ISSN 0915-633X

INTERNATIONAL ATOMIC ENERGY AGENCY

THIRTEENTH INTERNATIONAL CONFERENCE ON
PLASMA PHYSICS AND CONTROLLED NUCLEAR FUSION RESEARCH

Washington, DC, United States of America, 1–6 October 1990

IAEA-CN-53/G-1-5

NATIONAL INSTITUTE FOR FUSION SCIENCE

Engineering Design Study of Superconducting Large Helical Device

O. Motojima, K. Akaishi, M. Asao, K. Fujii, J. Fujita, T. Hino, Y. Hamada, H. Kaneko, S. Kitagawa, Y. Kubota, T. Kuroda, T. Mito, S. Morimoto, N. Noda, Y. Ogawa, I. Ohtake, N. Ohyabu, A. Sagara, T. Satow, K. Takahata, M. Takeo, S. Tanahashi, T. Tsuzuki, S. Yamada, J. Yamamoto, K. Yamazaki, N. Yanagi, H. Yonezu, M. Fujiwara, A. Iiyoshi and LHD Design Group

(Received – Sep. 14, 1990)

NIFS-51 Sep. 1990

RESEARCH REPORT
NIFS Series

NAGOYA, JAPAN

INTERNATIONAL ATOMIC ENERGY AGENCY

THIRTEENTH INTERNATIONAL CONFERENCE ON
PLASMA PHYSICS AND CONTROLLED NUCLEAR FUSION RESEARCH

Washington, DC, United States of America, 1-6 October 1990

IAEA-CN-53/G-1-5

Engineering Design Study of Superconducting Large Helical Device

O. Motojima, K. Akaishi, M. Asao, K. Fujii, J. Fujita, T. Hino, Y. Hamada, H. Kaneko, S. Kitagawa, Y. Kubota, T. Kuroda, T. Mito, S. Morimoto, N. Noda, Y. Ogawa, I. Ohtake, N. Ohyabu, A. Sagara, T. Satow, K. Takahata, M. Takeo, S. Tanahashi, T. Tsuzuki, S. Yamada, J. Yamamoto, K. Yamazaki, N. Yanagi, H. Yonezu, M. Fujiwara, A. Iiyoshi and LHD Design Group

National Institute for Fusion Science
Chikusa-ku, Nagoya 464-01, Japan

ABSTRACT

Large Helical Device (LHD) is a superconducting heliotron/torsatron type device. The SC coil system is composed of ℓ = 2 helical coils and 3 sets of poloidal coils with a total stored magnetic energy of 1.63 GJ. The main machine parameters, m number, ℓ number, major radius, coil minor radius, magnetic field, plasma minor radius, and plasma volume are 10, 2, 3.9 m, 0.975 m, 4 T, 0.65 m, and 30 m^3, respectively. This is an alternative toroidal device which aims at producing plasmas extrapolatable to the reactor regime. The currentless steady operation is the final goal of our LHD program, and there is no danger from the major current disruptions. The material of the super conductor is NbTi, and the cooling systems are pool-boiling for helical coils and forced-flow for poloidal coils. Since the current density of the helical coils is as high as 53.3 A/mm^2 with the maximum experienced magnetic field strength of 9.6 T, the refrigeration with the super-fluid helium is required. LHD has a divertor to control the steady particle recycling and to improve the confinement potentiality. The vacuum vessel has a dumbbell shaped poloidal cross-section making it possible to install the closed divertor chamber. The necessary R&D programs and detailed design are now in progress, and we start the construction of LHD from the next year. The construction of LHD will be completed in 1997.

K E Y W O R D : large helical device, LHD, engineering design, R & D, superconducting, super-critical helium, super-fluid helium, helical coil, cryostat, poloidal coil, support structure, vacuum vessel

1. Introduction

A Large Helical Device (called LHD) project is a major fusion activity of joint universities in Japan belonging to the Ministry of Education, Science and Culture. The construction of LHD is planned with superconducting (SC) coil systems at new Toki site, Toki city, Gifu Prefecture during the coming 7 years. The experiment will begin in 1997, April. LHD has a heliotron/torsatron configuration, and is regarded to mark a principal milestone in the alternative magnetic fusion approach. Due to the physics understanding of toroidal currentless plasma, LHD is expected to demonstrate the potentiality of the helical configuration attaining advanced plasmas extrapolatable to the reactor regime. Straightforwardly, the objective Q value is close to 0.1. The most strong impact of our project influencing to the tokamak approach is expected to be the demonstration of the predominance of the currentless and disruption-free features of steady plasmas. Therefore, the construction of SC device is required and the adequate divertor design is also required to develop the steady particle recycling control and to establish a scenario of confinement improvement due to the edge control.

Design efforts have been concentrated on the optimization of machine parameters self-consistently satisfying the physics requirements and engineering constraints. It is physically required for LHD design to bringing out the good properties for confinement and MHD, and long confinement capability for high energy particles. The magnetic surfaces necessarily have enough distance between wall and plasma boundary, good mod-B structure, enough magnetic shear θ (up to ~ 0.4), magnetic well (up to 1.5%) and large connection length of the magnetic field line for typical divertor field lines ($L \sim 3 \cdot 2\pi R$). Technically, the careful attention should be paid to the current density of the coils, stress level of support structure, toughness of vacuum vessel, selection of plasma facing material, and fabrication scenario.

LHD has been designed for past four years by Design Group organized inter universities in Japan [1,2,3] and was succeeded to the newly established National Institute for Fusion Science last year. The activity was accelerated by the new institute [4,5,6]. At present, necessary R & D program are in progress at new site. This paper deals with the engineering and technical aspects of LHD. Related physics design considerations are given in another paper of this conference [7].

2. Specifications and General Design Characteristics of LHD

The major missions of LHD are; (1) to study on transport for wide-ranged plasma parameters ($T(0) \leq 10$ keV, and $n \tau T = 2 \sim 10 \times 10^{19}$ keV/m^3sec), (2) to produce a high beta value of 5%, and (3) to attain a quasi-steady state operation using the helical divertor. The specifications of LHD are listed in Table 1. As is shown in the table, the experimental scenario is divided into two phases (I and II). This scenario is effective to reduce the load of the engineering developments and budget program. The coil aspect-ratio was recently reduced from 4.17 to 4 to improve MHD features. The value of $\gamma_c(=m/2\cdot a/R)$ was increased from 1.20 to 1.25 with a small reduction of the major radius (R=4.0 m to 3.9 m). In this optimization, the plasma volume was kept almost constant. It is also found that the slight pitch modulation of helical coils (α =0.1) is desirable for providing a clear separatrix configuration outside of the outermost closed magnetic surface.

In phase II experiment, the magnetic field is raised up to B=4 T, and heating power is also increased. In addition, we apply the real time vertical coil current control for the high β experiment. The maximum field changing rate in time is 0.04 T/sec for helical coil and 0.5 T/sec for poloidal coil. These values are 10 times smaller than those of the same sized tokamak device. In phase II of D-D plasma, the maximum shot per year is supposed 5,000 and total neutron irradiation estimated is 2.4×10^{17} n/shot. Reacting plasma experiments with tritium is not included in this project.

From the engineering viewpoint, much attention has been integrated into the following issues;(1) superconducting helical coils, (2) superconducting poloidal coils, (3) vacuum vessel with divertor, (4) power supply and coil protection circuit, (5) control system, (6) refrigeration system, (7) heating system, and (8) diagnostic system. In this paper, we report the results of these technical developments except (7) and (8). A bird's-eye view of the main experimental building and the poloidal cross-section of LHD are shown in Fig. 1(a) and (b). Outer diameter of LHD are ~13 m. The total weight is ~1,500 ton, of which helium cooled weight is 850 ton (helical coils: 140 ton, poloidal coils: 120 ton, and support structure:590 ton). Helical coils, poloidal coils and supporting structure are put together in the toroidal cryostat. A vacuum vessel with a dumbbell shape is installed between helical coils and plasmas. The complicated three dimensional shape is required to supply a sufficient room for the closed divertor operation. The tightest place locates at toroidally inside portion along the helical coils. At least 5 cm is necessary for the adiabatic shielding. Super insulation and LN_2 shield are installed among this space. Since the deformation brought during cooling down phase is about 2 cm in major radius direction, we have designed adiabatic supporting pedestals which have a sliding structure. More than 80 ports are installed to the vacuum vessel. The biggest one with a lozenge shape is located on the outer equatorial plane. The total value of the port area is ~20 m^2, which accounts for ~10% of the total chamber surface. The baking temperature of the vacuum vessel is about 100℃, and that of the divertor carbon tile is 350℃. The expected heat flux on the divertor plate is 1 kW/cm^2 for 5 sec and 30 MW heating power. This criterion is thought to compose an engineering R & D experiment for the tokamak engineering reactor program. The cooling water pipes of the vacuum vessel is fabricated to the inner surface (plasma side) keeping the feasibility for the necessary maintenance, because the damage due to runaway burst is suppressed at a sufficiently low level in SC helical system.

The current density of the coil package is an important parameter to improve the physics properties of LHD, especially to increase the boundary shear and to supply enough room for divertor. The current density is 40 A/mm^2 for 3 T operation (phase Ⅰ, 4.2K), and is increased to 53.3 A/mm^2 for 4 T operation (phase Ⅱ, 1.8K). These values are much larger than the value attained with the normal copper conductor. The material of the superconductor is NbTi and cooling system is pool-boiling for helical coils and forced-flow cooling for poloidal coils. Since there are no danger from the major current disruption, the AC loss, and repetitive mechanical fatigue problems, the helical steady device has an advantage to pursue the engineering design and to build up the practical construction scenario. Unlike a normal conducting coil device, LHD is also free from the thermal stress. Therefore, we could apply the stiff support structure concept to sustain a severe magnetic force exceeding several tens of thousand tons. To increase the dynamic range of the magnetic surface properties, the helical coils are divided into 3 layers, and independently connected to the power supply. Since a wide range of γ_c from 1.1 to 1.4 is available by changing the coil current in each layer, we may attain the various operation regime, i.e., well or hill, shift of axis, and high shear or low shear.

The size of the main experimental room is $^{w}45\text{m}\times{}^{l}75\text{m}\times{}^{h}40\text{m}$. To shield the neutrons of phase Ⅱ experiments, the required thickness of the wall and roof are 2 m and 1.3 m, respectively. All of the cables, pipes, and beam lines of the experimental room are guided into underground room and re-arranged in the common pit to reduce the neutron streaming. Around the main experimental room, power supplies of coils and heating system, refrigeration systems, cooling water system, diagnostic system, and control system are rationally distributed. The total area of the building is $\sim 20,000$ m^2.

Construction schedule from now is shown in Table 2. Next year, we start the constructions of inner vertical field coil, fabrication machine, and cryostat. Fabrication machine is shown in Fig. 2. It will be completed in 1993, and immediately after the completion, we start to manufacture the helical coils with it in Toki site. It takes one year and half. The vacuum vessel is assembled after the coil fabrication is completed. It takes about one year. The necessary technical preparation is now being developed by the R & D model of the vacuum vessel.

3. Conductor Development

The most urgent subject to be solved in a superconducting coil design is to develop a viable superconductor. From the point of view of large magnetic force ($\sim$ 1,000 ton/m) and torsional angle (Max $\sim 50^\circ$ /m), NbTi conductor was selected as a base material. Since the maximum field of the conductor exceeds 9 T, specifications of the NbTi conductor required for LHD is close to the limiting criterion. We are developing several candidates of the conductor, in which Aluminium is added for the improved stabilization. In Fig. 3, the cross-sections of the newly developed conductors are shown. The location and amount of Aluminum, distribution of NbTi strands are different in each design. Since the resistivity of the each strand is zero, it is necessary to study on the uniformity of the current distribution between strands. We have investigated uniform distribution type (1), multi-layer distribution type (2) $\sim$ (4), and transpose structure type (5). The stability property of conductors has been already confirmed, and no difference of the conductor capability due to the Aluminum stabilizer location has been observed. Anisotropic current distribution has not been observed. The most important result of this conductor R & D is that we have confirmed that our design technique based on the existing empirical stability scaling of the pool-boiling conductor is sufficiently applicable. Feasibility research of the coil fabrication is investigated from three kinds of the practical R & D small model coils (R $\sim$ 1 m, B $\sim$ 2 T, m = 3 $\sim$ 4)[8]. The necessary data base for the coil fabrication have been already obtained.

4. Mechanical Analyses of the Helical Coil and Vacuum Vessel

In helical systems, the error field due to a mechanical deformation of helical/poloidal coils is very dangerous for the magnetic field configuration. The deformation and the stress/strain of helical coils and support structures have been analyzed with a finite element method (FEM) code. The critical issues for designing support structure is to develop a reasonable structural design which ascertains the necessary and high accuracy of the helical coils ($\sim \pm$ 2 mm), because the stress level and the degree of the deformation strongly depend on the structural design idea. In Fig. 4, the typical results with the FEM code are presented for the the top half casing of the helical coils and the structural shell, where the averaged thicknesses of the helical coil casing and structural shell are 50 mm and 100 mm, respectively. Spatial restrictions by ports are taken into account of the calculation. The maximum stress appears on the inside surface of the helical coil casing and reached to ~ 35 kgr/mm^2. The stress level of the shell support is about 20 kgr/mm^2. These values are sufficiently accepted by the structural material of the stainless steel. The deformation of the helical coils and shell is suppressed within 2 mm, and this calculation gives the consistent supporting design.

As described previously, one of the critical issues of the vacuum vessel design is that the gap between helical coil and plasma is limited, especially on the inboard side of the torus. The distance between the helical coil and first wall including coil casing must be less than 15 cm to take the enough distance between the first wall and plasma and to avoid the severe plasma wall interaction. This means that an accurate fabrication and assembly of the vessel ($\sim \pm 5$ mm) are required. We have considered two types of loading on the vessel, atmospheric and magnetic ones. The latter is produced when the poloidal fields are changed in time or bootstrap current disappears in a short period. The FEM was also adopted to evaluate the stress level and deformation of the vacuum vessel.

The most severe stress due to the atmospheric force appears near the largest port on the equatorial plane. It is close to the value of 15 kgr/mm^2. The magnetic loading appears when a large eddy current is induced by a sudden drop of the induced plasma current. The design criterion is based on the condition to withstand the magnetic loading due to 150 kA/1 msec current disruption. In the high $n \tau T$ plasma production, in which the bootstrap current becomes maximum and is flowing in the direction to increase the vacuum rotational transform angle, the direction of the magnetic force is outward which is the same as the atmospheric force. It is concluded from the FEM analysis that the thickness of 15 mm is sufficient for the vacuum vessel to satisfy the design criterion and this is close to the limiting thickness for fabricating the vacuum vessel reliably after completing the helical coil fabrication.

ACKNOWLEDGEMENTS

LHD design work has been undertaken keeping the tight relation with all the staff of NIFS, especially with the theory, heating, diagnostic, CHS, and JIPP T-ⅡU groups of NIFS. We hope to express our thanks to them.

[REFERENCES]

[1] DESIGN GROUP Report, Green Book (1987) March

[2] DESIGN GROUP Report, Blue Book (1988) March

[3] DESIGN GROUP Report, Orange Book (1989) March

[4] MOTOJIMA, O., et. al., Fusion Technology 1988, eds. A.M.Van Ingen (Elsevier Science Publishers B.V., Amsterdam, 1988), pp.402

[5] YAMAZAKI, K., et. al., Fusion Technology 1988, eds. A.M.Van Ingen (Elsevier Science Publishers B.V., Amsterdam, 1988), pp.407

[6] IIYOSHI, A., et. al., Fusion Technology 17 (1990) 169

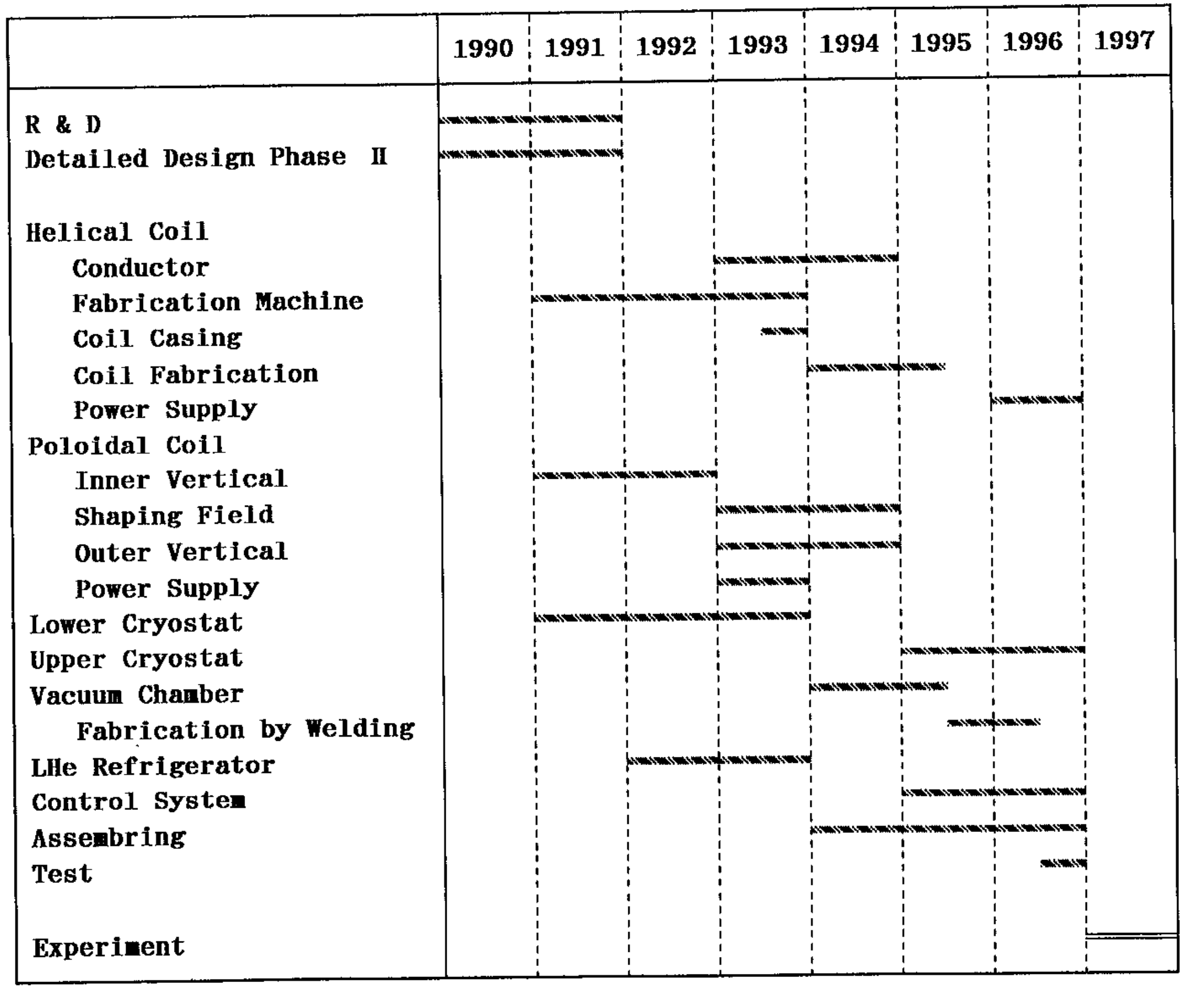
[7] YAMAZAKI, K., et. al., This Conference, IAEA-CN-53/C-4-11

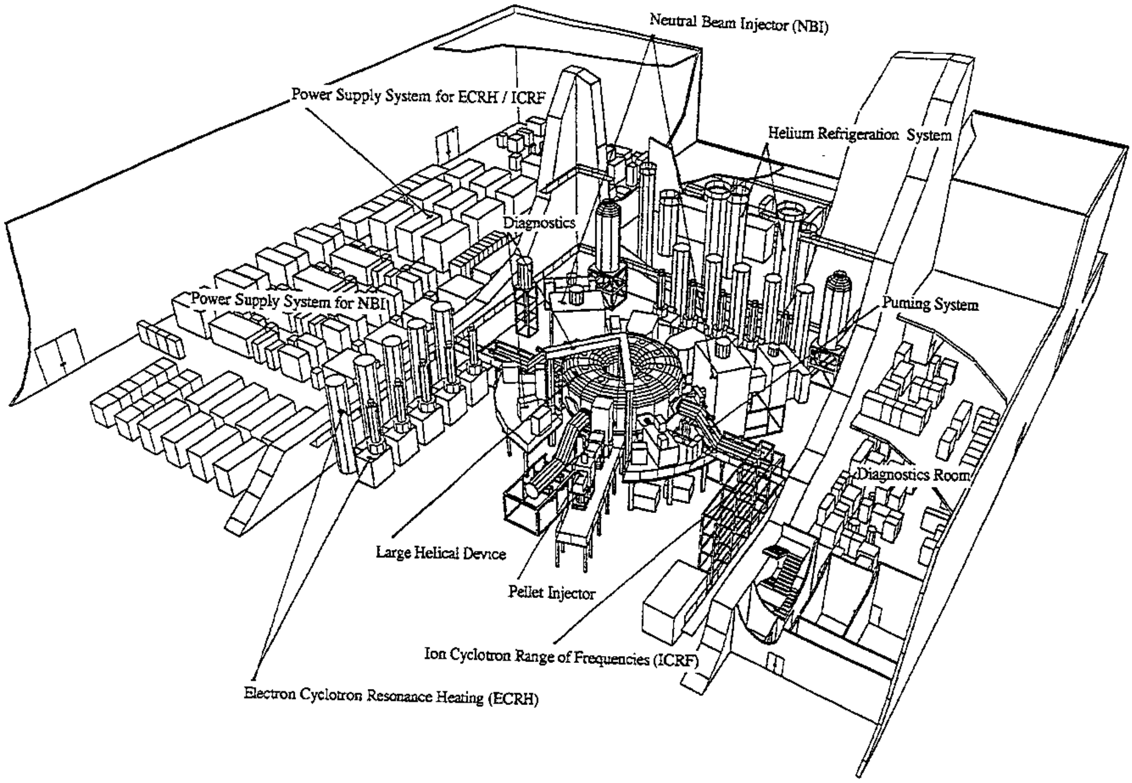
[8] YAMAMOTO, J., et. al., 1990 Applied Superconductivity Conference, Snowmass, USA, LN-2

Table 1 SPECIFICATIONS

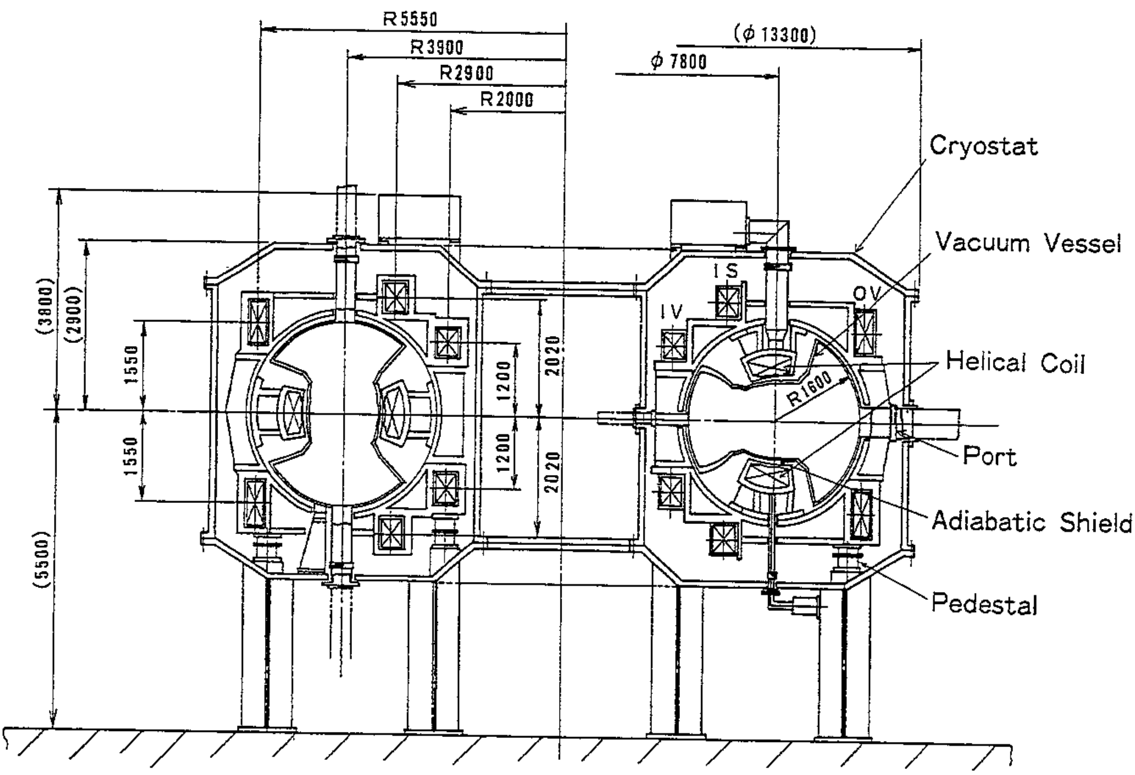
	PHASE I	PHASE II
MAJOR RADIUS	3.9 ■	←
COIL MINOR RADIUS	0.975 ■	←
AVERAGED PLASMA RADIUS	0.5～0.65 ■	←
PLASMA ASPECT RATIO	6～7	←
ℓ	2	←
■	10	←
γ_c=■/2·a_0/R (PITCH PARAM.)	1.25	←
α (PITCH MODULATION FACTOR)	0.1	←
MAGNETIC FIELD		
CENTER	3 T	4 T
COIL SURFACE	7.2 T	9.6 T
HELICAL COIL CURRENT	5.85 MA	7.8 MA
COIL CURRENT DENSITY	40 A/■■^{2}	53.3 A/■■^{2}
NUMBER OF LAYER	3	←
LHe TEMPERATURE	4.2 K	1.8 K
POLOIDAL COIL CURRENT	STEADY	REAL TIME
INNER VERTICAL	-4.4 MA	←
INNER SHAPING	-4.5 MA	←
OUTER VERTICAL	5.0 MA	←
PLASMA VOLUME	20～30 ■^{3}	←
ROTATIONAL TRANSFORM		
CENTER	＜0.5	←
BOUNDARY	～1	←
HELICAL RIPPLE AT SURFACE	0.2	←
PLASMA DURATION	10 sec	←
REPETITION TIME	5 ■in	←
HEATING POWER		
ECRH	10 MW	←
NBI	15 MW	20 MW
ICRF	3 MW	9 MW
STEADY	-----	3 MW
$D^\circ \rightarrow D^+$	-----	PRACTICE
NEUTRON YIELD	-----	2.4×10^{17} n/shot
COIL ENERGY	0.9 GJ	1.6 GJ
REFRIGERATION POWER	5～7 kW	10～15 kW

Table 2 Construction Schedule of LHD





(a)



(b)

FIG. 1 ; (a) A Bird's-Eye View of the Main Experimental Building
(b) The Poloidal Cross-section of LHD

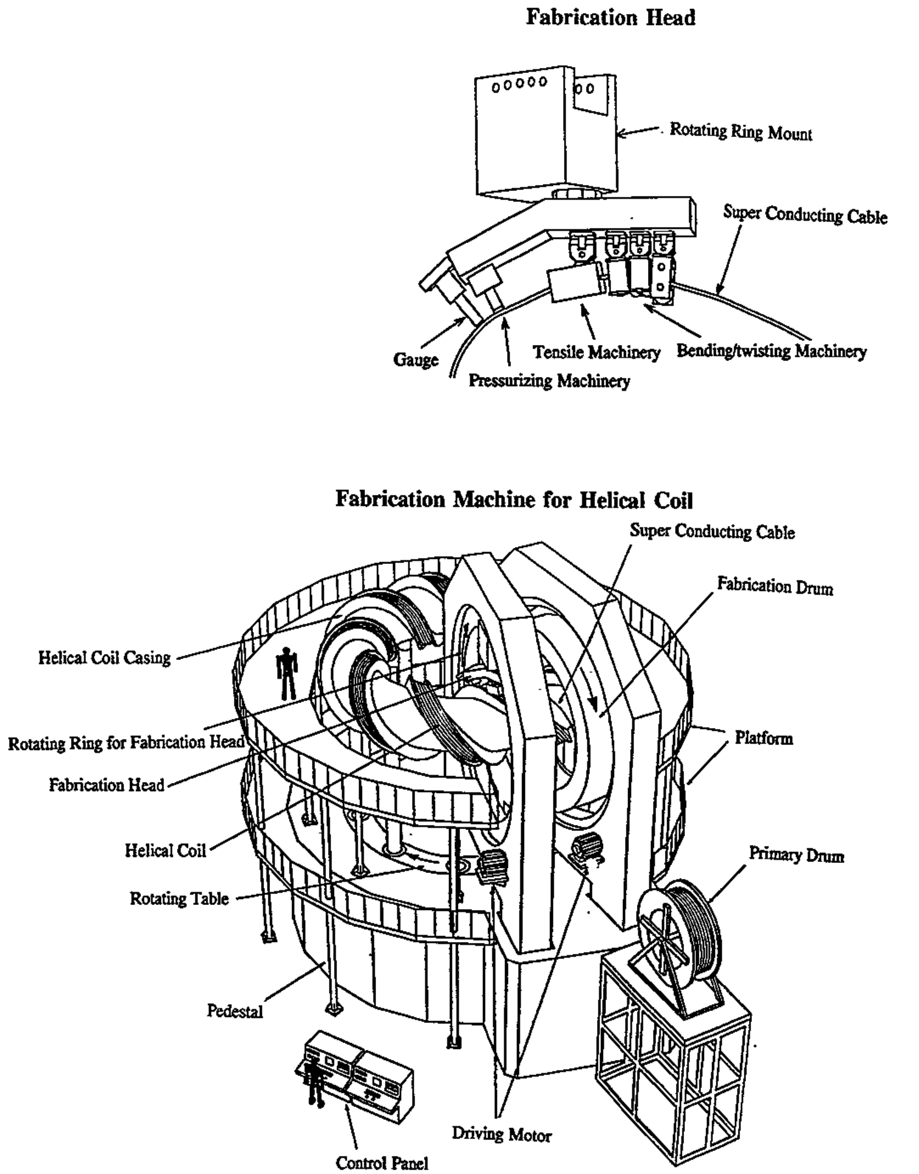


FIG. 2 ; A Bird's-Eye View of Fabrication Machine and Fabrication Head

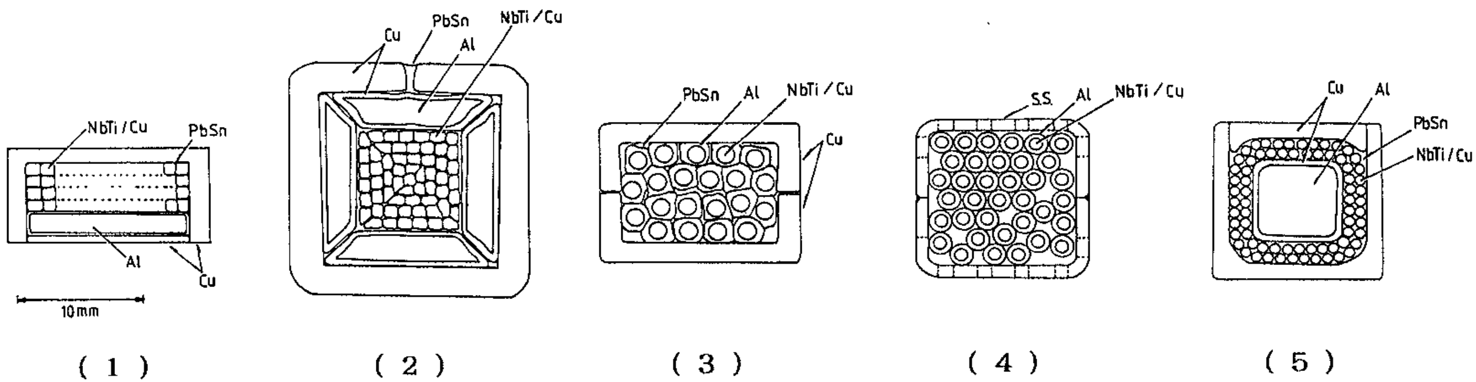


FIG. 3 ; the Cross-section of the Newly Developed Conductor

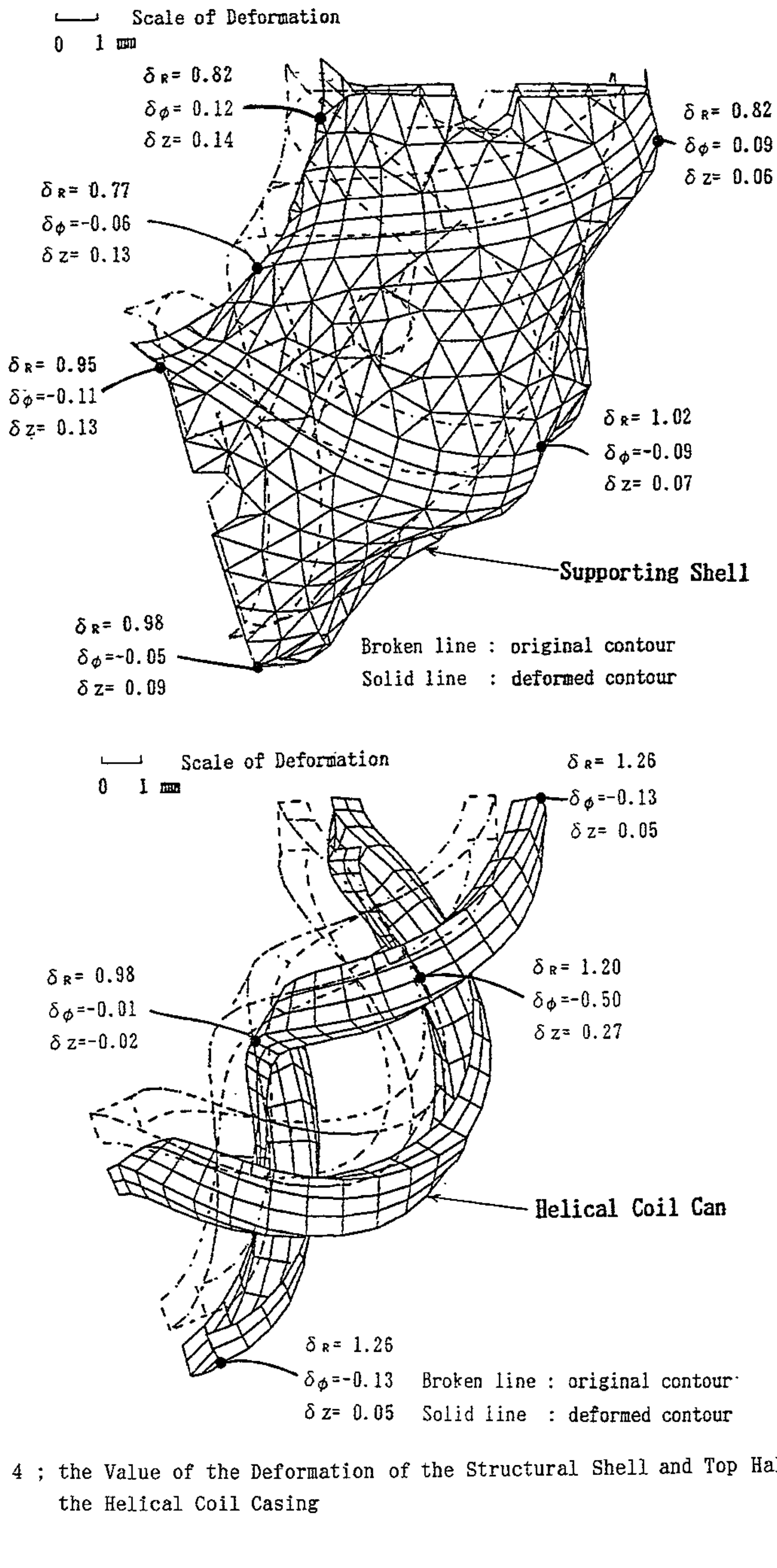


FIG. 4 ; the Value of the Deformation of the Structural Shell and Top Half of the Helical Coil Casing