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RECENT STUDIES OF THE LARGE HELICAL DEVICE

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Keywords; LHD, helical system, heliotron, stellarator, superconducting magnet

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ABSTRACT

The Large Helical Device (LHD) is presently under construction (1990-1997) in NIFS, Japan. The magnetic configuration is chosen as an optimized heliotron with double helix coils, succeeding Heliotron-E, ATF and CHS. Target plasma parameters of the LHD with fully superconducting coil systems are; $n\tau_E T \sim 10^{20} m^{-3} \cdot s \cdot keV$, $\bar{\beta}=5\%$ and steady operations with R_p=3. 75m, a_p=50-65cm and B_0=3-4T. The major objectives of the LHD project are to demonstrate good plasma confinement, high beta achievement and steady-state divertor operation in the helical system.

Various R&D results have been obtained on superconducting coils, negative ion source and special diagnostics for LHD. In our new site (Toki-city, Gifu-prefecture), the construction of the main experimental hall of the LHD Building was finished, and the winding of the helical coil has been started there. The construction of LHD will be completed in 1997.

1. INTRODUCTION

There is a growing interest in helical systems from the view-point of demonstrating a steady state helical system reactor. They are characterized by several merits such as steady-state operation without disruption, built-in divertor configuration, simple plasma control and small circulating power. For the demonstration of these advantages, the Large Helical Device (LHD) [1,2] is presently under construction (1990-1997) in NIFS, Japan.

The major objectives of LHD projects are

(1) good plasma confinement, high temperature and high beta achievements,
(2) steady-state operation with divertor,
(3) complementary studies with tokamak researches, and
(4) development of reactor technology.

For these purposes, world's largest superconducting helical coil system was adopted for LHD.

Figure 1 shows a schematic drawing of LHD machine with superconducting helical and poloidal field coils. The major radius of the main torus R_c is 3.9m, and the helical coil minor radius a_c is 0.975m. Three sets of poloidal coils are installed for positioning and shaping of the plasma column.

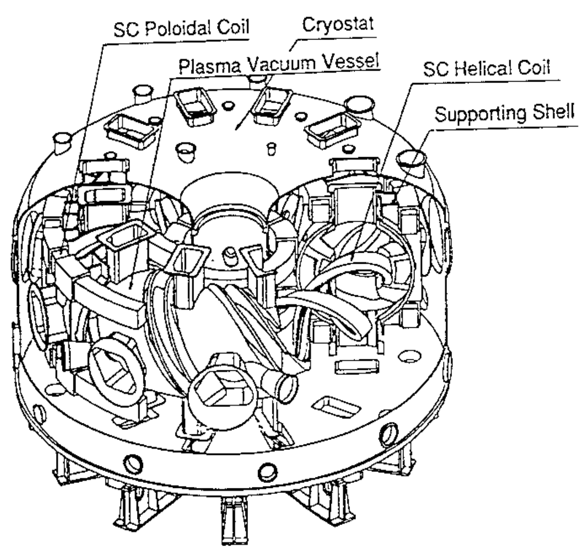


Fig.1 Schematic drawing of LHD

The magnetic configuration is an optimized heliotron with double helix coils, succeeding Heliotron-E, ATF and CHS. Target plasma parameters of the LHD with fully superconducting coil systems are; $n_0\tau_E T_{i0} \sim 10^{20} m^{-3} \cdot s \cdot keV$, $\bar{\beta}$=5% and steady operations with Rp=3.75m, a_p=50-65cm and B_0=3-4T. Figure 2 shows $n_0\tau_E$ - T_{i0} diagram for various fusion machines including target region for LHD. This $n_0\tau_E T_{i0}$ value of LHD is lower than the present front of Tokamak researches, however, the long pulse operation of LHD (Fig.3) leads to the new regime aiming at the next-generation device ITER and future commercial reactors. Different from Tokamak operations, helical systems do not require high-power current-drive and give rise to the easy access to steady-state reactors.

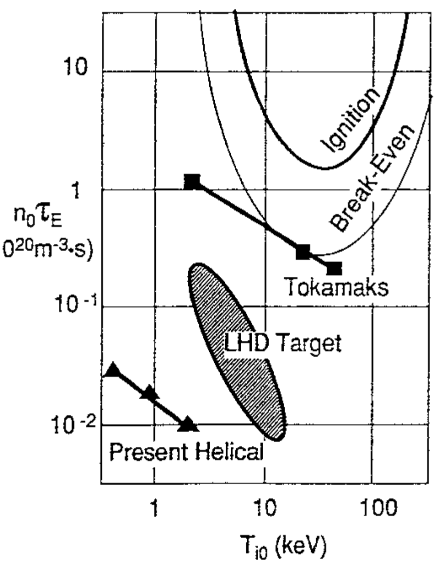


Fig.2 LHD target region in $n_0\tau_E$ - T_{i0} diagram

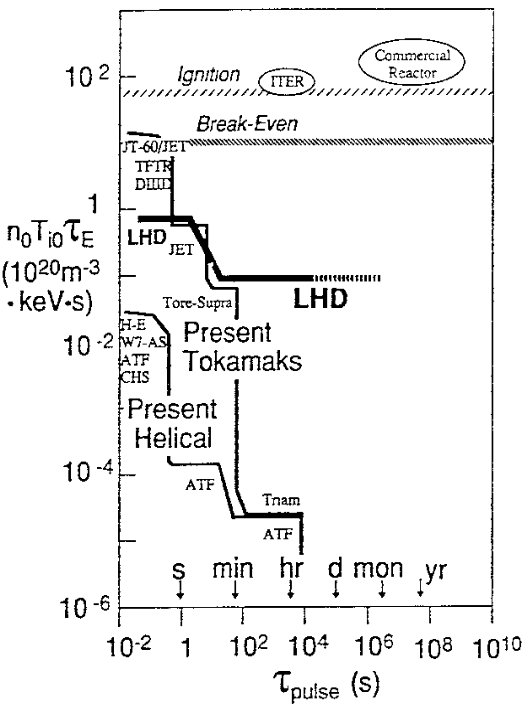


Fig.3 Long pulse operation regime

2. LHD BASIC DESIGN AND OPTIMIZATION

The configuration of LHD is based on the Japanese original Heliotron concept with continuous helical coils and several poloidal field coils. The continuous coil design is superior to the modular coil design because the clean magnetic field configuration with divertor separatrix can be produced. This is very important in the real experiment to clarify the points of physics. The basic optimization of LHD magnetic coil configurations were performed with respect to physics [3] and engineering[4].

The physics aspects, such as high-β achievement, good particle orbit, appropriate divertor-coil clearance and reliable transport scaling, are taken into account to choose standard LHD configurations. Figure 4 shows a typical example of these optimizations with equilibrium-beta, stability-beta and divertor-coil clearance. The helical pitch number m and the pitch parameter $\gamma\ (= m a_c/\ell R_c)$ of the helical coil is optimized for LHD configuration. The final LHD system is determined by m =10 and γ =1.25, and positive pitch modulation α =0.1 . The standard configuration of LHD (Fig.5) is 15cm inward shifted for the improvement of plasma parameters. The major plasma and machine parameters are summarized in Table I and II.

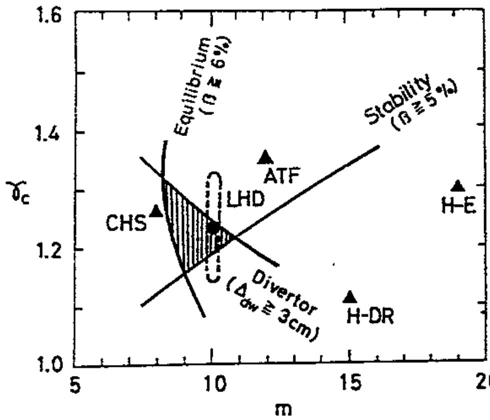


Fig.4 Optimization of magnetic configuration (from ref.[3])

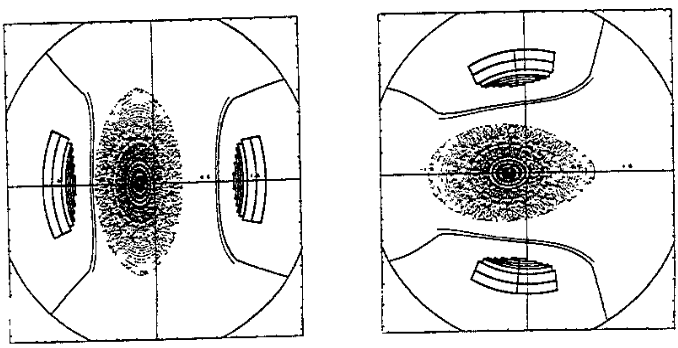


Fig.5 Magnetic surfaces of LHD standard configuration

Table II. Major parameters of LHD superconducting magnets

	Phase - I	Phase - II
Helical Coil		
cooling type	pool-boiling	
superconducting material	NbTi	
major radius	3.9 m	
magneto-motive force	5.85 MA	7.8 MA
coil current density	40A/mm^2	53.3A/mm^2
cooling temperature	4.4 K	1.8 K
maximum field	6.9 T	9.2 T
Poloidal Coil		
cooling type	forced-flow	
superconducting material	NbTi	
OV major radius	5.55 m	
magneto-motive force	-4.5 MA	
maximum field	5.0 T	
IS major radius	2.82 m	
magneto-motive force	-4.5 MA	
maximum field	5.4 T	
IV major radius	1.80 m	
magneto-motive force	5.0 MA	
maximum field	6.5 T	

Table I. LHD configuration parameters

	Phase - I	Phase - II
Machine Configuration		
ℓ	2	
m	10	
γ (picth parameter)	1.25	
α (modulation parameter)	0.1	
major radius	3.9 m	
magnetic field strength	3.0 T	4.0 T
stored magnetic energy	0.9 GJ	1.6 GJ
Plasma Configuration		
plasma major radius	3.75 m	
plasma minor radius	0.6 m	
plasma volume	30 m^3	
$\iota(0)/\iota(a)$	0.4 / 1.3	
Heating System		
ECH Power	10 MW	10 MW
NBI Power	15 MW	20 MW
ICRF Power	3 MW	9 MW

3. PHYSICS ANALYSIS ON LHD

The detailed design parameters of the LHD plasma are optimized in terms of particle transports, equilibrium, stability, bootstrap current and divertor property. Several new numerical codes have been developed for three dimensional analysis on MHD equilibrium, stability and transport.

First of all, the equilibrium of helical systems is studied. The good magnetic surfaces are sometimes destroyed by the error field and plasma beta effects (Fig.6). For this analysis, the HINT code has been developed and clarified the island formation due to beta effects[5]. The equilibrium β-limit obtained by this HINT code is shown in Fig.6. The scenario to suppress beta-induced islands by means of several compensation coils is also demonstrated by this HINT code.

MHD stabilities have been extensively studied by VMEC/STEP code. Figure 7 shows typical results of Mercier mode β-limits in LHD as a function of magnetic axis position[6]. The shaded area denotes unstable region and the 15cm inner shift can permit the achievement of the average beta value of 5%. These results are confirmed by the global mode stability analysis with STEP code.

In helical system, bootstrap currents might deteriorate the plasma stability and make the steady-state operation difficult due to possible current disruption. However, in the standard LHD configuration the bootstrap current is about 150k A for B_0=3T, which corresponds to the prediction by the LHD scaling for 25MW heating and low density case. The further reduction of bootstrap current is easier in helical systems than in tokamaks, because its currents strongly depend on the magnetic configuration, especially ellipticity and axis-shift, and are simply controlled by the poloidal field coils. Figure 8 shows the typical dependence of bootstrap current on central beta value β_0[7]. The circular points denote the self-consistent equilibrium calculation including bootstrap current in comparison with simple analytical scaling (solid line) and the current-free equilibrium analysis (square points)

The transport analysis on these LHD configurations has been carried out by using the global confinement scaling such as the LHD scaling[8], the neoclassical transport and the theoretical anomalous transport such as drift wave turbulence. The prediction of the LHD plasma confinement based on the empirical LHD scaling is shown in Fig.9. Recently, self-consistent electric-field formation due to fast ion loss during ICRF heating has been analyzed and the importance of radial electric field on LHD is clarified[9].

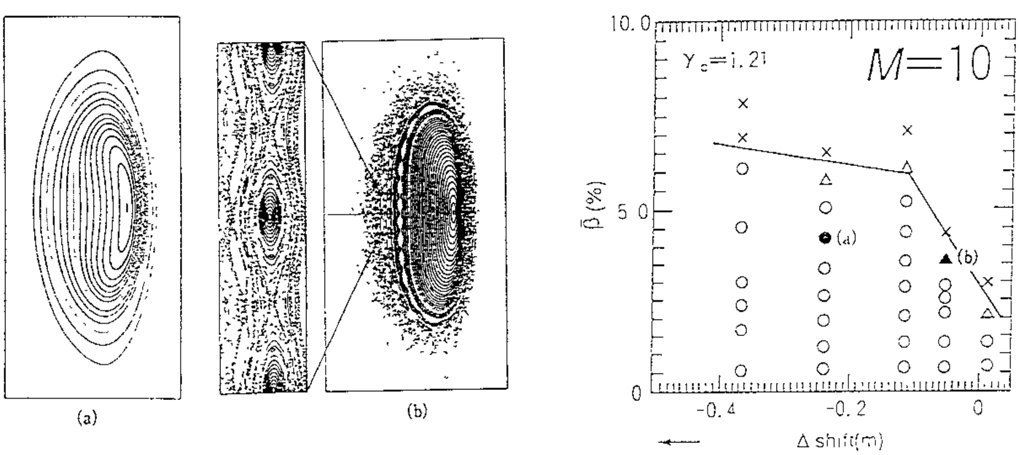
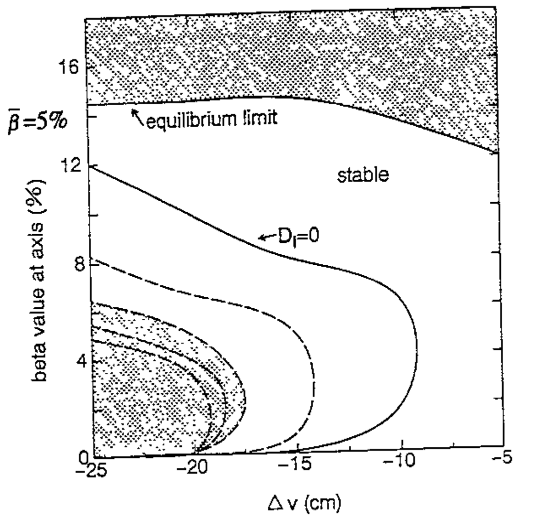
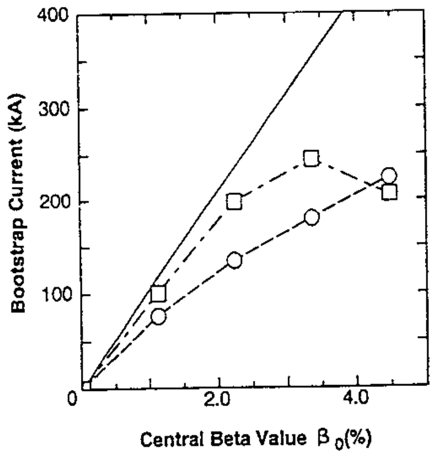


Fig.6 Equilibrium β-limit due to magnetic surface destruction (from ref.[5])
(a) inner shift case (averaged beta $\bar{\beta}$ = 4.5%), and (b) marginal limit case ($\bar{\beta}$ =3%)



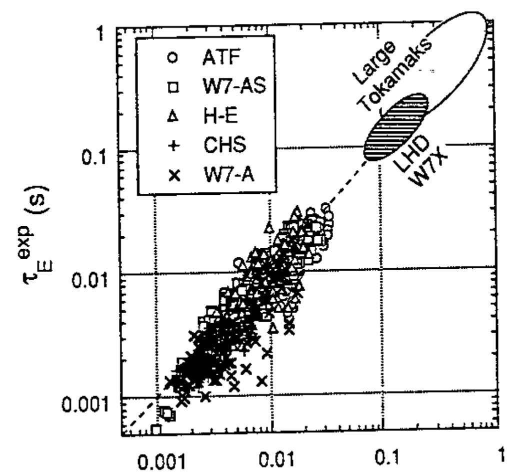
$$P = P_0(1 - \Phi_{norm}^2)^2$$

Fig.7 Mercier stability limits on LHD (from Ref.[6])



$$P = P_0(1 - \Phi_{norm}^2)^2$$

Fig.8 Bootstrap current in LHD (from Ref.[7])



$$\tau_E^{LHD} = 0.17\, P_{abs}^{-0.58}\, \bar{n}_e^{\,0.69}\, B^{0.84}\, a^2\, R^{0.75}$$
$$(s) \quad (MW) \quad (10^{20}m^{-3}) \; (T) \quad (m)\,(m)$$

Fig.9 Plasma confinement prediction based on LHD scaling[8]

4. STEADY-STATE OPERATION SCENARIO

The steady-state operation with high quality plasma is one of the major objectives of LHD. The LHD will be operated by using superconducting coils energized during day-time and the plasma will be maintained by means of steady-state 3~10MW RF heating methods, which in turn requires an active cooling of the vacuum vessel. For the density and impurity controls, helical divertor (Fig.10) is utilized on LHD. We will execute a first divertor experiment in helical system, which will give us fruitful information on helical divertor compared with the tokamak divertor. In addition to the helical divertor, a local island divertor has been recently proposed on LHD. By means of these excellent divertors and elaborate plasma control techniques, new regimes with enhanced plasma performance will be explored.

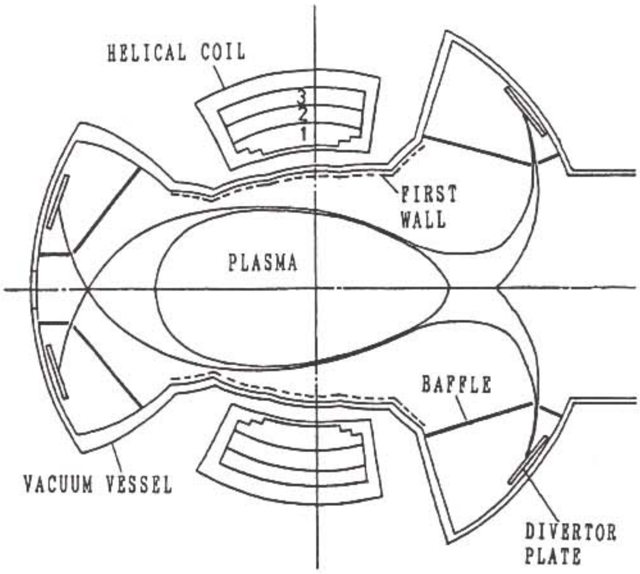


Fig.10
Divertor concept on LHD

5. R&D RESULTS FOR LHD

The various R&D for constructing SC magnet system, heating systems and diagnostic systems have been carried out in the past four years. After elaborate research and development of NbTi superconducting conductors[11], the final design of 13.0 kA (17.3kA in Phase 2) helical coil conductor with 12.5mm x 18.0mm cross-section was decided. The construction of the coil winding machine (Fig.11) for this pool-boiling type helical coil was completed and its winding has been started since this year. It will take more than one and half years. Two sets of forced-flow type superconducting poloidal coils (Inner Vertical coils and Inner Shaping coils) were constructed, and the largest poloidal coil set (Outer Vertical coil, 5.55m major radius) is now under construction. Figure 12 shows the cross-section of conductor and the coil configuration for IV coils. The lower half of cryostat, the liquid helium refrigerator and the poloidal coil power supply were also completed last fiscal year. All components for main torus have been designed and constructed on schedule.

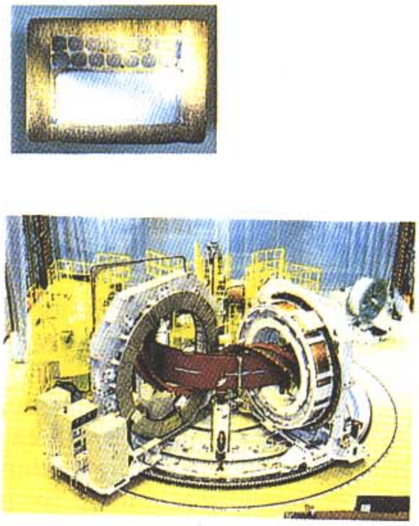


Fig.11 Helical coil winding machine
and the superconducting conductor

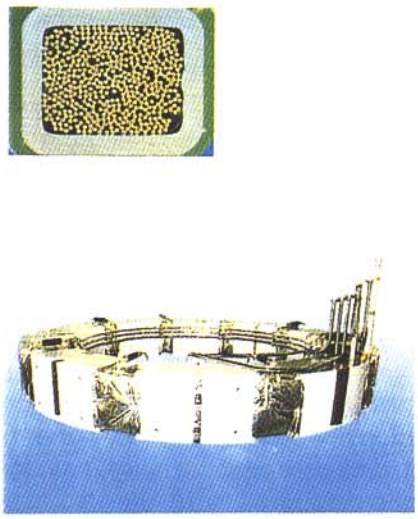


Fig.12 Inner vertical coil and
the superconducting conductor

Various R&D tests for ~20MW NBI system using negative ion source, ~10MW ECH(83GHz/166GHz) gyrotron and ~10MW ICRF antenna system have been performed. Several diagnostic equipments such as HIBP electric potential measurement and YAG Thomson scattering systems have also been tested for LHD. Especially, R&D researches for negative ion source with 45A, 125 keV H^- beam (Fig.13) have been successfully conducted[12]. Using 1/3-scale proto-type equipments, a world record of 16A extraction of negative ion beam has been performed with the current density of $45mA/cm^2$ (Fig.14).

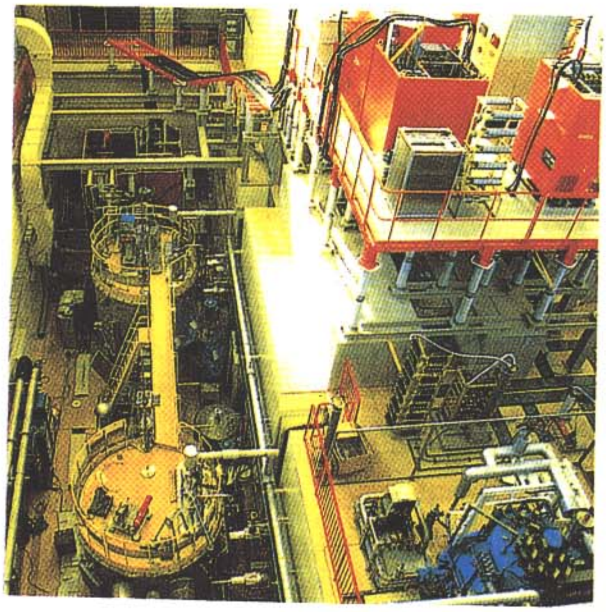


Fig.13 R&D facility for negative ion source and its power supply.

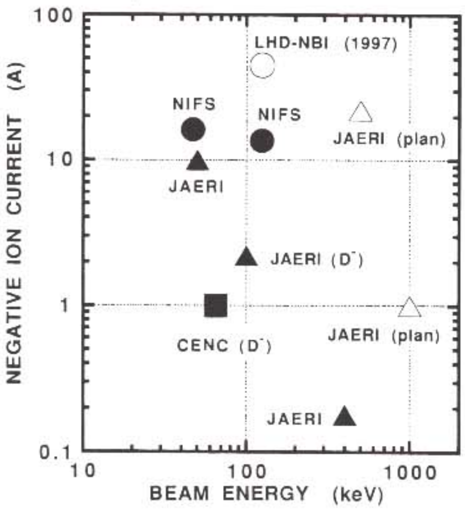


Fig. 14 Progress of negative ion beam development

6. SCHEDULE AND SUMMARY

In our new site (Toki-city, Gifu-prefecture), the construction of the main experimental hall of the LHD Building (Fig.15) was finished last fiscal year, and the above-stated winding machine and the lower cryostat were installed there.

Fig.15 Photograph of LHD Experimental Building.

The construction of the largest superconducting poloidal coil is also started in this experimental hall. Both the machine and the building constructions are on schedule.

In summary, the further understanding of LHD physics has been developed, and the steady-state operation scenarios have been clarified. After various R&D on superconducting coils, heating systems and diagnostic equipments, we have come to the stage of the fabrication of the real machine. The construction of the LHD machine will be completed in 1997.

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