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PSI issues toward steady state plasmas in the HT-7 tokamak

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Abstract

The main efforts of HT-7 superconducting tokamak are directed to quasi-steady state discharges and relevant physics. New doped graphite with an SiC gradient coating as a limiter material and ferritic steel used to reduce the ripples has been developed. Significant progress in obtaining high-performance discharges under quasi-steady state in HT-7 has been realized. The long pulse discharges with $T_e \sim 1$ keV and central density $\sim 1 \times 10^{19}$ m^{-3} have been obtained with a duration up to 20 s. Recycling has been studied in LHCD plasmas. The characteristic times of the recycling coefficients seem to be correlated with the physical absorption and adsorption. The uncontrolled density increase is accompanied by the impurity influx originated mainly from the parts of the inner vessel, located far from the plasma edge, which are caused mainly by slow heating by the radiated power and/or fast ion loss.

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Keywords: Plasma–surface interaction; Doped graphite; Ferritic steel; Long pulse discharge; Recycling; Low- Z impurity

1. Introduction

Steady-state operation of tokamak plasma is one of the basic requirements for a fusion reactor. The problems involved are non-inductive current drive, plasma control, heat exhaust, particle removal, etc. On the other hand, high performance, such as advanced tokamak operation modes, is needed for the economic use of fusion reactors. Investigations carried out in HT-7 tokamak experiments are contributing to these fusion reactor relevant issues and underlying physics. HT-7 is a medium sized tokamak with superconducting toroidal coils. Its main purpose is to explore high-performance plasma operation under steady-state conditions [1]. Wall conditioning, plasma facing materials (PFMs) and edge behavior of the plasma are very important issues to get high-plasma performance under steady-state conditions. A new technique based on RF plasma has been developed in the HT-7 tokamak [2] for wall conditioning by injecting waves in the ion cyclotron resonant frequency range. The impact of this technique on various issues and plasma performance has been reported [3].

For the long pulse discharge, the problem of heat and particle exhaust is the most serious constraint. It can be partially solved by special magnetic configuration, in parallel with the development of high-performance plasma facing components (PFCs). On the other hand, global particle recycling at all the internal elements in the vacuum vessel could prevent the high-performance long pulse discharges as observed in an one minute discharge of JET [4]. In this paper, we report the recent efforts and progress in developing high-performance PFCs and PSI studies for long pulse discharges in the HT-7 tokamak.

2. Development of the plasma facing components

Severe requirements on the main limiter materials are imposed for the long pulse discharges with LHCD. Carbon based materials are considered as attractive

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candidates for application in experimental fusion devices. A series of carbon/ceramic composites including B-, Si-, and Ti-doped graphite have been recently developed as low-Z PFMs for reducing the chemical sputtering (CS) and suppressing the radiation enhanced sublimation (RES). By optimizing the proportion of doped elements and the preparation technique, doped graphite by the name of GBST1308 (1%B, 2.5%Si, 7.5%Ti) with high-thermal conductivity up to 180 W/m K, has been successfully developed as the main limiter materials for HT-7 superconducting tokamak. The overall performance evaluation shows that the new type material has good thermal shock resistance, which can withstand 3 MW/m^2 high-heat loads for 60 s. The erosion experiment indicates that CS yield of these carbon/ceramic composites at 50 eV and 1 keV D^+ bombardment was decreased by a factor of 30% and 5%, respectively, in comparison with that of pure graphite. The total outgassing rate is 3×10^{-12} Torr L/s cm^2 at room temperature, which is close to the level of IG-430U isotropic graphite. Good vacuum engineering is favorite for reducing the recycling and density control. The thermal and mechanical properties are very stable when the surface temperature is not more than 1000 °C. Detailed description of the material development and its properties is given in the other paper elsewhere [5]. The overall performance evaluation of the GBST1308 doped graphite is compared with other graphite in Table 1.

As for the surface characteristics of the graphite, however, it is necessary to reduce the impurity generation due to its high-sputtering yield, the density build-up due to its large retention of hydrogen isotopes, and so on. Especially, the limited oxygen gettering capability of graphite allows considerable contamination of the core plasma by oxygen influx. Many efforts have been made to improve the surface characteristics of graphite by use of boron or silicon containing materials, which have a strong affinity for oxygen. Contrary to pure carbon materials, SiC has much lower chemical and high-temperature sputtering, is capable of oxygen gettering and lower hydrogen recycling. The SiC coatings for the limiter tile surface should endure the high-heat flux without large erosion or damage, and exhibit superior surface characteristics for a sufficiently long life. A thick SiC coated graphite is a good siliconized material because it does not have a deleterious effect on the good thermal characteristics of the graphite [6]. However, exfoliation of the coatings should be prevented. A new SiC coating technique, chemical vapor reaction (CVR) combined with chemical vapor infiltration (CVI), has been developed for application to the HT-7 limiter plate, which provides a gradient SiC coatings by the infiltration of reaction gas through open pores. By this method, all carbon titles were coated an SiC gradient coating greater than 200 μm without significant reduction of the thermal conductivity [6]. The SiC coated graphite was used as the limiter and the Faraday screen of the ICRF antenna material in the HT-7 tokamak.

To meet the requirements of the HT-7 device for long pulse discharges, the previous small molybdenum limiter has been replaced by a larger and full poloidal circular graphite limiter with water cooling in 2000. Its total surface area is about 0.3 m^2. Heat load flux is about 2 MW/m^2 for a medium heating power in the HT-7 tokamak (500 kW LHCD; 300 kW ICRF; 200 kW OH). Plasma performance has been significantly improved after installing the new graphite limiter in the HT-7 tokamak. In the previous experiments with a molybdenum limiter, significant hard X-ray emission and a large influx of metal impurity from the limiter surface can be observed due to bombardment of highly energetic electrons produced by LHCD. It often caused the discharge termination. The problems of strong hard X-ray emission from the limiter and metal impurities in the plasma with high-power LHCD have been alleviated after installing the new graphite limiter. Higher plasma parameters have been achieved, while a similar discharge with the previous Mo limiter was very difficult to obtain. The edge recycling, electron density control and impurities can be easily handled. All these results show that the new graphite with SiC coating as the limiter material is satisfactory for long pulse plasma discharges in the HT-7 tokamak. The ripple loss is a problem left to be solved for the high-performance long pulse discharges

Table 1
The main parameters of GBST1308 and its comparable materials

ITEM	IG-430U	CX-2002U	GBST1308	PG
Bulk density (g/cm^3)	1.82	1.65	2.02	1.83
Electrical resistivity (μΩ m)	9.2	2.7/3.4/5.1 (x, y, z)	6.4	12.4
Flexual strength (MPa)	53.9	47/43/17 (x, y, z)	56	52
Compressive strength (MPa)	90.2	48/45/52 (x, y, z)	94	89
Tensile strength (MPa)	37.2	35/30/11 (x, y, z)	47	36
Elastic modulus (GPa)	10.8	11/8.1/3.4 (x, y, z)	11.5	14
CTE for 350–450 °C (1×10^{-6} K^{-1})	4.8	1.7/2.3/5.3 (x, y, z)	4.2	5.1
Thermal conductivity (W/m K)	139	390/320/190 (x, y, z)	196	120
Total ashe content (ppm)	<5	<10	–	<50

with a high power of LHCD and ICRH. The HT-7 has larger ripple at the low-field side due to the engineering limitation of the superconducting toroidal coils. The ripple at the limiter radius of 27 cm is about 4.2%, which can cause significant loss of the energetic ions produced by LHCD and ICRH. The ripple trapped fast ions can interact with the vessel wall causing overheating of the internal elements and significant outgassing. The ripple can be reduced by installing ferritic steel at the locations of the toroidal coils [7]. After a series of tests, we developed a ferritic steel material named GYJ060, which the ingredients ($8\%\mathrm{Cr} + 2\%\mathrm{W} + 0.2\%\mathrm{V} + 0.06\mathrm{Ta} + \mathrm{Fe}$ remainder) that are similar to that used in the JFT-2M [7]. Testing shows reasonable ferromagnetic properties (not saturated up to 2 T) for the HT-7 toroidal magnetic configuration and acceptable characteristics for high vacuum (total outgassing rate $< 18.63 \times 10^{-6}$ Pa L/s at 200 °C), mechanical and thermal resistance. The ferritic steels (total 24 boards, each size 600 mm $\times$ 150 mm $\times$ 18 mm) were installed in 2001. Each ferritic steel board was installed at the locations corresponding to the toroidal coils in the vacuum chamber. They are isolated from the vacuum vessel at one end and covered by the stainless steel liner with baking wires. The ripple at the limiter radius is reduced from 4.2% to less than 1.6% at $B_t = 2$ T at the limiter radius of 27 cm according to calculations. Although there was no direct measurement of energetic particle loss due to the ripple, experimental results in LHCD and ICRH plasmas show better plasma confinement compared with previous discharges without installation of ferritic steels, but with the same poloidal graphite limiter. This fact means that the ripple loss problem is alleviated. Detailed engineering and experiments will be given elsewhere [8].

To increase the neutral removing capability from the edge, particularly from the area near the graphite limiter, two powerful cryogenic pumps (total $> 2 \times 10^4$ Torr L/s for H/D) were installed at the port of the limiter and near the toroidally opposite port in the HT-7 tokamak. Another important issue for high-performance steady state plasmas is the edge behavior. For this issue, new diagnostics including fast reciprocating Langmuir probes and some optical monitors, which are focused at the limiter surface, are developed. After the efforts mentioned above, significant progress has been achieved in high-performance long pulse plasma discharges. The boundary behavior under different wall conditions in the HT-7 tokamak was investigated using Langmuir probes. After installation of the SiC coated graphite limiter and ferritic steel boards, the edge electron temperature $T_e(a)$ was increased typically from $\sim$25 to $\sim$42 eV, density $n_e(a)$ from $\sim 1.2 \times 10^{18}$ to $\sim 1.8 \times 10^{18}$ m^{-3} for plasmas with $I_p = 140$ kA, $B_t = 2$ T and central line averaged density 1.5×10^{19} m^{-3}. Their decay lengths were decreased, while relative fluctuation levels of electron density was suppressed dramatically in the SOL [9]. These characters in the boundary plasma are favored for good confinement. Investigations indicate that the turbulent transport in the boundary plasma is correlated with the impurity influx, impurity radiation and the atomic physical processes, which are improved by efforts in the PSI field.

3. High-performance long pulse discharge

Long pulse discharges were performed, using a double control: plasma current and position were controlled by the ohmic poloidal system, and central line averaged electron density was controlled by feedback control of deuterium gas injection using a pulsed piezo-electric valve. The long pulse discharges with $T_e \sim 1$ keV and central density $\sim 1 \times 10^{19}$ m^{-3} have been obtained up to 20 s. Such discharges can qualify the new PFCs, power and particle injection and exhaust capabilities, diagnostics and feedback control loops, etc. The first investigation on long pulse discharge is to check particle removal capability. It addresses the feasibility of high-performance discharges under steady-state condition. Fig. 1 shows the main characteristics of an LHCD discharge, where $B_t = 2$ T, $I_p = 100$ kA, $n_e(0) = 1.6 \times 10^{19}$ m^{-3} $P_{\mathrm{LHCD}} = 180$ kW, $n_{\parallel} = 2.5$. The discharge duration of 5 s is about 400 energy confinement times and 2 current diffusion times, which means the plasma in quasi-steady-state condition.

The plasma current and electron density were pre-set and feedback controlled. Fig. 1(c) is the time evolution of the total amount (time integration) of gas supply ΣS_g. After a stable feedback control of the density for 4 s, an uncontrollable density increase collapses the plasma, although the feedback controlled gas injection stops as shown by a zero gradient of ΣS_g in Fig. 1(c). The density rise is correlated with an increase in both carbon (Fig. 1(g), solid line) and oxygen (Fig. 1(f)) radiation, but carbon radiation is dominant as shown in Fig. 1(g), because the slow characteristic time in the recycling coefficient is nearly the time scale of carbon radiation (see Section 4). The reason is that carborane ($C_2B_{10}H_{12}$) was used for boronization. The boronized wall surface contains a large amount of carbon, which was involved in the global recycling. This phenomenon is observed for all the long pulse discharges whatever the density and current are. It occurs earlier for higher $I_p n_e$ discharges. The experimental results show that LHCD power needed to sustain the plasma current is increased with the increase in the product of $I_p n_e$. This means, therefore, that the uncontrollable density increase is also directly correlated with the injected LHCD power. In a long pulse plasma of Tore Supra, the density became uncontrollable, that is considered to be caused by the increase in outgassing from internal elements which are located far from the plasma edge and slowly heated by

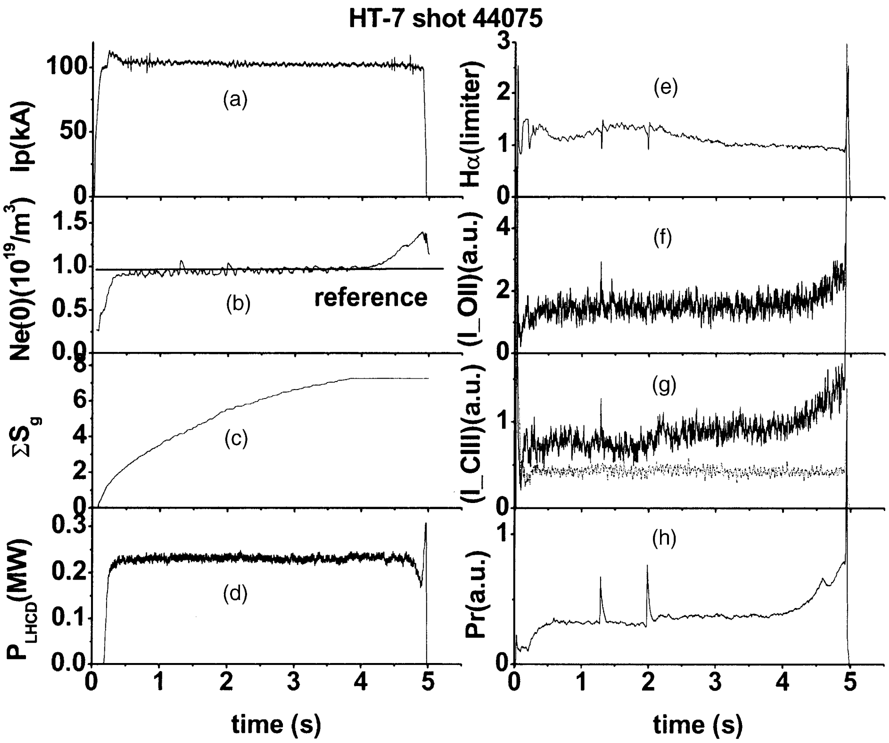

Fig. 1. Long pulse plasma discharge (shot 44075) for 5 s. (a) Plasma current, (b) central line averaged plasma density, (c) total amount of gas supply, (d) LHW power, (e) Ha emission from the limiter surface, (f) oxygen radiation from the inner vessel wall, (g) carbon radiation from the inner vessel wall (solid line) and the limiter surface (dotted line), (h) radiation power.

the radiated power [10]. The radiation of both H_α/D_α (Fig. 1(e)) and impurities (as shown by dotted line in Fig. 1(g) for CIII) observed at the limiter surface did not increase during the density rise. This may be due to the strong active cooling of the limiter, which prevents the outgassing from the limiter surface. Another reason may be due to the SiC coating of the graphite limiter, which has low-chemical and high-temperature sputtering and is capable of lower hydrogen recycling. However, the large influx of carbon and oxygen from the vessel wall causes the radiated power to increase exponentially before the density become uncontrollable. The experimental results show that the radiated power and LHCD power needed to sustain plasma current are increased with an increase in the product of $I_p n_e$. Another probable mechanism of the outgassing from the internal elements is the interaction between the suprathermal ions and the vessel wall as observed in Tore Supra [10]. The suprathermal ions are trapped in the ripple of toroidal field, which leads to localized heating of a small part of the inner vessel. Although there is not direct measurement of fast particle loss due to ripple in the HT-7 tokamak, the impurity radiation levels are decreased after installation of the ferritic steel boards, compared with previous experiments with the same poloidal graphite limiter. The reduced level of the impurity influx, particularly, influx of metal impurities, is at least partially contributed to by the reduced ripples, thus, reduced fast ion loss.

Fully non-inductive long duration discharges were performed up to 20 s with $I_p = 50$–100 kA and central density $\sim 1 \times 10^{19}$ m^{-3}. Fig. 2 shows the waveforms of such discharges. The lower $I_p n_e$, hence, lower LHCD power can prevent an earlier occurrence of the uncontrollable density rise and extend the discharge duration. Even in such long duration discharges, the major limitation of the density increase is clearly correlated with the radiated power and impurity influx, but without significant outgassing from the limiter surface. As shown in Fig. 2, in the shot 47694 the radiated power and impurity influx from the vessel wall increase dramatically before the density become uncontrollable, while the level of H_α emission and impurity influx (dotted line for CIII emission) from the limiter surface does not vary much. The density is well controlled to the pre-set level until discharge is terminated by pre-programming to prevent disruption in shot 47693. In this shot, stable low-radiated power was kept for the whole discharge duration.

The relation between $I_p n_e$ and T_N is summarized in Fig. 3, where T_N is the time, at which the density become uncontrollable. Shots chosen in Fig. 3 are from such plasmas, whose durations are only limited by uncontrollable density rise. T_N is decreased with the increase in

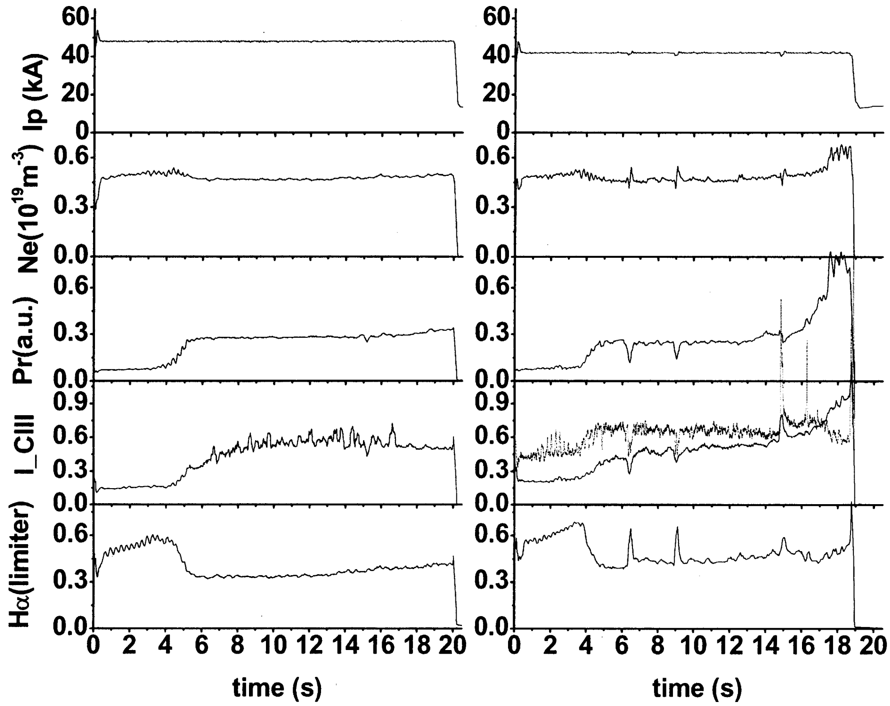

Fig. 2. Long pulse plasma discharge (left shot 47693 and right shot 47690) for ~20 s. From top to bottom: plasma current, central line averaged plasma density, radiation power, carbon radiation from the inner vessel wall and the limiter surface (left, dotted line), Ha emission from the limiter surface.

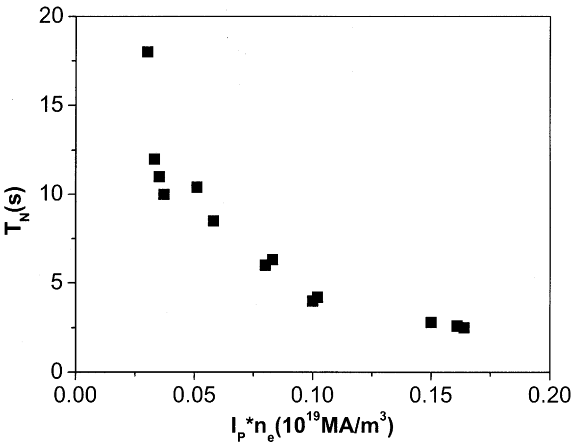

Fig. 3. Relation of T_N versus $I_p n_e$, where T_N is the time, at which the density become uncontrollable.

the product of $I_p n_e$, i.e. the radiated power. Clear correlation addresses the importance of the outgassing from the vessel wall through radiation and/or ripple loss in the particle control for steady-state plasma discharges. The impurity influxes (C and O), particularly, carbon, played a dominant role in density controllability as discussion above. The reason is that the boronized wall surface contains carbon, but suppresses oxygen. Therefore, carbon could be involved in the recycling in a steady-state discharge and affect the density controllability predominantly. Another possible process for density controllability is the particle emission from the LHCD and ICRF launcher. There is no direct monitoring at the launcher position. Coupling of low-hybrid wave into the plasma was quite stable in most of the long pulse discharges if the launched power does not exceed 200 kW. This is the case in the HT-7 tokamak for a steady-state plasma investigation. The outgassing from the limiter surface does not play a dominant role because of the water cooling and SiC coating in the present experiments as discussion above.

Significant progress in obtaining high-performance discharges under quasi-steady state in the HT-7 superconducting tokamak has been achieved. In relation to previous experiments, the features of IBW and LHCD are integrated to obtain long pulse high-performance discharges. Strong synergy effect with higher current drive efficiency was observed if the resonant layer of IBW is located in the LHCD power deposition region. By proper optimization of the operation, particularly, choosing a strategy to avoid MHD activities, high-performance discharges under quasi-steady state in the HT-7 tokamak have been realized under a very good wall condition with very low-impurity content ($Z_{eff} <$ 1.5) and recycling, which is realized by intensive RF

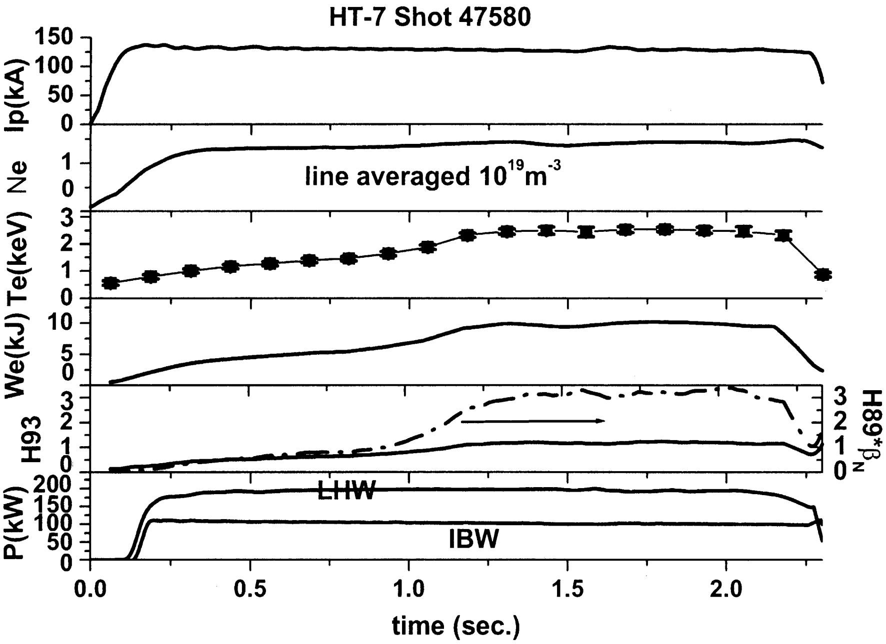

Fig. 4. A high-performance plasma discharge. The normalized performance indicated by the product $\beta_N H_{89} > 3$ is achieved for $> 50\tau_E$.

boronization [2]. Similar performance was not achieved before installation of new poloidal graphite limiter and ferritic steels plates although RF boronization was also applied in the previous experiment. Very long duration discharges up to 10 s (where $B_t = 2$ T, $I_p = 100$ kA, $n_e(0) = 1.0 \times 10^{19}$ m^{-3} $P_{LHCD} = 160$ kW, $n_{\|} = 2.5$) with the normalized performance H_{89} of around unity have been demonstrated, which are in the time scale of $> 800\tau_E$ and >4 current diffusive times. HT-7 has produced a variety of discharges with the normalized performance $\beta_N H_{89} > 1 \sim 4$ for a duration of several to several tens energy confinement times with non-inductive driven current of 50–80%. The normalized performance indicated by the product $\beta_N H_{89} > 3$ is achieved for $> 50\tau_E$ as shown in Fig. 4. The duration at the normalized performance of $H_{89} > 1.5$ with β_N close to unity has been extended to $> 130\tau_E$. The discharges at constant $H_{93} \sim 1.5$ with an ELMy-free edge have been sustained for $> 50\tau_E$ in high-power heated plasma, where electron temperature is about 3 keV. The reproducible long pulse discharge with $T_e \sim 1$ keV and central density $\sim 1 \times 10^{19}$ m^{-3} can be obtained easily with a duration of 10–20 s, which is within the time scale of several current diffusion times.

4. Wall recycling

As discussed above, the particle control is a serious constraint for the long pulse discharges. There is no sign of change in hydrogen/deuterium recycling up to 20 s based on H_α/D_α measurements at the limiter. If the global recycling coefficient increases to and above unity, the plasma density becomes uncontrollable. Properties of wall recycling have been investigated in LHCD discharges. The global recycling coefficient is obtained using the particle balance equation as definition in TRIAM-1M [11], where the density change due to low-Z gaseous impurities is included. The plasma density was controlled to be kept constant by feedback adjusting of the gas feed rate by the pulsed piezo-electric valve. Particle confinement time is calculated from total electron amount measured by a multi-channel interferometer and surface integrated particle flux measured by poloidal and toroidal H_α arrays.

Fig. 5 shows the time evolutions of recycling coefficients in two plasmas with different electron densities. The recycling increases with time. Two characteristic times can be distinguished from the time evolution of the recycling coefficient. It is fitted by two exponential decay functions, which determine two characteristic times. The values of the fast one are 0.688 and 1.1 s and those of the slow one 2.4 and 8.3 s for plasmas with central line averaged densities of 1×10^{19} m^{-3} (shot 44075) and 0.6×10^{19} m^{-3} (shot 47428) respectively. The recycling coefficient varies with two characteristic times. Both time values are increased by decreasing the product of $I_p n_e$ or radiated power as shown in Fig. 6. The recycling coefficient for shot 44075 shown in Fig. 5 increased above unity at 4 s, correspondingly, the density became

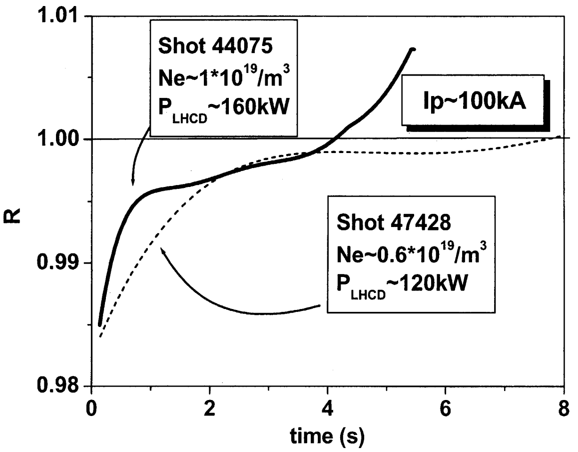

Fig. 5. Time evolution of recycling coefficients for two discharges at $I_p = 100$ kA with higher density (shot 44075) and lower density (shot 47428). Two characteristic times can be distinguished.

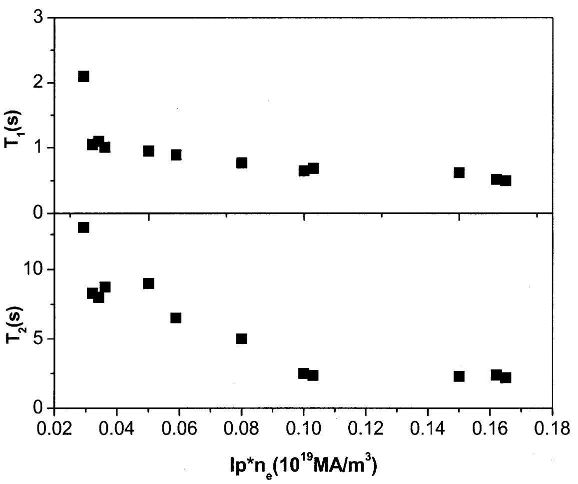

Fig. 6. The values of two characteristic times of the recycling coefficient are increased by decreasing the product of $I_p n_e$.

uncontrollable. Density behavior seems to be determined by release of low-Z impurity due to outgassing from the in-vessel components, since the results in Fig. 5 do not reflect exact behavior of the hydrogen/deuterium recycling as shown in Fig. 1. The slow characteristic time in the recycling coefficient is nearly the time scale of the carbon radiation rise as shown in Fig. 1(g). Other discharges show similar behavior, which indicates a predominant role for carbon impurity from the vessel wall in recycling processes.

The mechanisms of the wall recycling are mainly physical adsorption, chemical adsorption and absorption. The chemical adsorption does not contribute much in recycling because of little increase in oxygen radiation. The physical adsorption and absorption of charge exchange neutrals may be a possible contribution in the long term recycling processes as observed in TRIAM-1M [11]. There are not quantitative results in HT-7 tokamak to confirm this statement. The most probable process is outgassing from the internal elements via radiation heating and interaction with trapped fast ions in the ripple. The evidence can be found in Fig. 1. The gas supply shown in Fig. 1(c) was stopped at about 3.7 s. However, the electron density kept a slow rise, during which the carbon radiation increased as shown also in Fig. 1(g). The outgassing from the internal vessel elements should provide the particle source to keep density rise.

5. Summary and conclusion

The new GBST1308 (1%B, 2.5%Si, 7.5%Ti) doped graphite material was developed as low-Z PFM for reducing the CS and suppressing the RES. This material has high-thermal conductivity and good thermal shock resistance. By the method of CVR, all carbon titles were coated with an SiC gradient coating greater than 200 μm to reduce the high-sputtering yield and large retention of hydrogen isotopes, and endure the high-heat flux without large erosion. It was used as a material of the limiter and the Faraday screen of the ICRF antenna. After installing the new water-cooled graphite limiter with two powerful cryogenic pumps, the edge recycling, electron density control and impurity influx can be easily handled. All results show that the new water-cooled graphite with SiC coating as the limiter material are satisfactory for long pulse plasma discharges in the HT-7 tokamak. After a series of tests, we developed a ferritic steel material. Testing shows reasonable ferromagnetic properties for the HT-7 toroidal magnetic configuration and acceptable characteristics for high vacuum, mechanical and thermal resistance. After installation of the ferritic steels, the ripple is reduced from 4% to about 1.6% at the limiter radius. The experimental results in LHCD and ICRH plasmas show good evidences that the ripple loss problem is alleviated.

After the efforts mentioned above, significant progress has been achieved in high-performance long pulse plasma discharges. HT-7 has produced a variety of discharges with $\beta_N H_{89} > 1 \sim 4$ for durations of several to several tens of energy confinement times with a non-inductive driven current of 50–80%. The reproducible long pulse discharge with $T_e \sim 1$ keV and a central density $\sim 1 \times 10^{19}$ m^{-3} can been obtained with a duration of up to 20 s. The duration of long pulse discharges is mainly limited by uncontrolled density rise, which is strongly correlated with radiated power and impurity influx from the internal elements located far from the plasma edge. Recycling studies confirm the important effect of radiation heating and/or interaction with trapped fast ions in the ripple on outgassing from the internal components. The results described in this paper suggest a necessary development of a complete particle control system for further progress in high-performance discharges under steady-state condition.

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