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Keywords; LHD, Wendelstein 7-X, helical system, heliotron, stellarator, superconducting magnet

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ABSTRACT

Helical systems have strong merits of steady-state operations for fusion reactors. Two large helical devices with fully superconducting coil systems are presently under design and construction. One is the LHD (Large Helical Device, major radius = 3.9m, magnetic field = 3-4T, under construction during 1990-1997) in NIFS (National Institute for Fusion science)-Nagoya/Toki, Japan, which provides with continuous helical coils and clean helical divertor focusing on edge configuration optimization. The other one is the W7-X (Wendelstein 7-X, major radius = 5.5m, magnetic field = 3T, under approval) in MPIPP(Max-Planck Institute for Plasma Physics)-Garching, Germany, which adopted a modular coil system after elaborate optimization studies of core plasma physics. These two programs are complementary to promote world helical fusion research and to extend the comprehensive understanding of toroidal plasmas by comparing with large tokamaks.

I. INTRODUCTION

Helical systems*) are defined as a toroidal plasma configuration produced by the external conductors, and are considered the leading alternative to tokamaks when recalling obvious inherent potential physics advantages: stationarity since the equilibrium does not rely on a net toroidal plasma current, no danger of disruptions for the same reason, less free energy because of the external generation of the entire confining field.

*) Helical system is used as a generic name of stellarator, heliotron, torsatron, heliac and helias.

The two types of next large helical devices, the Large Helical Device (LHD) [1,2] and the Wendelstein 7-X (W7-X) [3] are under design and construction. The LHD aims at clean helical divertor experiments using double helix continuous helical coils as an extension of the medium-sized devices, Heliotron-E, the Advance Toroidal Facility (ATF) and the Compact Helical System (CHS). On the other hand, W7-X adopted modular coil concept for the comprehensive optimization of plasma core confinement with Wendelstein 7-AS (W7-AS), its scientific and technical predecessor. These two devices will play complementary roles in the improvement of the helical concept and the understanding of toroidal plasmas.

In this paper, these two concepts are overviewed. The LHD design and construction are given in Sec. II, and the W7-X design is described in Sec. III. Section IV contains the summary.

II. LHD DESIGN AND CONSTRUCTION

A. Objectives of the LHD Project

For the demonstration of merits of double helix systems (helical polarity $\ell = 2$) , the LHD [1,2] is presently under construction (1990-1997) in NIFS, Japan. The major objectives of LHD projects are

(1) good 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. The schematic drawing of LHD plasma produced by superconducting helical and poloidal field coils is shown in Fig. 1. The magnetic configuration is an optimized heliotron with double helix coils, succeeding Heliotron-E, ATF and CHS. The bird-eye view of LHD machine is shown in Fig.2. 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. Target plasma parameters of the LHD are; fusion triple product $n_0\tau_E T_{i0} \sim 10^{20} m^{-3} s \cdot keV$, averaged beta value $\bar{\beta}$=5% and steady operations with plasma major radius R_p=3.75m, plasma minor radius a_p=50-65cm and magnetic field strength B_0=3-4T. Figure 3(a) shows $n_0\tau_E$ -T_{i0} diagram for various fusion machines including target area for LHD and W7-X. 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(b)) leads to the new regime aiming at next-generation device ITER and future commercial reactors. Helical systems do not require high-power current-drive power and therefore give rise to easy access to the steady-state reactors.

B. Physics Design of LHD

The configuration of LHD is based on the Japanese original Heliotron concept with continuous helical coils and several poloidal field coils. The continuous helical coil design has an advantage for producing magnetic field configurations with well-defined divertor separatrix at free spaces between the double helix coils. This is very important in the real experiment to clarify the points of edge physics.

The physics aspects such as high-β, particle orbit, divertor-coil clearance and transport scaling are taken into account for the determination of the LHD configuration[4]. Especially, the *m*-number and the pitch parameter γ (=$ma_c/\ell R_c$) of the LHD helical coil are chosen for the achievement of equilibrium-β & stability-β (averaged beta value $\bar{\beta} \geq 5\%$) and sufficient divertor-coil clearance. The final LHD system is determined by m =10 and γ =1.25. The optimized standard configuration of LHD is 15cm inward shifted with respect to the major radius for the improvement of plasma performance. It should be noted that the magnetic axis is easily controlled by the external poloidal coils. The major machine parameters and plasma parameters of LHD are summarized in Table I and II.

Through the current-less plasma experiments in the operating helical devices and the progress of simulation analysis by super computers, the understandings of the physics on helical system are progressing day by day. Several new numerical codes have been developed for three-dimensional analysis on MHD equilibrium, stability and transport.

The plasma equilibria in helical systems are studied in details. The good magnetic surfaces are sometimes destroyed by the error field and plasma beta effects. For this analysis, the HINT code[5] has been developed and clarified the island formation due to the increase in the plasma beta value. The suppression of the island formation by means of several compensation coils is also demonstrated by this HINT code.

Extensive surveys of Mercier mode β-limits in LHD[6] are carried out as a function of magnetic axis position. The standard configuration with 15cm inner shift can permit averaged beta of 5%. The global mode analysis with STEP code also confirmed these results.

In helical system, bootstrap currents might deteriorate the plasma stability and make the steady-state operation difficult due to possible current disruption. In the standard LHD configuration with quasi-circular averaged cross-section the bootstrap current is about 150kA at B_0=3T. The further reduction of bootstrap current is easier in helical system than in tokamaks, because its currents strongly depend on the magnetic configuration, especially ellipticity and axis-shift and are simply controlled by the poloidal coils.

The transport analysis on these LHD configurations has been carried out by using global confinement scaling, neoclassical transport and theoretical anomalous transport such as drift wave turbulence. Figure 4 shows the LHD scaling[7] comparing with the existing data of helical systems and large tokamaks. The target values of LHD and W7-X are also indicated. It should be noted that no big deference can be seen in global confinement between helical systems and tokamaks.

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.5)[8] 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 mean of these excellent divertors and elaborate plasma control techniques, a new regime with enhanced plasma performance will be explored.

C. Engineering Design and Construction of LHD

The various R&D for constructing SC magnet system, heating systems and diagnostic systems have been carried out in the past four years. Especially, the scale of LHD superconducting magnets is biggest among those existing and being under construction in the world as shown in Fig. 6, therefore a lot of research and developments are required. After elaborate research and development of NbTi superconducting conductors[9] the final design of 13.0 kA (17.3kA in the experimental phase 2) helical coil conductor with 12.5mm x 18.0mm cross-section was decided. The construction of the coil winding machine (Fig.7(a)) for this pool-boiling type helical coil was completed and its winding has been started this year. It will take more than one and half years. Two sets of forced-flow type superconducting poloidal coils (Inner Vertical coils (Fig.7(b)) and Inner Shaping coils) was constructed and the largest poloidal coil set (Outer Vertical coil, 5.55m major radius) is now under construction. 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.

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 have been successfully conducted[10]. 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 $45mA/cm^2$.

In our new site (Toki-city, Gifu-prefecture), the construction of the main experimental hall of the LHD Building was finished last fiscal year, and the above-stated winding machine and the lower cryostat were installed there. The construction of the largest superconducting poloidal coil is also started in this experimental hall. The LHD machine will be completed in 1997.

D. LHD in Japanese Fusion Program

Fusion researches in Japan have been carried out under two organizations; the Monbusho (Ministry of Education, Science and Culture) and the STA (Science Technology Agency). The Japan Atomic Energy Institute (JAERI) under STA is aiming at the fusion engineering reactor by using Tokamak concept. On the other hand, university institutes including the NIFS (National Institute for Fusion Science) under the Monbusho are making basic researches for a better demonstration reactor by means of proof-of-principle experiments of alternative concepts such as helical systems in the field of magnetic fusion. These two organizations play complimentary roles for the achievement of the fusion reactors. The helical concept has a chance to be taken into account as a proto-type reactor next to the tokamak experimental reactor such as ITER. The fusion research and development require a lot of time and a lot of talented researchers, and this multi-path research strategy in Japan seems important for these long-term requirements.

III. WENDELSTEIN 7-X DESIGN

A. Objectives of WENDELSTEIN 7-X

The overall goal of W7-X[3], is progress in the understanding of physics and engineering relevant to stellarator reactor-grade fusion plasmas, in particular relevant to Advanced Stellarators, continuing and augmenting W7-AS experience and knowledge. The configuration selected for W7-X will be of the Helias type (HELIcal Advanced Stellarator) which is a toroidal plasma equilibrium with appropriately optimized properties[11]. Added to the general potential advantages, W7-X has potential engineering advantages resulting from the possibility for only one single coil system for generating the entire confining magnetic field. Furthermore, experimentation with W7-X and comparison of the results with those of tokamaks will allow investigations on the influence on transport of net toroidal plasma currents and of the different behavior of trapped particle orbits so that W7-X may foster deeper understanding of toroidal magnetic confinement in general.

B. Basis for W7-X and selection of its magnetic configuration

This optimization of stellarator configurations was carried out with respect to the following set of criteria:

(1) high quality of vacuum-field magnetic surfaces (regular boundary, avoidance of low-order rational values of twist (or rotational transform) ι ('resonances'), adjustment of the shear, sufficiently small thickness of islands)

(2) good finite-β equilibrium properties (small shift of the magnetic axis [Shafranov shift], small change of ι with β at fixed external currents)

(3) good MHD stability properties (stability with respect to local resistive interchanges and ideal ballooning at $\bar{\beta} \geq 0.05$)

(4) small neoclassical transport in the $1/\nu$-regime (equivalent ripple $\delta e \leq 0.02$)

(5) small bootstrap current in the lmfp-regime(ratio of bootstrap current in a stellarator to the bootstrap current in a tokamak with same aspect ratio and rotational transform, $J_{BS,stel} \leq 0.1 J_{BS,tok}$)

(6) good collisionless α-particle containment at operational values of β (fractional prompt loss < 0.1)

(7) good modular coil feasibility (sufficiently large distance between coil and plasma, and a sufficiently small coil curvature)

The small axis shift and the small change of rotational transform and shear with increasing plasma pressure are a consequence of the achieved small parallel current density. A comparison of a convenient measure of the plasma currents, $< j_{\parallel}^2 / j_{\perp}^2 >$, between W7-AS and a Helias configuration shows a reduction (partly due to an increase of ι) by one order of magnitude. More generally, compatibility and simultaneous achievement of all of the above criteria has been proven; in particular, the goals concerning the neoclassical behavior have been surpassed.

While the optimization only takes into consideration classical physics goals it also results in very interesting and desirable perspectives as far as anomalous transport is concerned: many conjectured mechanisms for exciting anomalous transport(stochasticity in vacuum and finite-β fields, trapped orbits, instabilities such as ballooning, tearing and trapped-particle drift modes) are reduced substantially. In addition, the observed potential of stable operating a stellarator at unusually high densities, with the confinement time found to increase with density, is another route to increase confinement.

C. Description of W7-X

The W7-X is a modular Advanced Stellarator realizing a 5-period Helias configuration. A sketch of the configuration is displayed in Fig. 8.

In the standard configuration the rotational transform on axis is above 5/6 and approximately 1 at the boundary, thus providing larger shear than in W7-AS but still small enough to avoid major resonances in the confinement region. The equilibrium properties are characterized by strongly reduced Pfirsch-Schluter currents with the balancing currents less than 0.75 of the diamagnetic currents. MHD stability at $\bar{\beta} \geq 0.04$ is achieved by providing a vacuum field magnetic well and small Pfirsch-Schluter currents rather than by magnetic shear. Neoclassical transport is strongly reduced and characterized by an equivalent ripple $\delta_e \leq 0.015$. The residual bootstrap current is very small. In the boundary region, a separatrix and magnetic islands of corresponding topology can be arranged. A divertor concept based on the favorable geometrical properties of this region has been developed[12]; a sketch of this concept is shown in Fig. 9 and the appropriate experimental means are being developed. Experimental flexibility with respect to resonance as well as trapped-particle physics manifests itself in the possibility for variation of the rotational transform by ± 0.2, of the magnetic field up to 0.1. Major design data can be found in Table III and IV.

D. Technical realization of W7-X

The project has evolved to the state of an existing conceptual design being based on results of detailed study contracts launched in industry. These have led to the following solution: The confining magnetic field of W7-X will be generated by modular superconducting coils. The maximum magnetic field at the coils is sufficiently low to allow conventional NbTi superconductor technology, the operational reliability of which is demonstrated by many applications of superconducting magnets in various areas(e.g. accelerators, medical applications, fusion magnets).

The magnet consists of 50 non-planar coils with geometrical characteristics less demanding than those of W7-AS, 10 per field period, with 1.25m average radius. There are only five geometrically different coil types. A system of 20 superimposed planar coils, four per field period, allows variation of the rotational transform by ± 0.2. Separate current feeds(five in total)to the groups of ten equivalent coils allow modification of the mirror field along the magnetic axis by up to 0.1. Each coil is surrounded by a stainless steel housing, and the whole magnet is embedded in a cryostat. A sketch of the system is seen in Fig. 10. The total cold mass is about 350 tons. The refrigerator will work at a temperature of about 4K. This is in the same temperature range as had been chosen for the LCT-project. The experience gained with LCT is fully accounted for by the collaboration with KfK Karlsruhe. With a magnetic energy of about 0.6 GJ, the magnet system of W7-X is also comparable with that of the TF magnet of TORE SUPRA as shown in Fig.6.

Representative lengths of a superconducting cable (200m) have already been fabricated and a small solenoid with 0.2m inner radius-representing the smallest bending radii encountered in WENDELSTEIN 7-X-has been produced which is being tested in the STAR facility at KfK. A full-size demonstration coil has been ordered in industry at KfK. One sector of the vacuum and cryostat system has been designed and the corresponding call for tender has been placed in industry in the fall of 1994. The completion of these key components will result in a firm and derailed basis for manufacturing the whole magnet system.

IV. SUMMARY

The two types of next large helical devices, LHD (Large Helical Device) in NIFS-Nagoya, Japan and W7-X(Wendelstein 7-X) in IPP-Garching, Germany, are under design and construction. The LHD aims at clean helical divertor experiments using continuous helical coils. On the other hand, W7-X adopted the modular coil concept for the comprehensive optimization of plasma core confinement. The high-shear and medium-shear / medium-well plane-axis concepts were unified to the LHD configuration, while the shear-less helical system and helical-axis concept were combined to the W7-X configuration. In 1997, the LHD will start operation and while W7-X will produce a first plasma in the next decade.

In conclusion, we summarize in the followings:

(1) The recent current-less plasma experiments in the existing machines and simulation analyses using super computer are giving rise to better understanding of the physics of the helical confinement configurations.

(2) The next device of helical system is converged to the two concepts; the LHD with continuous helical coils and the W7-X with modular coils.

(3) These two programs are complementary to promote world helical fusion research and also extend the comprehensive understanding of toroidal plasmas by comparing with large tokamaks.

(4) The scenarios for the realization of helical reactors will be clarified by these two next large helical devices in the beginning of the 21st century.

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REFERENCES

1. A.Iiyoshi, M.Fujiwara, O.Motojima, N.Ohyabu, and K.Yamazaki, Fusion Technol., **17** (1990) 169;
2. O.Motojima, K.Akaishi, K.Fujii, S.Fujiwaka, S.Imagawa, H.Ji, H.Kaneko, S.Kitagawa, Y.Kubota, K.Matsuoka, T.Mito, S.Morimoto, A.Nishimura, K.Nishimura, N.Noda, I.Ohtake, N.Ohyabu, S.Okamura, A.Sagara, M.Sakamoto, S.Stoh, T.Satow, K.Takahata, H.Tamura, S.Tanahashi, T.Tsuzuki, S.Yamada, K.Yamazaki, N.Yanagi, H.Yonezu, J.Yamamoto, M.Fujiwara, and A.Iiyoshi, Fusion Engineering & Design **20** (1993) 3.
3. G.Grieger, C.D.Beidler, H.Massberg, E.Harmeyer, F.Herrnegger, J.Junker, J.Kisslinger, W.Lotz, P.Merkel, J.Nührenberg, F.Rau, J.Sapper, A.Schlüter, F.Sadei, and H.Wobig, *in Plasma Physics and Controlled Fusion Nuclear Research 1990* (International Atomic Energy Agency, Vienna, 1991), Vol.3, P525.
4. K.Yamazaki, N.Ohyabu, M.Okamoto, T.Amano, J. Todoroki, Y.Ogawa, N.Nakajima, H.Akao, M.Asao, J.Fujita, Y.Hamada, T.Hayashi, T.Kamimura, H.Kaneko, T.Kuroda, S.Morimoto, N.Noda, T.Obiki, H.Sanuki, T.Sato, T.Satow, M.Wakatani, T.Watanabe, J.Yamamoto, O.Motojima, M.Fujiwara, A.Iiyoshi, *in Plasma Physics and Controlled Fusion Nuclear Research 1990* (International Atomic Energy Agency, Vienna, 1991), Vol.2, P709.
5. T.Hayashi, A. Takei, and T.Sato, Phys. Fluids **4** (1992) 1539.
6. K. Ichiguchi, N.Nakajima, M.Okamoto, Y.Nakamura, and M.Wakatani, Nucl. Fusion **33** (1993) 481.
7. S.Sudo, Y.Takeiri, H.Zushi, F.Sano, K.Itoh, K.Kondo, and A.Iiyoshi, Nucl. Fusion **30** (1990) 11.
8. N. Ohyabu, T.Watanabe, H.Ji, H.Akao, T.Ono, T.Kawamura, K.Yamazaki, K.Akaishi, N.Inoue, A.Komori, Y.Kubota, N.Noda, A.Sagara, H.Suzuki, O.Motojima, M.Fujiwara, and A.Iiyoshi, Nucl. Fusion **34** (1994) 387.

9. J.Yamamoto, O.Motojima, T.Mito, K.Takahata, N.Yanagi, S.Yamada, H.Chikaraishi, S.Imagawa, A.Iwamoto, H.Kaneko, A.Nishimura, S.Satow, H.Tamura, S.Yamaguchi, K.Yamazaki, M.Fujiwara, and A.Iiyoshi, *in Proc. 15th International Conf. on Plasma Physics and Controlled Fusion Nuclear Research (Seville, Spain, 1994)* IAEA-CN-60/F-P-3.

10. A.Ando, K.Tsumori, Y.Oka, O.Kaneko, Y.Takeiri, E.Asano, T.Kawamoto, R.Akiyama, and T.Kuroda, Phys. Plasmas **1** (1994) 2813.

11. J.Nührenberg, R.Zille, Phys. Lett. A **114** (1986) 129; J.Nührenberg, R.Zille, Phys. Lett. A **129** (1988) 113.

12. J.Nührenberg, E.Strumberger, D.Sünder, and D.Reiter, *in Plasma Physics and Controlled Fusion Nuclear Research 1992* (International Atomic Energy Agency, Vienna, 1993), Vol.2, P449.

13. J.Sapper and W7-X Technical Group, Fusion Engineering & Design **20** (1993) 23.

Table I Major Machine and Plasma Parameters of LHD

	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

Table II Major Superconducting Coil Parameters of LHD

	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	$40 A/mm^2$	$53.3 A/mm^2$
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 III Characteristic Nondimensional Data of the Experiment W7-X

Rotational transform, ι, on axis/boundary	0.86/0.99
Variation of ι	± 0.2
Variation of shear	± 0.1
Variation of mirror field	0.1
Pfirsch-Schlüter currents, $\langle j_{\parallel}^2 / j_{\perp}^2 \rangle$	0.5
Magnetic well depth	0.01
MHD stability limit, $\langle \beta \rangle_{\mathrm{st}}$	0.043
Equivalent ripple, δ_{e}	0.015
Ratio of bootstrap currents, $J_{\mathrm{BS,stel}} / J_{\mathrm{BS,tok}}$	$\lesssim 0.1$

Table IV Characteristic Dimensional Data of the Experiment W7-X

Average major radius, R_0	5.5 m
Average plasma radius, r_a	0.53 m
Average coil radius, r_c	1.25 m
Min. distance plasma – coils, Δ_{pc}	0.30 m
Min. distance plasma – wall, Δ_{pw}	0.12 m
Induction on axis, B_0	3.0 T
Max. induction at coils, B_m	$\approx$ 6 T
Total magnetic energy, W_m	0.61 GJ
Max. net force (one coil), F_{res}	3.6 MN

FIGURE CAPTIONS

Fig.1 LHD basic plasma configuration by a pair of double helix coils (in yellow) and three pairs of poloidal coils (in green).

Fig.2 Schematic drawing of superconducting LHD machine.
Helical coils (in yellow) and poloidal coils (in red, orange and green) supported by the toroidal shell structure (in blue) and the plasma vacuum vessel (in gold) are shown in the cryostat (in light green).

Fig.3 Plasma parameters ,
(a) $n\tau_E T$ diagram, (b) long pulse operation diagram.

Fig.4 LHD confinement scaling τ_E^{LHD} vs. experimental confinement time τ_E^{exp}.

Fig.5 LHD helical divertor concept.

Fig.6 Progress of superconducting magnet construction and planning.

Fig.7 Photograph of LHD machine construction,
(a) Helical coil winding machine, (b) Inner vertical poloidal coil.

Fig.8 Sketch of W7-X basic configuration.
Shown are three and a half periods of the plasma boundary, the kidney-shaped cross-section at the position of strongest plasma-column curvature, the triangular cross-section at the position of smallest plasma-column curvature, one and a half periods of the helix-like magnetic axis (in red), a field line on the plasma boundary (in green), 35 of the 50 modular coils (in blue).

Fig. 9 Sketch of W7-X divertor concept.

Shown are the plasma boundary, one field line (in green) outside the separatix, five so-called helical troughs which cover the five helical edges which are the regions of field line diversion. The field line shown starts and ends on the plasma-facing side of the upper right helical trough. This concept avoids leading edges, exhibits small angles of incidence of the field lines on the troughs, and leads to a complete separation between the scrape-off layer field lines from those starting at the inner vacuum wall.

Fig.10 Sketch of W7-X technical components.

Shown are the inner and outer cryostat wall (in brown), the modular coil set (in dark blue), the auxiliary coil set for experimental flexibility (in light blue), and various ports(in green).

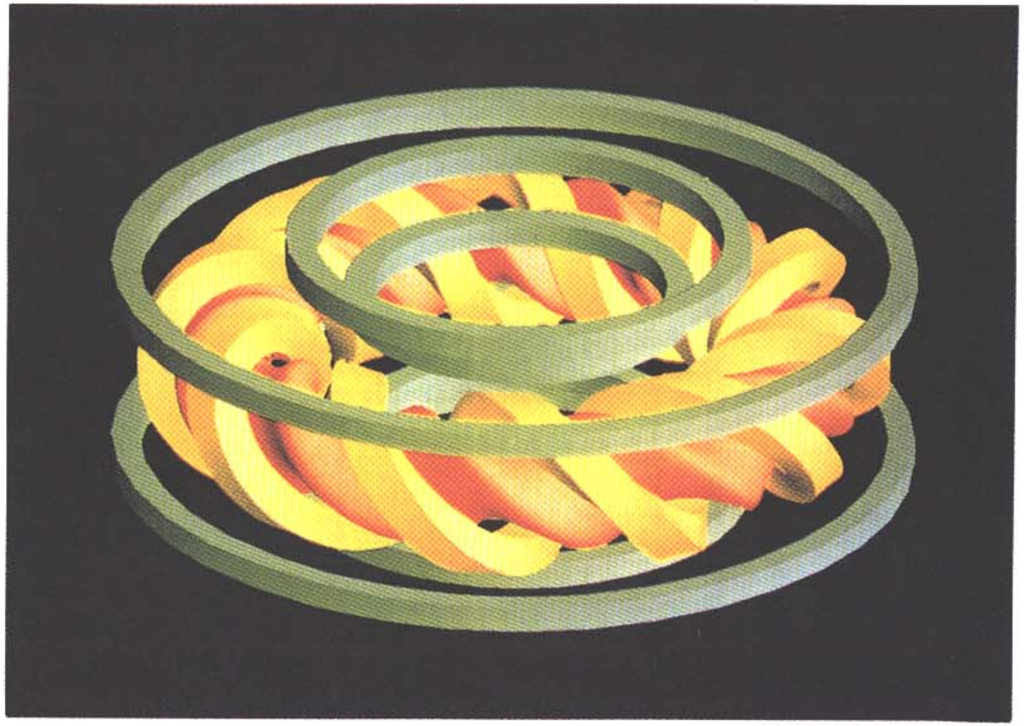


Fig. 1 A.Iiyoshi et al.

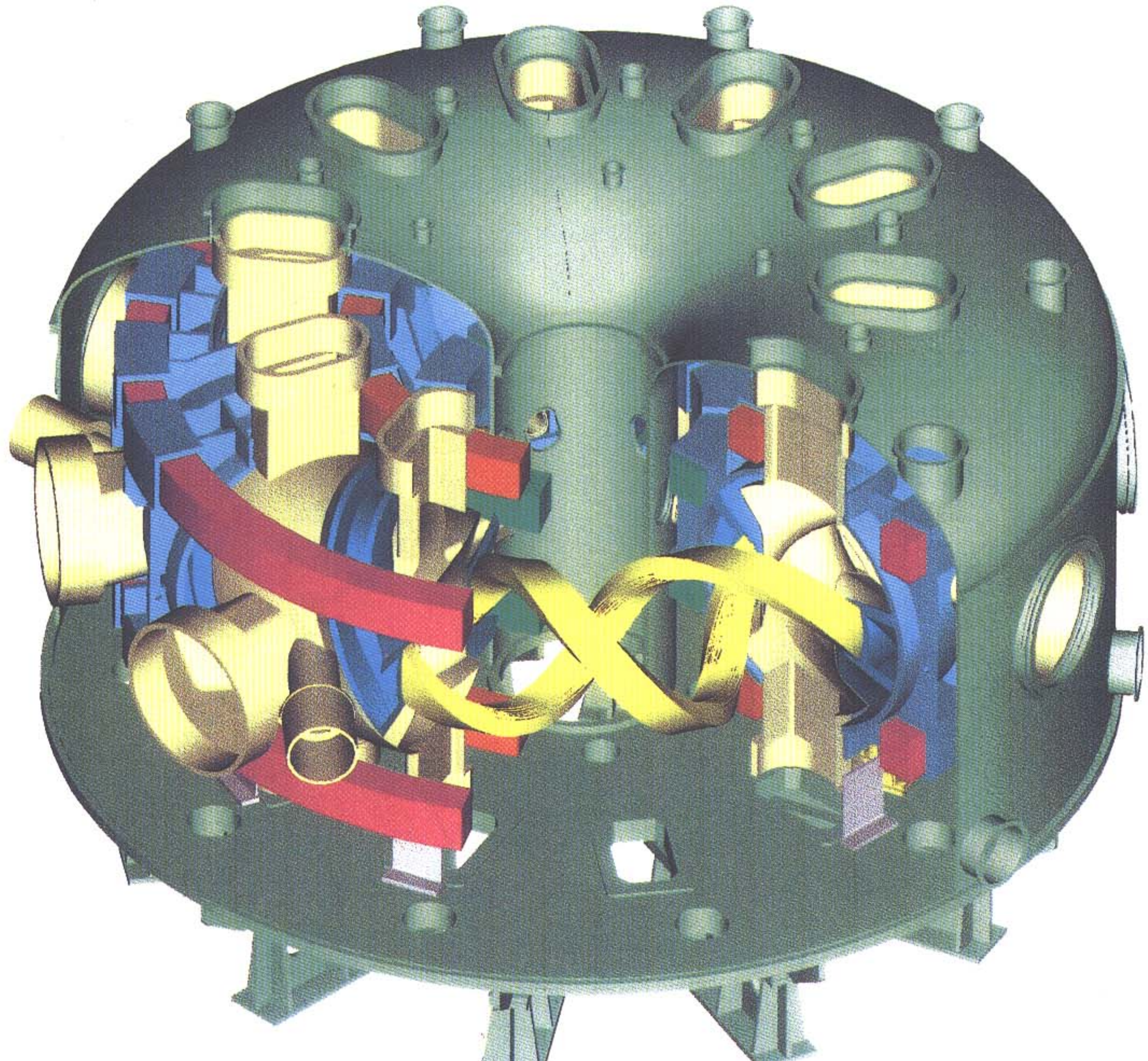


Fig. 2 A.Iiyoshi et al.

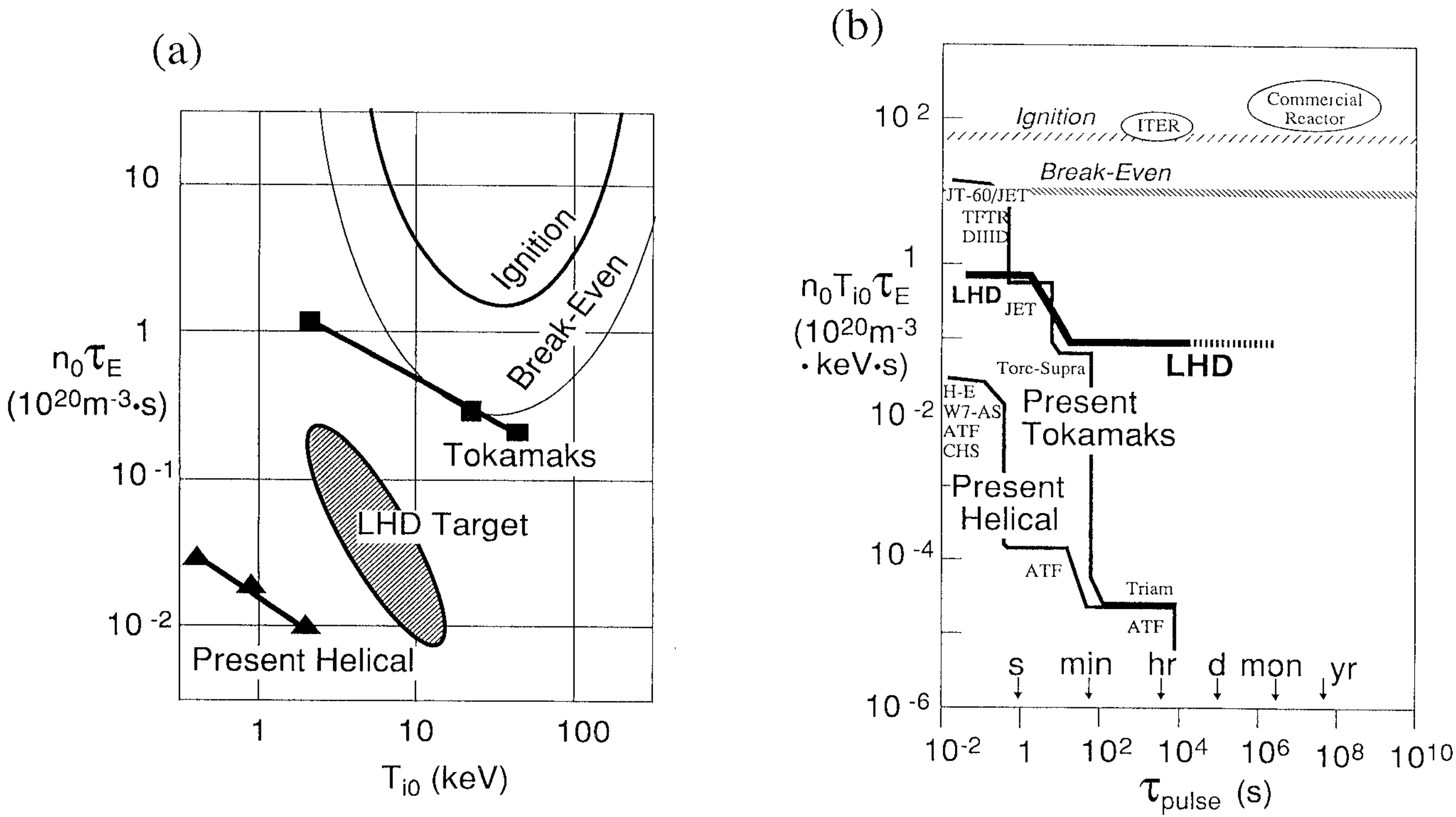
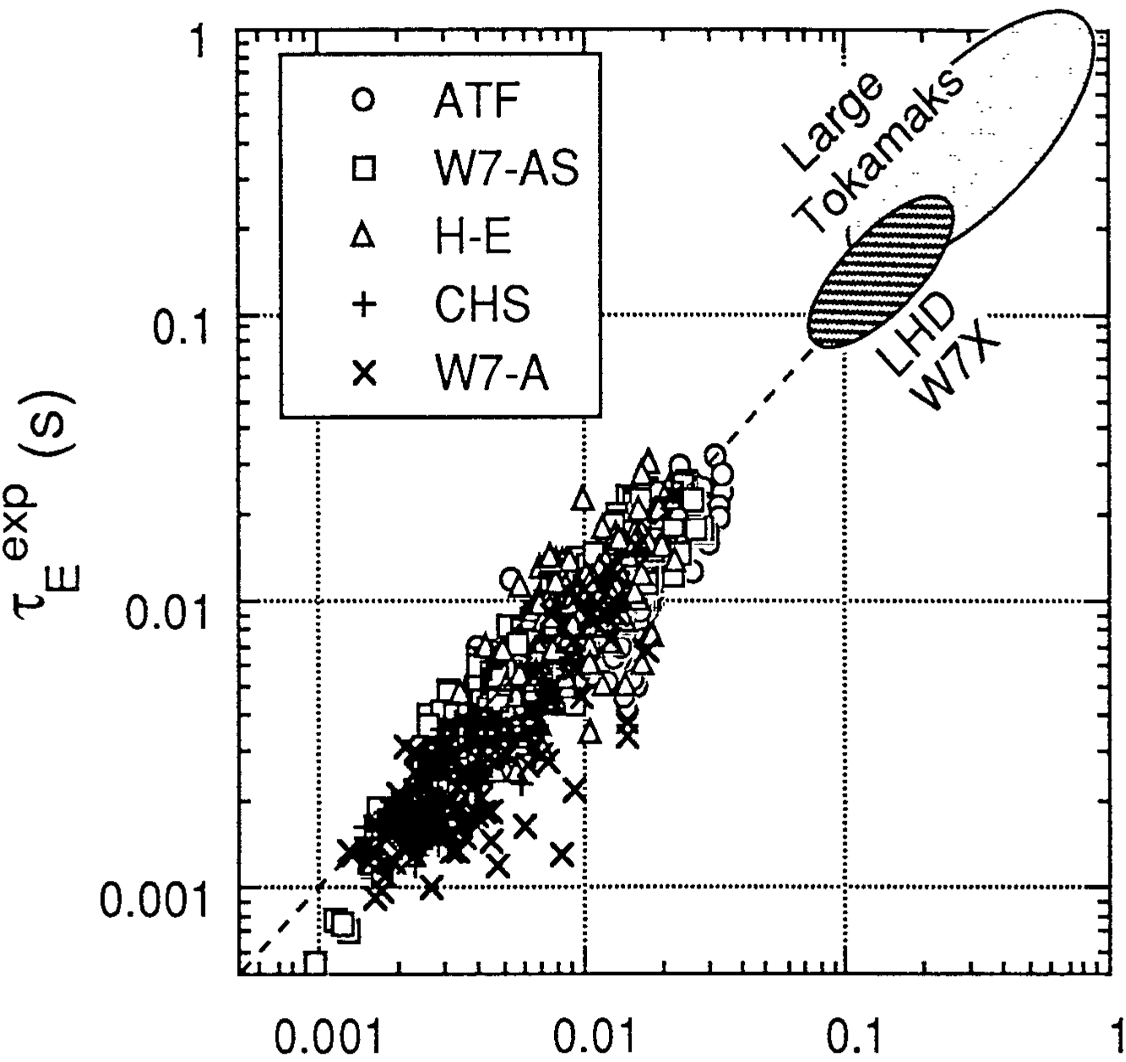


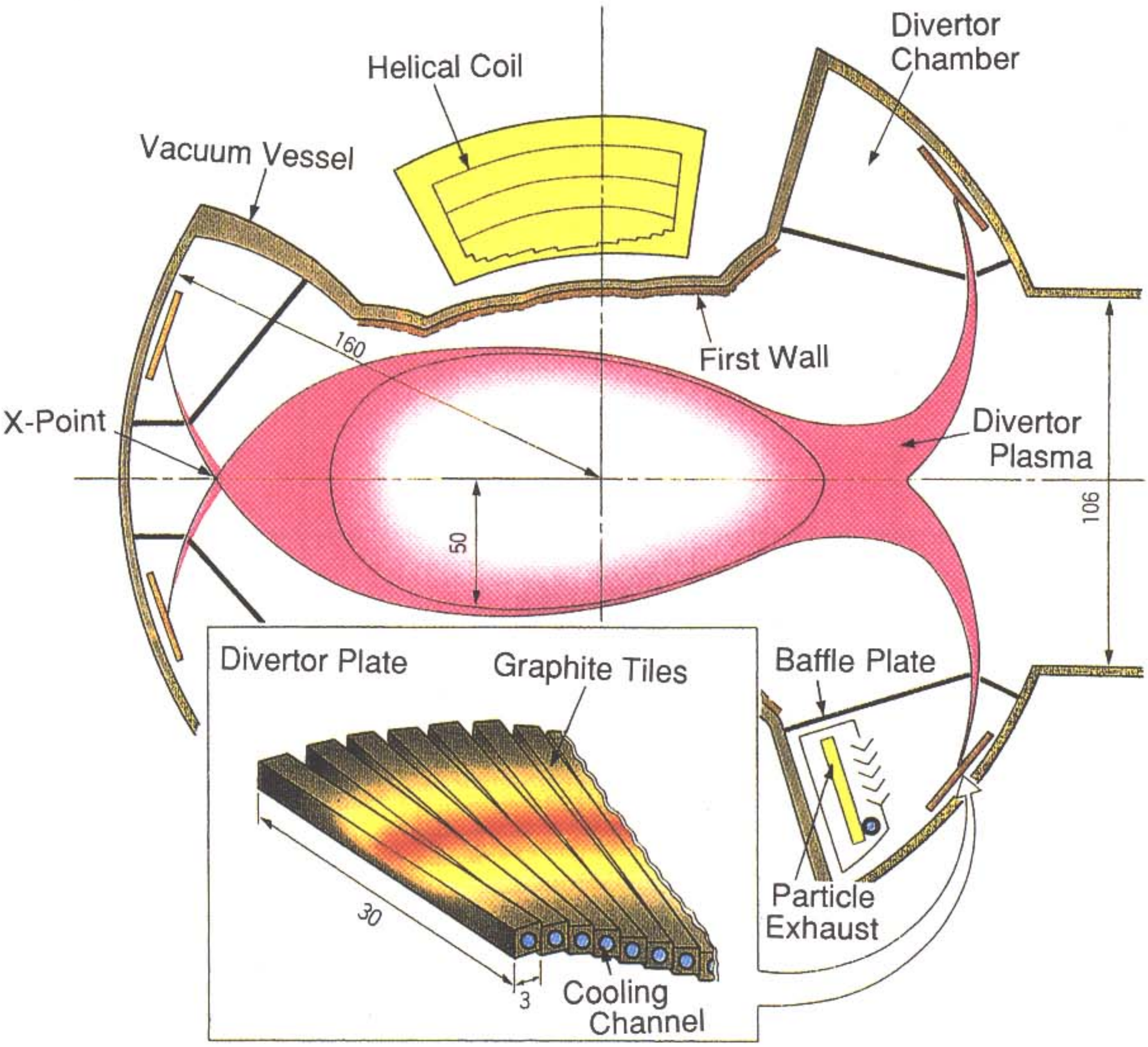
Fig. 3 A.Iiyoshi et al.



$$\tau_E^{LHD} = 0.17\ P_{abs}^{-0.58}\ \bar{n}_e^{\ 0.69}\ B^{0.84}\ a^2\ R^{0.75}$$

$$(s)\qquad (MW)\quad (10^{20} m^{-3})\ (T)\quad (m)\ (m)$$

Fig. 4 A.Iiyoshi et al.



LHD Helical Divertor

Fig. 5 A.Iiyoshi et al.

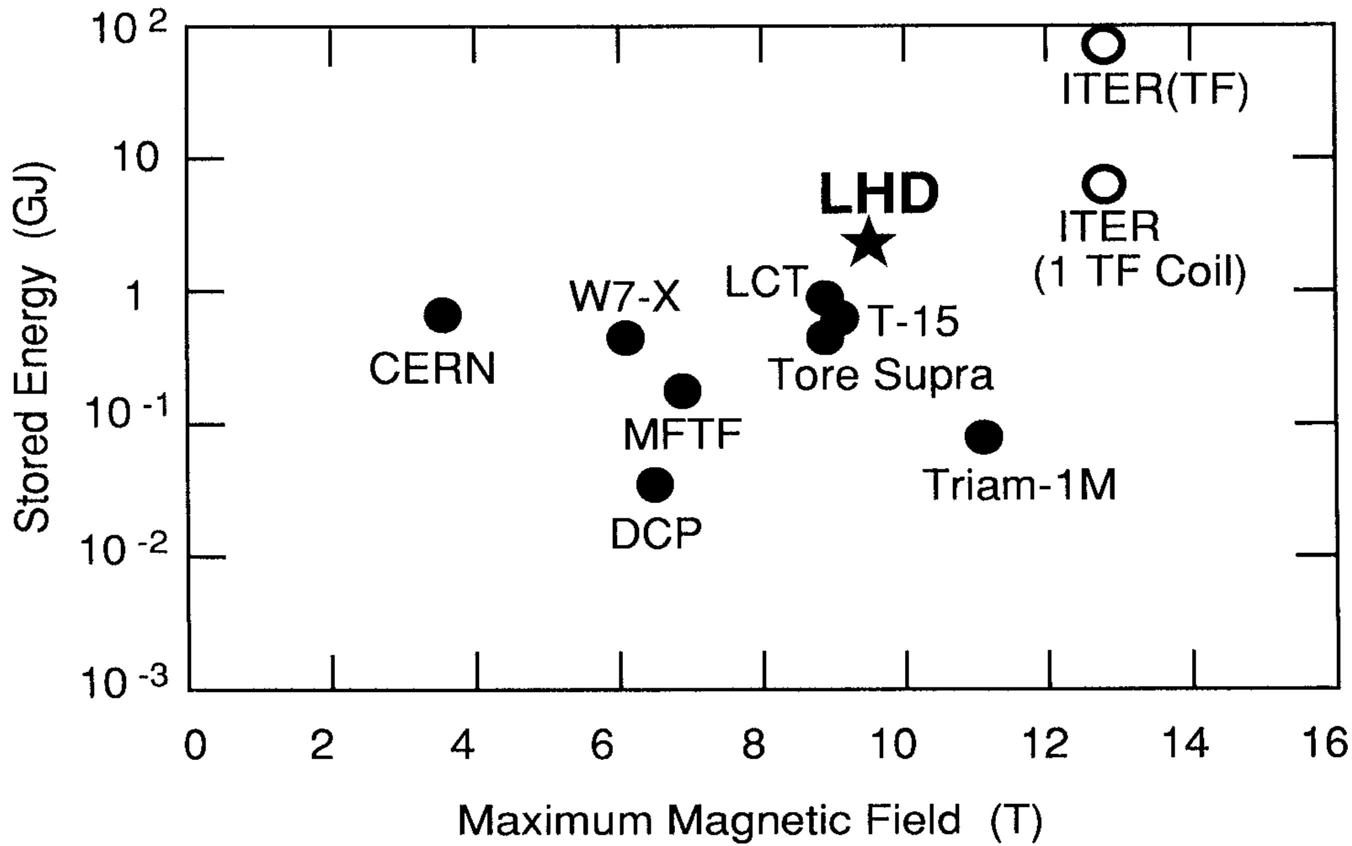
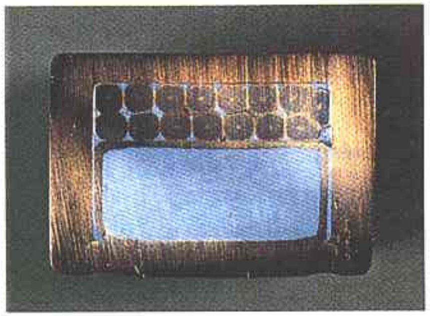
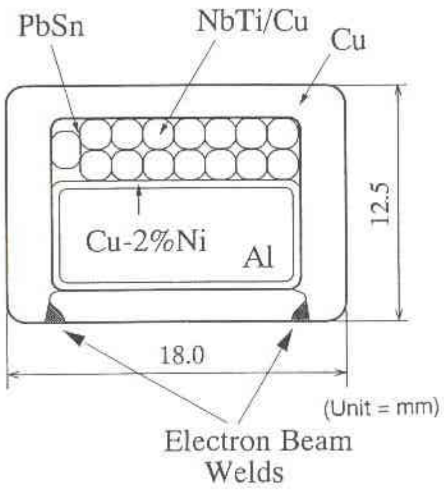


Fig. 6 A.Iiyoshi et al.

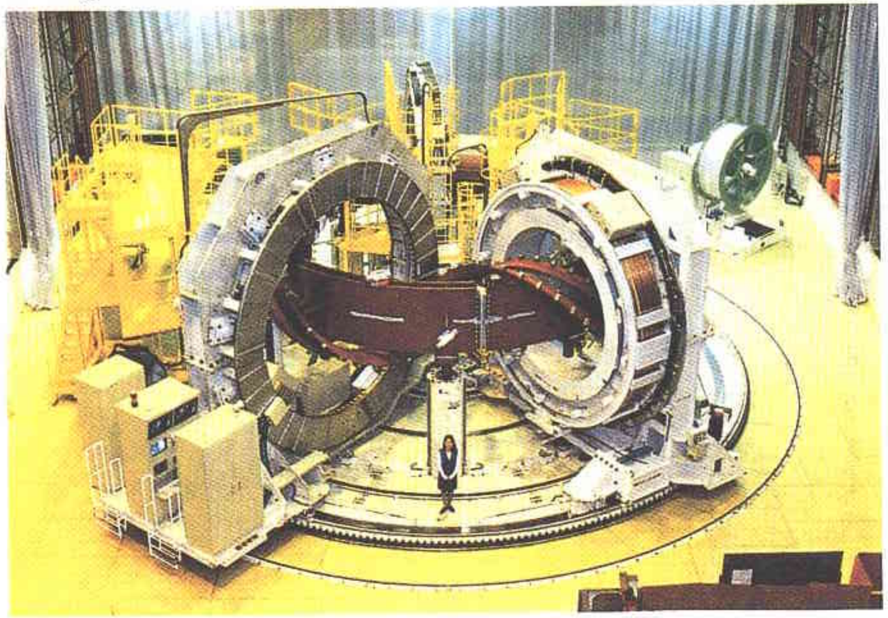
(a)

Cross-section of the superconductor (KISO-32) for the helical coils



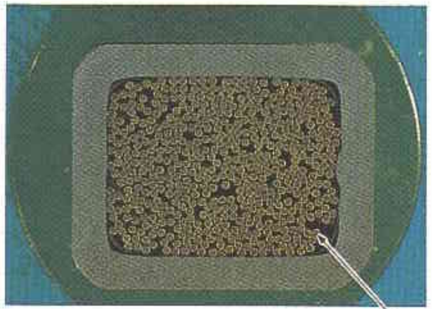


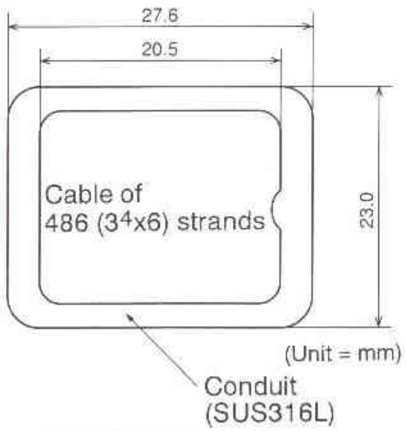
Winding machine for the helical coils

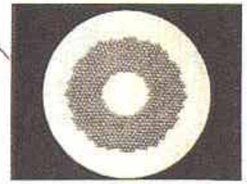


(b)

Photograph of the cross-section of the IV conductor

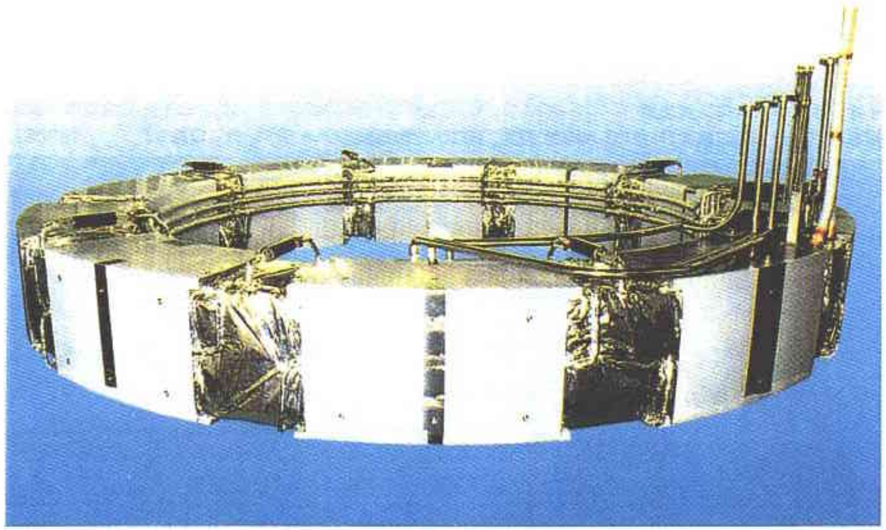






Cross-section of the NbTi/Cu strand

Inner Vertical (IV) coil



Outer radius : ~4 m Hight : ~0.4 m Weight : ~16 t
Operating current : ~20kA Maximum field : 6.5 T

Fig. 7 A.Iiyoshi et al.

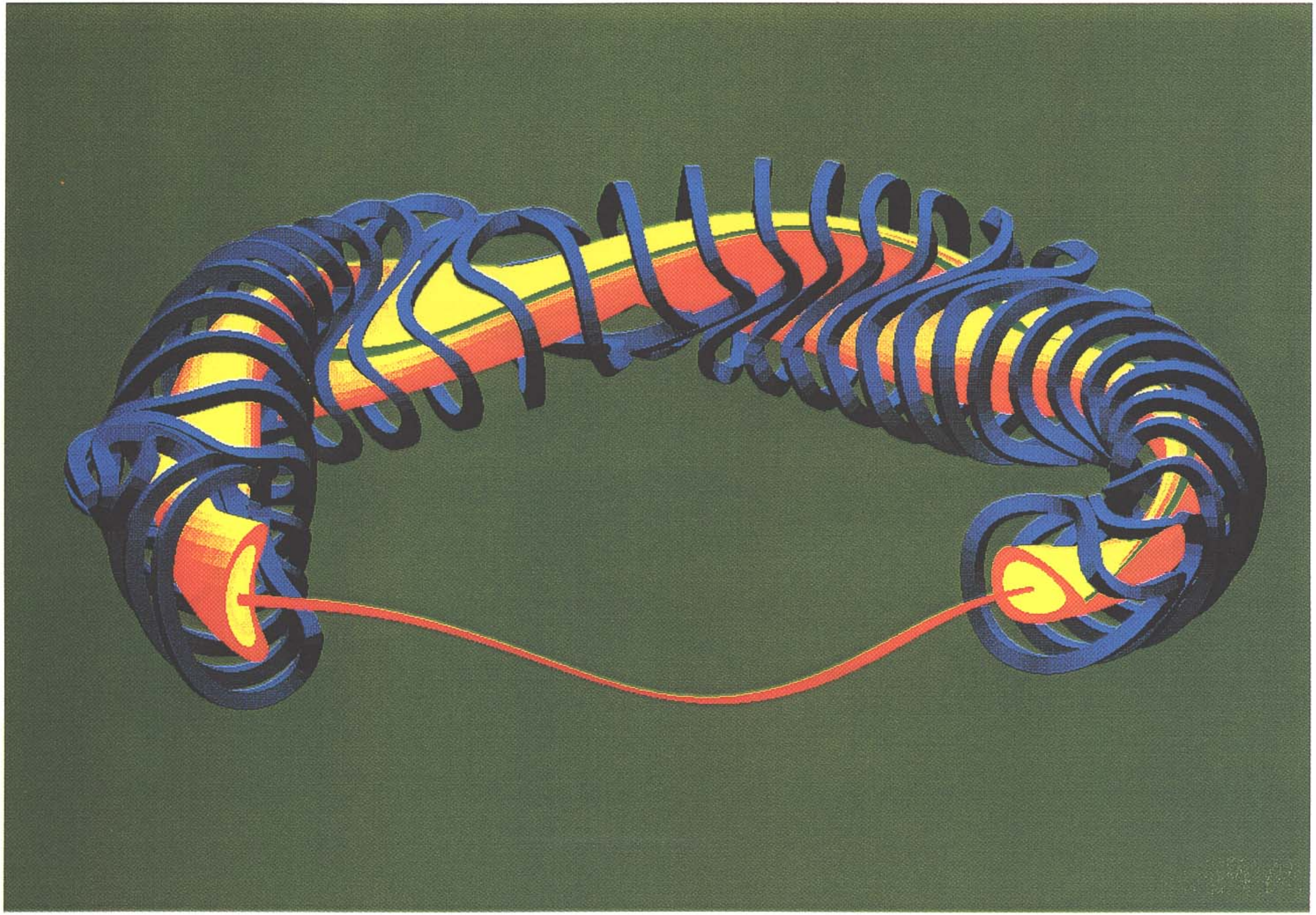


Fig. 8 A.Iiyoshi et al.

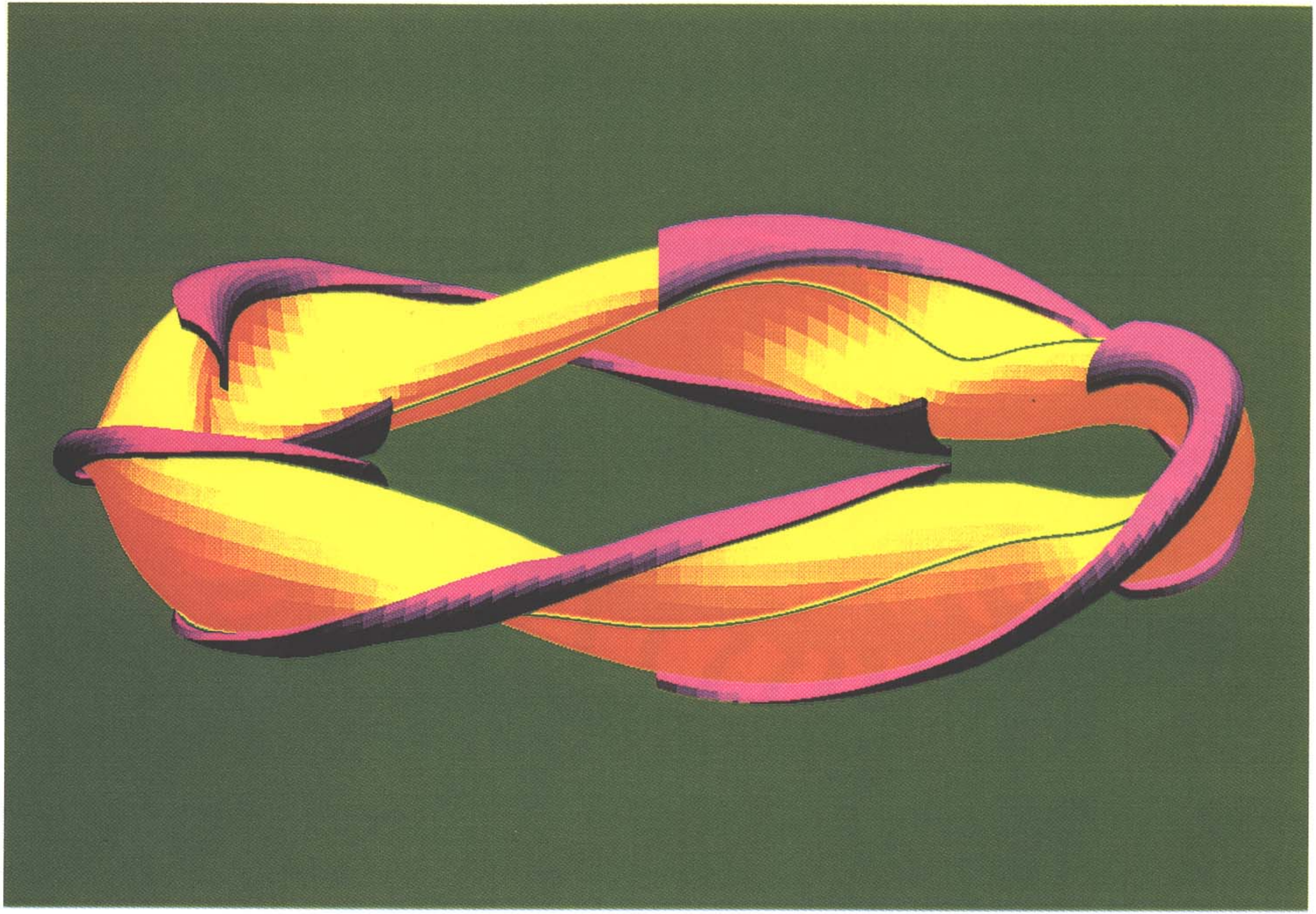


Fig. 9 A.Iiyoshi et al.

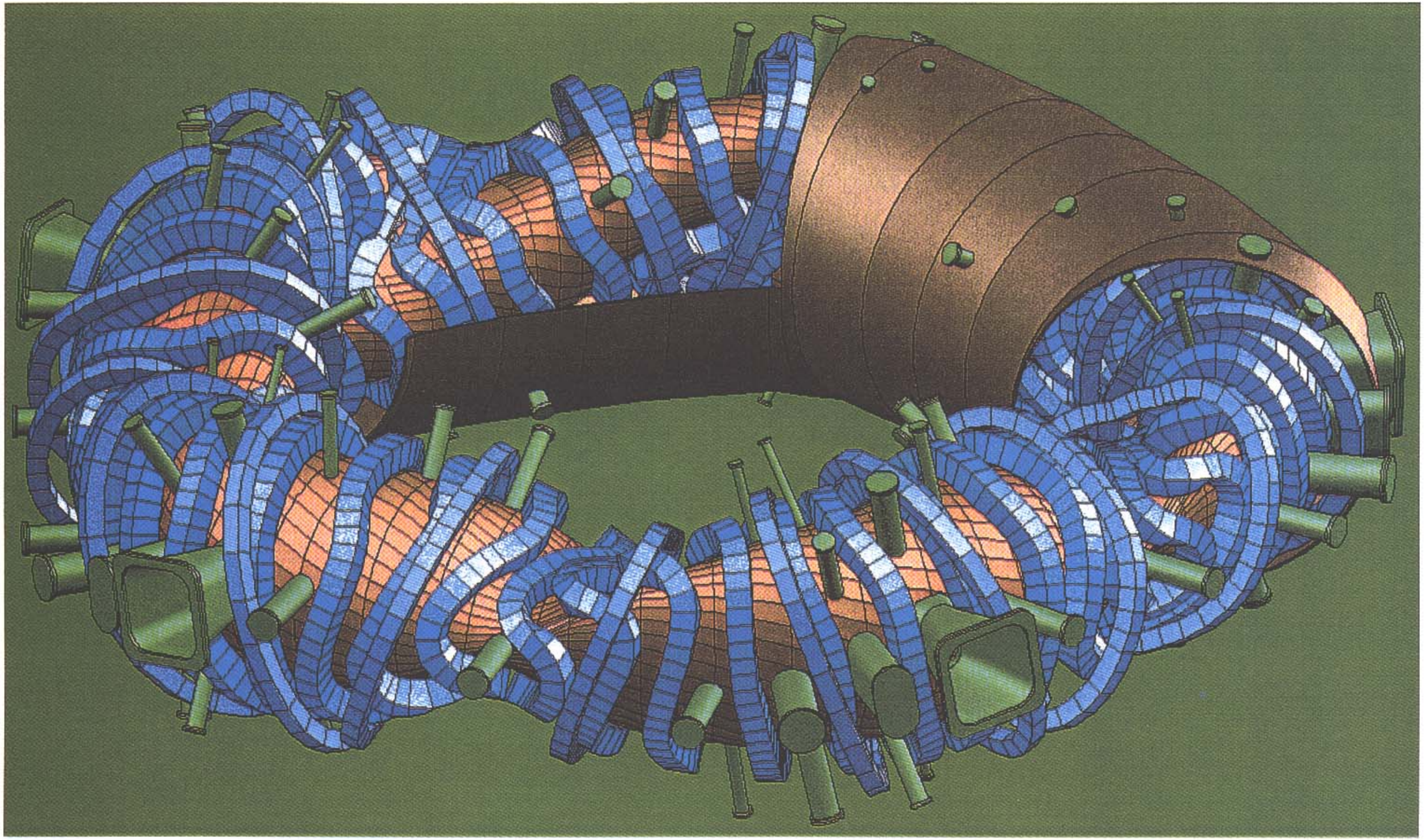


Fig. 10 A.Iiyoshi et al.

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Innovative Divertor Concepts for LHD; July 1994

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Formation of Positive Radial Electric Field by Electron Cyclotron Heating in Compact Helical System; July 1994

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Boronization Study for Application to Large Helical Device; July 1994

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Effects of Impurities Released from High Z Test Limiter on Plasma Performance in TEXTOR; July. 1994

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Experimental Study on Scaling Law of Outgassing Rate with A Pumping Parameter, Aug. 1994

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Magnetic Mirror Effect as a Trigger of Collisionless Magnetic Reconnection, Aug. 1994

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Anomalous Transport Theory for Toroidal Helical Plasmas,
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New Evaluation Method of Superconductor Characteristics for Realizing the Large Helical Device; Aug. 1994 (IAEA-CN-60/F-P-3)

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Local Island Divertor Concept for LHD; Aug. 1994 (IAEA-CN-60/F-P-4)

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Impact of Rotational-Transform Profile Control on Plasma Confinement and Stability in CHS; Aug. 1994 (IAEA-CN-60/A6/C-P-3)

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Dynamical Model of Pressure-Gradient-Driven Turbulence and Shear Flow Generation in L-H Transition; Aug. 1994 (IAEA/CN-60/D-P-I-11)

NIFS-301 Y. Hamada, A. Nishizawa, Y. Kawasumi, K.N. Sato, H. Sakakita, R. Liang, K. Kawahata, A. Ejiri, K. Narihara, K. Sato, T. Seki, K. Toi, K. Itoh, H. Iguchi, A. Fujisawa, K. Adachi, S. Hidekuma, S. Hirokura, K. Ida, M. Kojima, J. Koog, R. Kumazawa, H. Kuramoto, T. Minami, I. Negi, S. Ohdachi, M. Sasao, T. Tsuzuki, J. Xu, I. Yamada, T. Watari,
Study of Turbulence and Plasma Potential in JIPP T-IIU Tokamak;
Aug. 1994 (IAEA/CN-60/A-2-III-5)

NIFS-302 K. Nishimura, R. Kumazawa, T. Mutoh, T. Watari, T. Seki, A. Ando,

S. Masuda, F. Shinpo, S. Murakami, S. Okamura, H. Yamada, K. Matsuoka, S. Morita, T. Ozaki, K. Ida, H. Iguchi, I. Yamada, A. Ejiri, H. Idei, S. Muto, K. Tanaka, J. Xu, R. Akiyama, H. Arimoto, M. Isobe, M. Iwase, O. Kaneko, S. Kubo, T. Kawamoto, A. Lazaros, T. Morisaki, S. Sakakibara, Y. Takita, C. Takahashi and K. Tsumori,
ICRF Heating in CHS; Sep. 1994 (IAEA-CN-60/A-6-I-4)

NIFS-303 S. Okamura, K. Matsuoka, K. Nishimura, K. Tsumori, R. Akiyama, S. Sakakibara, H. Yamada, S. Morita, T. Morisaki, N. Nakajima, K. Tanaka, J. Xu, K. Ida, H. Iguchi, A. Lazaros, T. Ozaki, H. Arimoto, A. Ejiri, M. Fujiwara, H. Idei, A. Iiyoshi, O, Kaneko, K. Kawahata, T. Kawamoto, S. Kubo, T. Kuroda, O. Motojima, V.D. Pustovitov, A. Sagara, C. Takahashi, K. Toi and I. Yamada,
High Beta Experiments in CHS; Sep. 1994 (IAEA-CN-60/A-2-IV-3)

NIFS-304 K. Ida, H. Idei, H. Sanuki, K. Itoh, J. Xu, S. Hidekuma, K. Kondo, A. Sahara, H. Zushi, S.-I. Itoh, A. Fukuyama, K. Adati, R. Akiyama, S. Bessho, A. Ejiri, A. Fujisawa, M. Fujiwara, Y. Hamada, S. Hirokura, H. Iguchi, O. Kaneko, K. Kawahata, Y. Kawasumi, M. Kojima, S. Kubo, H. Kuramoto, A. Lazaros, R. Liang, K. Matsuoka, T. Minami, T. Mizuuchi, T. Morisaki, S. Morita, K. Nagasaki, K. Narihara, K. Nishimura, A. Nishizawa, T. Obiki, H. Okada, S. Okamura, T. Ozaki, S. Sakakibara, H. Sakakita, A. Sagara, F. Sano, M. Sasao, K. Sato, K.N. Sato, T. Saeki, S. Sudo, C. Takahashi, K. Tanaka, K. Tsumori, H.Yamada, I. Yamada, Y. Takita, T. Tuzuki, K. Toi and T. Watari,
Control of Radial Electric Field in Torus Plasma; Sep. 1994 (IAEA-CN-60/A-2-IV-2)

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Behavior of Magnetic Islands in 3D MHD Equilibria of Helical Devices; Sep. 1994 (IAEA-CN-60/D-2-II-4)

NIFS-306 S. Murakami, M. Okamoto, N. Nakajima, K.Y. Watanabe, T. Watari, T. Mutoh, R. Kumazawa and T. Seki,
Monte Carlo Simulation for ICRF Heating in Heliotron/Torsatrons; Sep. 1994 (IAEA-CN-60/D-P-I-14)

NIFS-307 Y. Takeiri, A. Ando, O. Kaneko, Y. Oka, K. Tsumori, R. Akiyama, E. Asano, T. Kawamoto, T. Kuroda, M. Tanaka and H. Kawakami,
Development of an Intense Negative Hydrogen Ion Source with a Wide-Range of External Magnetic Filter Field; Sep. 1994

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Evolution of Magnetic Islands in a Heliac; Sep. 1994

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Intermittent Energy Bursts and Recurrent Topological Change of a

Twisting Magnetic Flux Tube; Sep.1994

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Effect of Anomalous Plasma Transport on Radial Electric Field in Torsatron/Heliotron; Sep. 1994

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Current-profile Flattening and Hot Core Shift due to the Nonlinear Development of Resistive Kink Mode; Oct. 1994

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Modification and Damping of Alfvén Waves in a Magnetized Dusty Plasma; Oct. 1994

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Physical Mechanism Determining the Radial Electric Field and its Radial Structure in a Toroidal Plasma; Oct. 1994

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Non-Taylor Magnetohydrodynamic Self-Organization; Oct. 1994

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Collisionless Magnetic Reconnection Associated with Coalescence of Flux Bundles; Nov. 1994

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Macro-EM Particle Simulation Method and A Study of Collisionless Magnetic Reconnection; Nov. 1994

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Second Order Focusing Property of 210° Cylindrical Energy Analyzer; Nov. 1994

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Complexity in Plasma - A Grand View of Self- Organization; Nov. 1994

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MHD-Vlasov Simulation of the Toroidal Alfvén Eigenmode; Nov. 1994

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Computer Simulation of a Magnetohydrodynamic Dynamo II: Nov. 1994

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Theory of Pressure-induced Islands and Self-healing in Three-dimensional Toroidal Magnetohydrodynamic Equilibria; Nov. 1994

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Recent Studies of the Large Helical Device; Nov. 1994