Prospect Toward Steady-State Helical Fusion Reactor Based on Progress of LHD Project Entering the Deuterium Experiment Phase

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***Abstract*— Large Helical Device (LHD) is one of the world largest superconducting fusion experiment devices, having demonstrated its inherent advantage for steady-state operation since the start of experiments in 1998. LHD has also demonstrated reliable operation of the large-scale superconducting magnet system for almost two decades. Development of the challenging heating systems, such as negative-ion-based neutral beam injection (NBI), high-power and high-frequency electron cyclotron heating, and steady-state ion cyclotron heating, have led to wide-ranging physics and engineering achievements. LHD has progressed to the next stage, that is, the deuterium experiment starting in March 2017, which should further extend plasma parameters toward reactor-relevant regime. For establishing firm basis for designing steady-state helical fusion reactor, advanced physics research, such as on isotope effect, energetic particle confinement, and plasma–wall interaction, will be intensively performed in the deuterium experiments. In an engineering aspect, the upgrade of NBI system has been carried out in preparation to the deuterium experiment, and it should contribute to future NBI development for fusion reactors including ITER. For enhancement of the particle control, the closed divertor system has been installed with pumping capability. Diagnostics for neutron measurements are newly developed and installed for the deuterium experiment. Aligned with all the progress of LHD project in terms of engineering and physics aspects, the conceptual design activity of the LHD-type helical fusion reactor, FFHR-d1, has been programmatically conducted. In parallel to the design study, engineering research and development for the component development have been performed, including those based on employing challenging ideas such as high-temperature superconductor, liquid metal ergodic divertor, and molten-salt breeder blanket. The present status of LHD project entering the deuterium experiment phase is overviewed with putting emphasis on the engineering aspects, and then the engineering research and development activities toward steady-state helical fusion reactor are described.**

***Index Terms*— Challenging engineering ideas, deuterium experiment, FFHR-d1, high-performance plasmas, Large Helical Device (LHD), plasma heating devices, steady-state operation, superconducting magnets.**

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I. Introduction

The Large Helical Device (LHD) is one of the world largest superconducting helical devices. The LHD is an optimized heliotron with simply and continuously wound helical coils, characterized by currentless, i.e., disruption-free plasmas with inherent capability of steady-state operation [1].

The LHD has been operated since 1998, without any severe cryogenic troubles. LHD is equipped with three kinds of heating systems, neutral beam injection (NBI) system [2] currently with three negative-ion-based injectors and two positive-ion-based injectors, electron cyclotron heating (ECH) system [3] with powerful gyrotrons, and ion cyclotron heating (ICH) system [4] mainly utilized for long-pulse discharge. ICH is currently removed for smooth initiation of the deuterium experiment. These heating systems have much contributed to wide-ranging parameter extension and physics experiments.

The LHD, as the one of helical experiment devices, has played complementary and alternative roles to tokamak approach, and made progress on comprehend physics understandings of toroidal plasmas and on establishing scientific basis for a helical fusion reactor.

II. Brief Overview of Achievements in LHD (Hydrogen Plasma Experiment Phase)

Here, in this section, achievements in LHD during the previous hydrogen plasma experiment phase are briefly overviewed. High-performance and steady-state plasmas are required to realize fusion reactor. LHD has pursued this based on Japan-originated heliotron concept, with inherent advantage for steady-state operation.

LHD has demonstrated almost 48-min-long discharge with a few kiloelectronvolt range with a megawatt-class heating, achieving the world record of total injected energy, 3.36 GJ [5]. LHD has explored a long-pulse operation regime toward a reactor-relevant regime while the improvement of the plasma performance (such as measured by the fusion triple product) is required. On the other hand, tokamaks should achieve steady-state operation keeping its high plasma performance. In such a way, synergetic progress on high-performance and steady-state operation beyond the confinement concept is quite important.

TABLE I
PLASMA PARAMETERS ACHIEVED IN LHD (HYDROGEN PLASMA EXPERIMENT PHASE)

Parameters	Achieved (H phase)	Key physics	Target
T_i	8.1 keV ($n_e = 1\times10^{19}$ m^{-3})	Ion ITB Impurity hole	10 keV ($n_e = 2\times10^{19}$m^{-3})
T_e	20 keV (2×10^{18} m^{-3}) 10 keV (1.6×10^{19}m^{-3})	Electron ITB	10 keV (2×10^{19}m^{-3})
Density	1.2×10^{21}m^{-3} (T_e = 0.25 keV)	Super dense core	4×10^{20} m^{-3} (T_e = 1.3 keV)
β	5.1 % (B_T = 0.425 T) 4.1 % (1 T)	MHD in current-free plasmas	5 % (B_T = 1 - 2 T)
Steady-state operation	54min. 28sec (0.5MW, 1keV, 4×10^{18}m^{-3}) 47min. 39sec. (1.2MW, 2keV, 1×10^{19}m^{-3})	Dynamic wall retention	1 hour (3 MW)

Key plasma parameters (Ti: ion temperature, Te: electron temperature, electron density, beta value, and steady-state operation) achieved in LHD (hydrogen plasma experiment phase) are summarized along with the relevant key physics and the target values of these parameters.

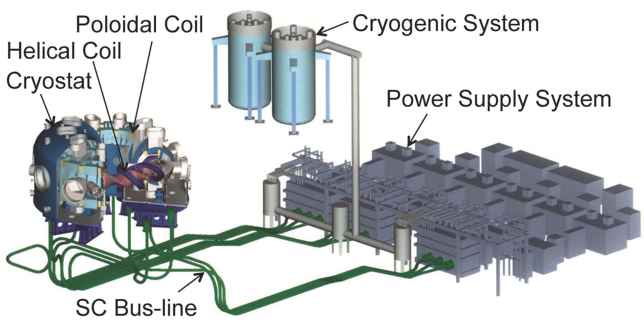


Fig. 1. Superconducting system of LHD [7, Fig. 1].

Progress toward high-performance plasmas has also been made in recent years in LHD. Table I shows the summary for key plasma parameters achieved in hydrogen plasma experiment phase. Plasma temperature, density, and beta values, have been progressing, accompanied with physics findings, highlighted by impurity hole formation [6], super dense core mode [7], and so on. Target values are also indicated in Table I, to be challenged in the ongoing deuterium experiment. It should further extend these parameters toward reactor-relevant regime, in which advanced research can be performed for establishing firm basis for designing steady-state helical fusion reactor.

III. Cutting-Edge Engineering Aspects of LHD

The above-mentioned achievements and expected further progress on research have been and will be firmly based on the cutting-edge engineering aspects of LHD in many regards. Some highlighted features of those aspects are described in the following.

A. Large-Scale Superconducting Magnet System

The superconducting system of LHD consists of a pair of helical coils, three pairs of poloidal coils, a cryostat, nine superconducting bus lines, a helium liquefier/refrigerator, and six dc power supply systems, as shown in Fig. 1 [8]. The stored magnetic energy of 0.9 GJ is the largest among the superconducting experimental devices for nuclear fusion research.

TABLE II
AVAILABILITY OF THE LHD CRYOGENIC SYSTEM

Campaign (FY)	Run time [hours] Total	Steady state	Fault time [h]	Availability [%]
1 (1997-8)	2,895	1,350	138.0	95.2
2 (1998)	3,522	2,359	5.3	99.8
3 (1999)	5,361	3,965	12.4	99.7
4 (2000)	5,016	3,538	13.8	99.2
5 (2001)	5,294	3,791	38.2	99.9
6 (2002)	4,920	3,366	0.1	100.0
7 (2003)	5,001	3,446	64.5	98.7
8 (2004)	4,814	3,216	54.1	98.9
9 (2005)	5,035	3,470	2.3	100.0
10 (2006)	5,037	3,436	0	100.0
11 (2007)	5,184	3,607	23.4	99.5
12 (2008)	3,856	2,272	0.2	100.0
13 (2009)	3,688	2,152	7.0	99.8
14 (2010)	4,128	2,591	228.7	94.8
15 (2011)	3,527	2,112	0	100.0
16 (2012)	3,742	1,272	168.0	95.7
17 (2013)	3,581	2,088	0	100.0
18 (2014)	3,717	2,256	0	100.0
Total	78,318	50,287	756.0	99.5

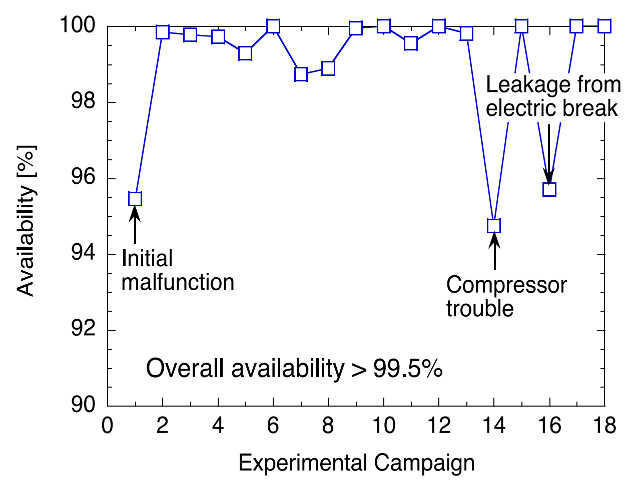


Fig. 2. Availability of the LHD cryogenic system.

The total cold mass at 4 K is 822 ton (the helical coils of 240 ton, poloidal coils of 172 ton, and supporting structures of 410 ton). The helium refrigerator/liquefier has cooling capacities of 5.65 kW at 4.4 K, 650 L/h liquefaction, and 20.6 kW at 80 K [9]. Since the completion of eight-year construction (1990–1997), several types of remodeling and replacement have been carried out in the superconducting system to attain high availability, for example, a double-loop control system, additional helium tanks, enhancement of impurity reduction, and replacement to high-speed CPU boards [8]. As the results of the efforts of maintenance, overall availability higher than 99% has been achieved, as shown in Fig. 2 and Table II. Recent major remodelings [10] are summarized here.

At the 14th campaign (fiscal year of 2010), despite of periodic regular inspections, thrust bearings of a screw compressor broke down after the start of the operation. The thrust bearing failure was considered to be caused by accumulated metal powders in the compressor lubricant oil that had not been changed in order to prevent contamination of the cryogenic system with impurities such as water. The bearings and the lubricant oil of the compressors were exchanged for the repair. After the operation in the 14th campaign, the chemical composition of the lubricant oil and the amount of metal powders in the oil has been checked during the maintenance period. In addition, a set of redundant compressors in the first and second stages have been added to back up the existing eight compressors.

During cool down in the 16th campaign, a cold leakage occurred in an electric break of one of the poloidal coils. The electric break is made of a glass fiber reinforced plastic tube that is bonded to metallic conduits with the adhesive. A cold leakage occurred from the bonded part of the tube due to the aging of the adhesive. The 16th campaign was carried out after repairing the electric break with epoxy resin and glass tapes. All electric breaks (120 places) in the poloidal coils were replaced after the completion of the campaign.

The control system of the helium liquefier/refrigerator was designed and developed as an open system utilizing Versa Module Europa (the name of computer bus) controllers and UNIX workstations that was the newest system at the construction of LHD. However, the control equipment has been significantly advanced. Compact Peripheral Component Interconnect (PCI; the name of computer bus) controllers have been adopted in order to simplify the system configuration and to improve the system reliability. The new system consists of compact PCI controller and remote I/O connected with Ethernet/IP. Making the system redundant becomes possible by doubling CPU, LAN, and remote I/O, respectively.

B. Heating Systems

LHD has been equipped with NBI, ECH, and ICH for heating plasmas up to temperature regime relevant to those of fusion reactors, as summarized in Table I. They have also been used for perturbing plasmas such as NBI blip experiment, ECH modulation, to conduct detailed physics research on dynamical behavior of plasmas in [11] and [12]. ICH has been playing crucial roles to explore long-pulse operation [13].

Here, present status of these heating systems is described although all the ICH antennas have been temporarily removed for smooth initiation of deuterium experiment.

1) NBI: Negative-ion-based NBI is an indispensable heating device to modern fusion machine having large plasma size, because longer beam penetration depth is necessary to deposit the beam power at the core part of the larger torus plasmas. In the case of proton/deuteron beam, the neutralization efficiency decreases to zero, when the beam energy increases from 100 keV to 1 MeV. On the other hand, the efficiency of hydrogen/deuterium negative ion (H^-/D^- ion) is almost constant in that energy range and even at higher energies [14]. For this reason, the negative-ion source has been intensively investigated to adopt the NBI systems [15], [16]. The negative-ion-based NBI systems have been developed and utilized for the first time to LHD as well as JT-60 [17], [18]. Now, LHD is the only fusion experiment device equipped with negative-ion-based NBI.

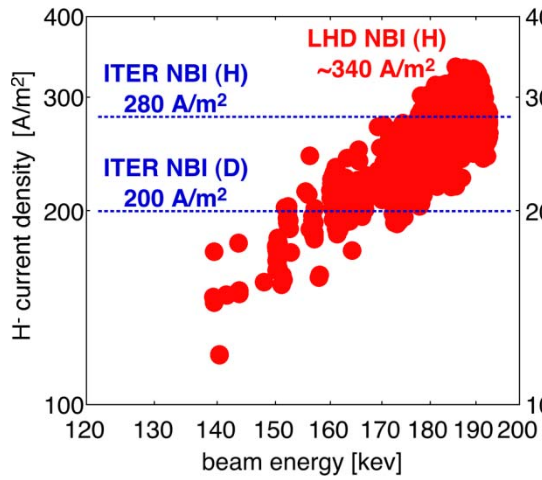


Fig. 3. H^- current density obtained with the new beam accelerator as a function of beam energy.

By means of the developed high transparent accelerator including the slot grounded grid [19], [20], intense H^- beam injection with current of 37 A (340 A/m^2), which is beyond the ITER requirement: 280 A/m^2 for H^- and 200 A/m^2 for D^-, has been achieved [16]. The current density of H^- ion as a function of beam energy is shown in Fig. 3. The minimum beam divergence of the beams extracted with the accelerator is 5 mrad [21] and is better than that in the case of RF sources.

The other important issues to improve the injection power were the increases of Cs-seeding efficiency and arc efficiency, which is defined as a ratio of accelerated H^- current to input arc power. In the former case, reduction of the filament current decreases the tungsten evaporation from heated filament and consequently seeded Cs on the ion source walls is kept cleaner and effective to produce H^- ions. The arc efficiency optimizations for Cs-seeding and magnetic structure improved the injection powers of the other two NBIs and they lead to the stable operation for plasma experiment in LHD.

After the improvements described above, the beam power injected from each negative-ion-based NBI increased year by year. The progress of the total beam power of the three NBIs is shown in Fig. 4. The total port-through power was more than the nominal injection power of 15 MW for several years and the maximum power of 16 MW has been achieved with the negative-ion-based NBIs in the acceleration of H^- ion [2], [22].

In the LHD experimental campaign in 2017, both hydrogen and deuterium injections were conducted for each NBI beamline. The optimal magnetic-field strengths and configurations of negative-ion source are different in H^- and D^- ion beam operations and it was necessary to tune the field strength without opening the ion source to avoid the degradation of Cs effect and for safe to the radioactivity at NBI beamline. The negative-ion sources were fixed to the optimal condition for H^- ion beam except for the magnetic strength of the residual beam dump, and the characteristic difference of H^- and D^- ion beams have been monitored with the same ion source configuration for comparison. Compared with H^- beam

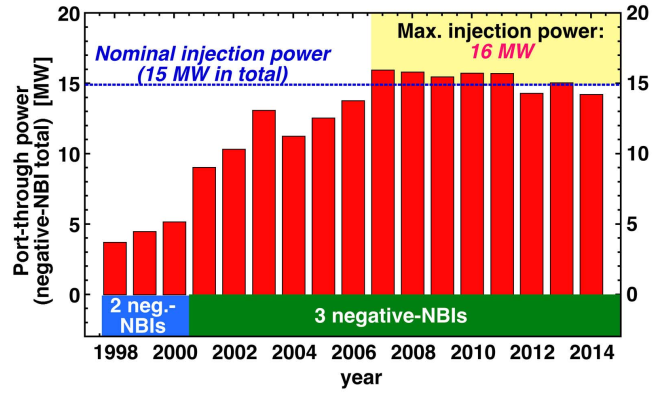
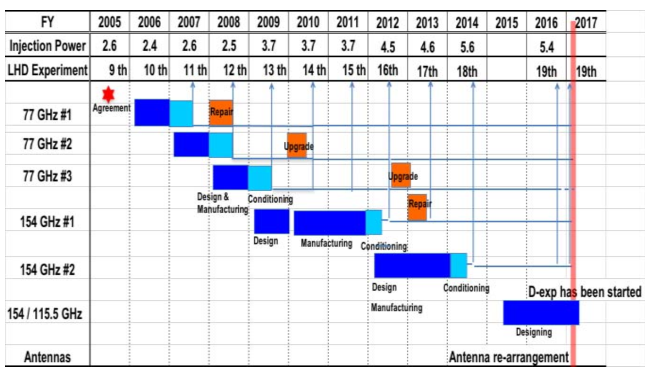


Fig. 4. Progress of total port-through beam power obtained with two NBI systems (1998–2000) and three of them (2001–2014). The maximum port-through power of 16 MW was achieved in 2007, and more than 15 MW of the port-through power was obtained several years after the achievement.

TABLE III

HISTORY OF ECH GYROTRON DEVELOPMENT IN NIFS



The history of ECH gyrotron development in NIFS for those with over 1 MW 77 and 154 GHz, based on collaboration with University of Tsukuba concluded in 2005. Currently, total injection power of 5.5 MW can be injected into LHD plasmas. A dual-frequency (154/115.5 GHz) gyrotron is being designed and partly manufactured.

operation, the D^- ion current density decreased and electron current density increased as expected.

2) ECH: Table III shows the history of the gyrotron development in National Institute for Fusion Science (NIFS). Instead of previous half-megawatt 84- and 168-GHz gyrotrons, we have started to develop over 1 MW 77-GHz and 154-GHz gyrotrons in collaboration with the University of Tsukuba, including some technologies of the 170-GHz International Thermonuclear Experimental Reactor (ITER) gyrotrons of QST (National Institutes for Quantum and Radiological Science and Technology, Japan). After making an agreement on scientific research between NIFS and the University of Tsukuba in 2005, the development of 77 GHz MW gyrotrons was started under the collaboration contract. So far, three 77 GHz and two 154 GHz · MW gyrotrons have been developed and been used for plasma production, electron heating, current drive, and collective Thomson scattering in LHD experiment [23]–[25]. Recently, a dual-frequency gyrotron (154/115.5 GHz) is being designed and manufactured to expand the possible magnetic configurations for heating [26]. Then, over 6-MW injection system will be completed in the near future.

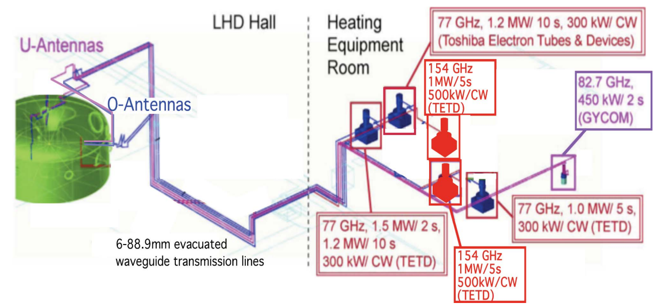


Fig. 5. Present configuration of the ECH system in LHD [28, Fig. 2].

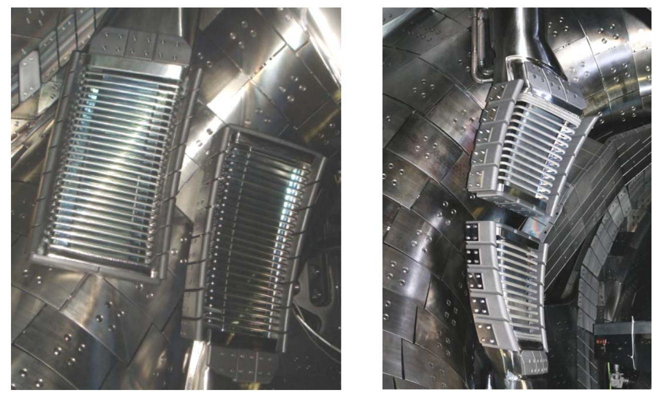


Fig. 6. Two kinds of ICH antennas in the LHD. HAS antenna (left) and FAIT antenna (right).

Fig. 5 shows the latest ECH system in LHD. The system consists of gyrotrons, transmission lines, and antennas. The 77- and 154-GHz gyrotrons contribute to the fundamental and second harmonic ECH at 2.75 T, respectively. The former can also be used at 1.375 T as the second harmonic resonance heating. The transmission lines consist of evacuated corrugated waveguide with 88.9 mm in diameter and several 90° miter bends some of which have the functions as the power monitor, polarizer, and arc detector. All transmission lines are evacuated for high-power transmission. Their transmission efficiencies are over 80%. The injection antennas consist of two to four mirrors. The final mirror is adjustable to change the focal position in the target plasma [27], [28]. Total injection power of 5.5 MW can be injected into LHD plasmas.

3) ICH: Fully controlled stable long-pulse plasma operation is one of the advantageous features for helical systems, and the LHD has demonstrated and extended research in this issue. The long-pulse plasma operations were conducted mainly by ICH with the minority-ion heating scheme and ECH. Heating optimization has been performed in short-pulse plasma duration ($<$10 s). To sustain long-pulse and high-performance plasma, steady-state heating sources have been improved to be up to 3 MW, and relevant plasma physics with long timescale ($>$a few 1000 s) has been investigated by repeated long-pulse plasma operations [30].

A few percent of loss of heating power was a critical issue to make high-performance steady-state plasma duration with heating power in a range of a few MWs, and ICH antenna designs have been investigated to reduce the heating loss associated with parasitic heating at plasma edge and local heat loads by unfamiliar plasma heating (Fig. 6). The hand shake-type (HAS) antenna was designed to confirm the field-aligned effect and large wavenumber excitation by phasing dipole to toroidal direction [31], and maximum plasma heating efficiency (η) of the HAS antenna was obtained with minority ratio of ~10% in He(H). The field-aligned impedance transforming (FAIT) antenna was designed to increase wave power density from the one current strap. However, the wave emitted region of the FAIT antenna was approximately half of that of the HAS antenna, and the impedance transformers were assembled in the FAIT antenna to reduce maximum standing wave voltage (SWV) on transmission lines [32]. Finally, the SWV has decreased to 1/5 with the same power supply without the impedance transformer inside of the vacuum vessel. Large amount of RF power could be derived to the FAIT antenna, and RF power density was up to 9.3 MW/m^2, which is comparable to the power density of ITER ICH antenna. Maximum power density is limited by heating sources, and we have been preparing to increase RF power supply using power combiner [32]. Maximum η using the FAIT antenna was achieved with the minority ratio of ~20% in He(H). Optimized minority concentrations were different between using the HAS antenna and using the FAIT antenna with field-aligned straps, and the optimum concentrations were close using the FAIT antenna and poloidal array antenna [33], which was designed to increase emitted power form single strap without field-aligned strap.

C. Closed Helical Divertor

The control of the neutral particles using a divertor is one of the critical issues for future fusion reactors. The reduction of heat and particle load is also one of the key issues to be resolved. To exploit the inherent advantage of the heliotron concept, that is the built-in helical divertor, a closed helical divertor with baffle structure for high neutral particles compression has been installed in LHD at the inboard side since 2010 [34], [35]. The closure had been completed at nine inner toroidal sections in 2013. Since then, the divertor development has been focused on the installation of the divertor cryosorption pump for the density control by exhausting the particles highly compressed by the closed helical divertor. Cryosorption pumping systems inside the divertor have been installed step by step and now completed in five sections to demonstrate its functions. Here, it is noteworthy that the development of the divertor cryosorption pump has progressed in NIFS [36].

The pumping test of the divertor cryosorption pumps was conducted. The pumping speed and pumping capacity were approximately 70 m^3/s and 58 000 Pa · m^3, respectively. The pumping speed has reached to be equivalent to that of the main vacuum pumping system of LHD, and the pumping capacity is equivalent to even 20 000 hydrogen/deuterium pellets. Based on the progress on the closed divertor system, low base density, and low efficient global particle confinement time (τ_p^*) are obtained during the divertor pumping, as shown in Fig. 7. The low recycling state was obtained by the helical divertor pumping for the first time [37].

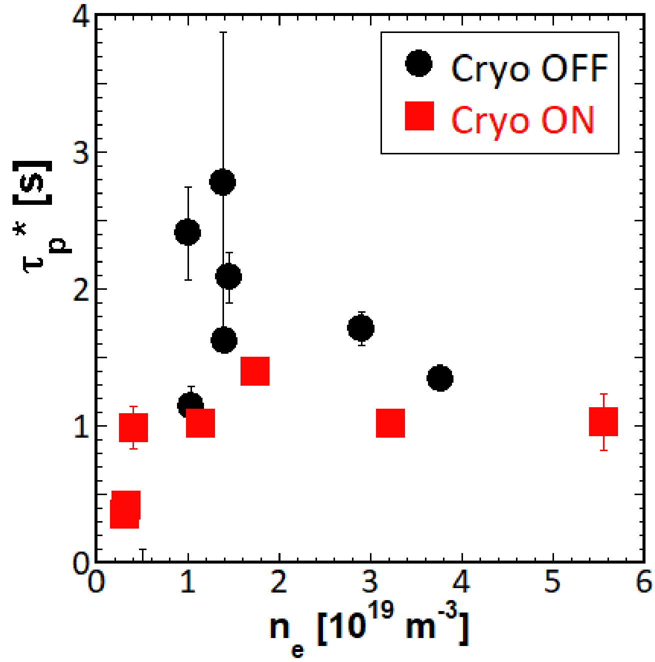


Fig. 7. Global particle confinement time (τ_p^*) for various density regimes for cases with cryosorption pumping OFF (●) and ON (■).

The divertor pumping will contribute to long-pulse discharges. It is observed that the wall recycling affects the density control in the long-pulse discharges in LHD [37]. This powerful divertor pumping will change the global particle balance which does not depend on the wall recycling, leading to good density control. The long-pulse discharge will be planned in latter phase of deuterium experiment.

D. Tritium Removal System

From the viewpoints of tritium safety management and public acceptance, the tritium removal system is one of the key devices to conduct the deuterium plasma experiment in LHD. Since tritium is produced in the LHD vacuum vessel in the deuterium plasma discharge, and then exhausted from the vacuum pumping system, the tritium removal system is installed at the downstream of the pumping system. It mainly consists of the catalytic recombiners for the oxidation of hydrogen isotopes gas and hydrocarbons, adsorption column packing with molecular sieves for tritiated water vapor adsorption, as shown in Fig. 8. The vacuum exhaust gas under the deuterium plasma experiments has some features; high hydrogen isotopes concentrations more than 50%, oxygen-free, dry gas condition which has a moisture dew point below −20 °C, various gas flow rate from low flow rate less than few Nm^3/h, and temporary large flow rate more than 10 Nm^3/h by the regeneration of cryosorption pumps used in NBI [39]. Therefore, the tritium removal system for LHD has been designed to treat various gas stream conditions with a safety processing system for hydrogen oxidation.

The maximum amount of tritium production by the deuterium plasma experiment in LHD is assumed to be annually 55.5 GBq. The tritium gas is exhausted from the vacuum vessel with deuterium gas as main operation gas and passed

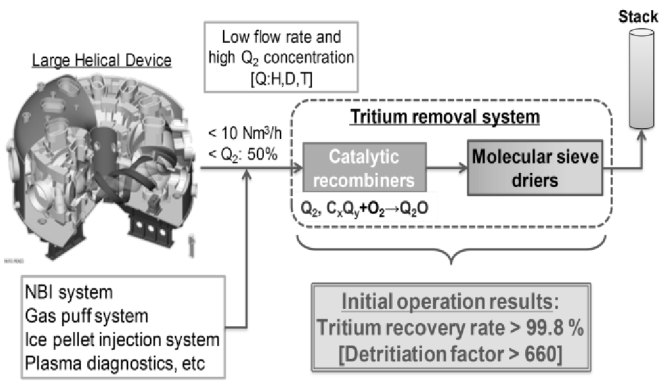


Fig. 8. Schematic of tritium removal system: the tritium removal system treats all exhaust gas from LHD and the auxiliary system.

through the tritium removal system. Then, the treated gas from the tritium removal system is released into the environment via the stack after the dilution of room air ventilation gas. Tritium concentration in the stack is monitored by the active tritium sampler [40]. The annual amount of tritium release from the stack must be less than 3.7 GBq according to the agreement with the local governments. Thus, the detritiation factor of the tritium removal system is required higher than 20 corresponding to tritium recovery rate of 95%.

In the first one month after the start of deuterium plasma experiment, the average tritium concentration in the exhaust gas was about 3.3×10^{-3} Bq/cm^3. This corresponds to about 11% of tritium production in vacuum vessel [41]. Tritium concentration in the exhaust gas gradually increases with the number of plasma discharge shots. As the results of initial tritium removal operation, the tritium recovery rate was successfully achieved more than 99.8% corresponding to the detritiation factor of more than 660.

IV. LHD Entering the Deuterium Experiment Phase

After the conclusion of agreement for the deuterium experiment with local governments in 2013, we have made substantial efforts for accomplishing extensive preparations [42]: not only upgrades of diagnostics and heating devices, but also closed helical divertor, neutron diagnostics system [43], radiation monitoring and safety system, tritium removal system, upgrade of vacuum pumping system, and controlled area management system have been upgraded or newly installed.

Then, the LHD deuterium plasma experiment started on March 7, 2017. The first light of the deuterium plasma is shown in Fig. 9. The main objectives of the deuterium experiment are as follows:

1) high-performance plasmas through confinement improvement for promoting scientific research in more reactor-relevant conditions;
2) clarification of the isotope effect on confinement, which has been the long-standing mystery in world fusion research;
3) the demonstration of the confinement capability of energetic ions in heliotron magnetic configuration for obtaining a perspective toward LHD-type helical reactor;

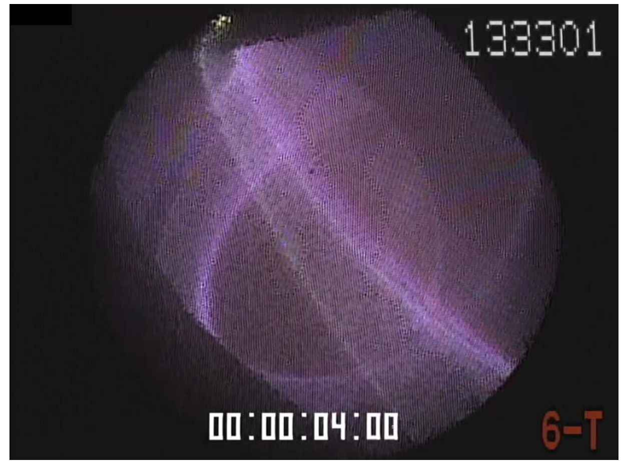


Fig. 9. First light of LHD deuterium plasma on March 7, 2017 (#133301).

4) plasma–wall interaction (PWI) or plasma–material interaction (PMI) study in deuterium plasmas.

Within one week after the start of deuterium experiment, the ion temperature has exceeded 9 keV, which is already beyond the record value in hydrogen experiment phase, 8.1 keV (see Table I), and has been approaching the target value: 10 keV. Also, the neutron emission rate measurement has allowed to conduct quantitative research on energetic particles (EPs) confinement, and also on interaction among MHD modes and EPs confinement, and plasma confinement property such as achievable ion temperature. These remarkable progresses being obtained in the ongoing deuterium experiment will be reported elsewhere.

V. Progress on Conceptual Design of the LHD-Type Helical Fusion Reactor, FFHR-d1

The conceptual design studies on the helical fusion reactor have been intensively conducted by the Fusion Engineering Research Project at NIFS based on close collaboration with LHD experiment. The latest design, FFHR-d1, is four times as large as LHD, and the major radius of the helical coils is 15.6 m [44]. The operation point has been explored by a design integration code, HELIOSCOPE [45], using the "direct profile extrapolation" of the LHD plasma parameters [46]. A self-consistently obtained operation secures the energy multiplication factor $Q \sim 10$. The confinement improvement in the ongoing deuterium plasma experiments, when confirmed, should lead to the self-ignition ($Q = \infty$).

For the engineering design of FFHR-d1, several innovative ideas have been proposed for the following three purposes:

1) to overcome the difficulties related with the construction and maintenance of 3-D complicated large structures;
2) to enhance the passive safety;
3) to improve the plant efficiency.

They are summarized in Fig. 10. For the superconducting magnet, the high-temperature superconductor (HTS) using REBCO tapes is considered [47] as an alternate option to the cable-in-conduit conductor using low-temperature superconducting Nb_3Sn strands. One of the purposes for selecting HTS is

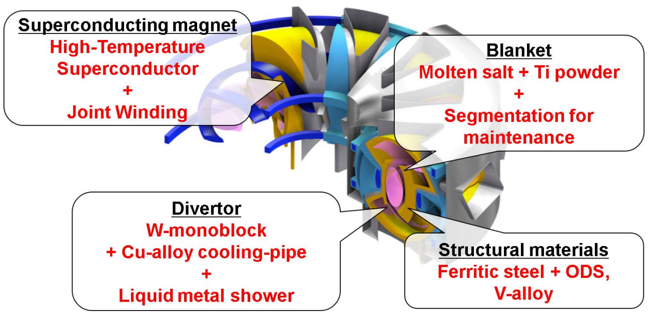


Fig. 10. Innovative engineering ideas to realize the helical fusion reactor FFHR-d1.

to facilitate the 3-D winding of helical coils by connecting prefabricated segmented conductors. A mechanical lap joint technique with low joint resistance has been developed at Tohoku University. About 3-m-long short-sample conductor has successfully achieved 100-kA current at a magnetic field of 5 T and a temperature of 20 K.

For the tritium breeding blanket, we have chosen the liquid blanket option with molten-salt FLiNaBe (melting point: 580 K) from the view point of passive safety [44]. To increase the hydrogen solubility, an innovative idea was proposed to include powders of titanium [44]. An increase of hydrogen solubility over five orders of magnitudes has been confirmed in an experiment [48], which makes tritium permeation barrier less necessary for the coating on the walls of cooling pipes. The "Oroshhi-2" facility [49] was constructed as a platform for international collaboration, having a twin loop for testing both molten-salt (FLiNaK) and liquid metal (LiPb) under the perpendicular magnetic field of 3 T. For the structural material of blankets, a dissimilar bonding technique has been developed to join the vanadium alloy, NIFS HEAT2 [50], and a nickel alloy.

For the helical built-in divertor, the divertor tiles are placed at the backside of the blankets where the incident neutron flux is sufficiently reduced by an order of magnitude [51]. It is thus expected that a copper alloy could be still used for cooling pipes since the maximum neutron fluence is limited to be lower than the allowable limit of $\sim$1 dPa for copper within the operation period. It is noted that the peak heat flux on the helical divertor is expected to reach up to $\sim$20 MW/m^2 because of the nonuniform strike point distributions, but this heat flux could be still accommodated using copper alloys for cooling pipes. With direct irradiation of incident neutrons on the divertor tiles, the reduced-activation ferritic/martensitic steel could be the solution, with which the heat flux should be limited to be lower that $\sim$5 MW/m^2 heat load profile. The maintenance scheme for the full-helical divertor is also a critical issue. To solve these problems, a new concept of liquid divertor, REVOLVER-D, has been proposed [52]. Ten units of molten tin shower jets (falls) are proposed to be installed on the inboard side of the torus to intersect the ergodic layer. It is proposed that the vertical flow of tin jets could be stabilized using metal chains imbedded. This works as an ergodic limiter, and the full-helical divertor becomes less necessary although they could, or should, be situated at the rear. Neutral particles are expected to be efficiently evacuated through the gaps between liquid metal showers.

Maintenance is one of the important and difficult issues to realize the helical fusion reactor. For the blanket, a toroidally segmented system, T-shell, was proposed [53] divided in every 3° in the toroidal angle. Another new idea is the cartridge-type blanket concept, CARDISTRY-B [54]. The discussion about the maintenance scheme both for the blankets and divertor is ongoing.

VI. Concluding Remarks

LHD has progressed as a large-scale superconducting device since 1998, without any severe cryogenic troubles. Improvement of plasma performance and demonstration of steady-state operation have been made based on high-specification and reliable technological and engineering bases, as described in this paper.

The deuterium experiment has just been successfully initiated, and will further provide research opportunities in more reactor-relevant regime. It should also make an impact on ongoing LHD-type helical reactor design.

The LHD and engineering research and development platforms are highly open to domestic and international collaborators, to which your active participation is anticipated.

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