCD. The ramp-up rate below the technical limit of 150 kA/s for ITER can be achieved by choosing LH power. A model is proposed to interpret current evolution during a ramp-up phase having two characteristic time constants.

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