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Heliotron Research

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ABSTRACT

This "Festvortrang" was presented by the author in the special colloquium in honor of Dr. G. Grieger's 60th birthday, held in Max-Planck-Institut fur Plasmaphysik on March 18, 1991.

The author reviewed the history of the development of the stellarator/heliotron system, and pointed out the important role of the radial electric field in plasma transport in helical devices.

After reviewing the history, the author expects that the 1990s will be a decade of new significant developments for helical fusion research.

key words:
stellarator, heliotron, historical review, Wendelstein device, Large Helical Device, radial electric field

Development of The Stellarator / Heliotron Research

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It is my great honor and pleasure to give a "Festvortrag" titled development of the stellarator/heliotron research in honor of Dr. Grieger's 60th birthday. For many years, Dr. Grieger has been a friend of mine and also a friendly rival in developing stellarator systems. I always have a great respect for Dr. Grieger as a great scientist and leader in the fusion community. I would like to congratulate him on his 60th birthday and his many contributions in fusion research, particularly in stellarator research. As you know, Dr. Grieger has achieved many scientific accomplishments, and he has been not only the leader of the Wendelstein project, which started in the late 1950s, but also a leader of the stellarator research community in the world. The series of Wendelstein experiments have been strong vehicles, which have provided momentum and strong motivations to stellarator research. Today I would like to review the history of the development of the stellarator system, in which the Wendelstein project played a major part. Because of limited time, I will focus on the highlights of the history.

The concept of the stellarator originated from the 1950s. In 1951, L. Spitzer invented the stellarator. Many basic physics concepts, associated with toroidal confinement, such as Magnetic Surface, Rotational Transform, Magnetic Shear, Magnetic Well, Separatrix, Divertor emerged in those days.

Stellarator is the most generalized form of the toroidal confinement system. Even the tokamak, the most developed confinement system, can be regarded as a kind of stellarator system with axisymmetry. Because of axisymmetry, a tokamak needs the current flowing through the plasma in order to have the rotational transform which is essential for toroidal confinement. On the other hand, most stellarators are characterized by a currentless plasma since the confining fields are generated by the external coils. Spitzer first invented the Figure 8 stellarator (Stellarator A, B) and tested it experimentally. But it was unsuccessful. At that time E. Tellar pointed out the danger of the MHD instability. Magnetic shear was believed to be effective in suppressing the MHD instability. Therefore a stellarator with shear was proposed by Spitzer. This stellarator had helical winding coils and was called C-stellarator. Since then, stellarators with L=2 (shearless) or L=3 (high shear) helical winding coils in addition to the toroidal coils, became standard. After the Geneva Conference, held in 1958, fusion research became open to the scientists in many countries. Immediately, the Max-Planck-Institute constructed a device named Wendelstein I-A and started stellarator research, and continued with WI-B (l = 2, racetrack, Fig.1). Several countries also started research by constructing e.g. the L1 device in the USSR, or Clasp in the UK. The experiments in these devices were the major research efforts in the 1960s. In this period, there existed a great controversy between Munich and Princeton. In the C-stellarator experiment at Princeton Laboratory, the energy confinement of the ohmically heated plasmas could not exceed lmsec, suffering a pump-out phenomenon, so called Bohm diffusion. On the other hand, Pfirsh-Schlüter transport, i.e. classical resistive transport was observed in the currentless Cs plasma experiment by the

Max-Planck-Institute. Subsequently, a similar result was obtained in the Wendelstein II-A (l=2, circular) experiment with Ba plasma. These are the first demonstrations that plasma was confined classically in a real toroidal device. In the middle of 1960s, I happened to work in the C-stellarator experiment at Princeton. I recall that Princeton researchers in those days were wondering about the Munich mystery, and Princeton had sent Dr. Stodiek to Munich to check the Munich experimental results. At that time, Russian tokamak experiments showed promising results and the C-stellarator was converted into the ST tokamak in 1969. This was a very unfortunate setback for stellarator research. Ironically, breakup of the magnetic surfaces was found in the C-stellarator configuration by diagnostics of the vacuum magnetic surface just before its shutdown. It is now believed that flux surface breakup was the major cause for the pump-out phenomenon in the C-stellarator. It was attributed to inaccurate manufacturing of the coils.

Despite the unsuccessful experiments and shutdown of the C-stellarator, stellarator experiments were continued in the world, encouraged by the Wendelstein observation of classical transport. Those experiments were WII-B (l=2, circular with OH transformer in Garching, Fig.2), URGAN for USSR, JIPP-I for Japan, Proto-Cleo for UK. Later Heliotron D (Fig.3) joined the stellarator club as the first Heliotron/Torsatron field experiment. The Heliotron/Torsatron system is simpler than the standard stellarator because it could do without toroidal coils and used just half the number of helical windings, and it has a built-in divertor in the free space between the helical windings.

After learning the C-stellarator lesson, it has become customary to confirm the integrity of the vacuum magnetic surfaces before the start of new stellarator experiments.

The experiments in these devices utilize a plasma gun or low power ECRH heating to establish and maintain the hydrogen plasma, and they demonstrated that the observed transport agreed with the neo-classical transport within a factor of ten in both the Pfirsh-Schlüter regime and the Galeev-Sagdeev plateau regime.

However, from the late 1960s to early 1970s, several theoretical papers were published, which predicted very pessimistic transport in the future stellarator devices. For the plasma in future device, the temperature is high and plasma becomes collisionless. In such conditions a fraction of the plasma particles are trapped locally by the local helical ripples so that they do not circulate around the magnetic surface and instead they escape from the plasma interior by the toroidal drift. Therefore the plasma transport is greatly enhanced. This is called "Superbanana transport". If true, prospects for a stellarator reactor system become very dim. In those days, the importance of the radial electric field was not recognized so that this superbanana transport was a very serious concern. Two scenarios or schemes were suggested to minimize the superbanana transport. One is to reduce the helical ripple (even though it tends to reduce the rotational transform) and to increase the aspect ratio. This was the basic design philosophy of Wendelstein 7A (Fig.4). The other scenario is to operate in the high density regime which is close to Galeev-Sagdeev regime. This, in turn requires a high-β plasma. This was the design philosophy of Heliotron E (Fig.5) whose configurational characteristics are high rotational transform and strong shear. These two devices, W-7A, H-E were constructed in the late 1970s, aiming at proof of principle experiments in the helical system.

In the 1980s, these devices including ATF in USA generated many good and encouraging results. The major highlights of the experiments are successful heating of currentless helical plasma by NBI for W7-A and by high power ECRH heating for Heliotron E. In W7A, plasma with Ti=1 KeV, n $=10^{14}cm^{-3}$, $\tau_E=15$ msec was achieved despite a small plasma radius of 10cm. This result really surprised the fusion community. More importantly, it demonstrated that the radial electric field did improve the transport substantially. Perpendicular NBI injection was by far more effective than expected. The reason for this is that the radial electric field, generated by loss of the high energy particles near the boundary reduces ion heat conduction.This experimental result motivated theorists to study the effect of the radial electric field on the transport. Subsequent theoretical study also revealed that the radial electric field, naturally created to satisfy the ambipolar diffusion condition, plays a crucial role in confinement in the helical system, especially in the collisionless regime. Shaing & Callen (1983) made a detailed and systematic analysis of the neoclassical transport. They found that in some conditions multiple solutions of Er which satisfy the ambipolar condition. The solution of positive Er is called electron root and the negative Er solution is called ion root. When the positive electric field is large, namely, the plasma is deeply in the electron root regime, the superbanana diffusion is significantly reduced and also impurities are expected to be impeded.

The Wendelstein experimental results did not agree with the theory since the electric field was produced mainly by the fast ion losses. Nonetheless the experimental demonstration of a favorable influence of the radial electric field on the plasma transport is an important step in the history of stellarator research. In the Heliotron E experiment with a combination of ECRH and ICRF heating, the observed increase in the central ion temperature and the decrease in the impurity concentration appear to be due to a transition from the ion root to the electron root. From these studies, I strongly believe that the radial electric field plays a key role in plasma transport in helical devices. In order to understand the role of the electric field more clearly, direct measurements of the radial electric field and its fluctuations by a heavy ion beam probe, and charge exchange recombination spectroscopy needs to be done.

Recently the effect of the radial electric field on the transport has been argued to be a key element in the so-called L-H transition in Tokamak discharges. Theorists such as Shaing and the Itohs, who have been involved with stellarator theory have made great contributions to H-mode transition theory.

Encouraging results of W7-A naturally lead to the next step, the Advanced Stellarator W7-AS (Fig.6), which started construction in 1980. The meaning of advanced are twofold. One is to minimize the Shafranov shift of the magnetic axis which occurs when β increases by nearly eliminating the Pfirsh-Schlüter current. The other is to minimize the particle orbit loss. Such physics requirements are found to be met in a device with a modular coil system. It is worthwhile to mention that twisted modular coil system was invented in Garching. The Advanced Stellarator experiment started in 1986 and the measured magnetic surfaces were perfect as expected from the numerical calculation, which demonstrated practicability of the modular coil system. After the initial experiments, the experiments with ECRH heating and NBI heating are being continued, aiming at improvement of the plasma parameters such as the temperature and better understanding of the plasma confinement mechanism. The experiments demonstrated clearly the reduction of the

Shafranov shift. It has been also made clear that the bootstrap current must be controlled well in the weak shear systems. So far the W7-AS experiment has acheived a record of Te~3kev. High density discharges up to $3\times10^{14}cm^{-3}$, were obtained demonstrating that the density limit is less severe compared with tokamak discharges. This is an important implication in achieving higher plasma performance since there exists a favorable density dependence of the confinement scaling. This graph (Fig.7) compares the confinement results from the various devices. This empirical scaling used is called LHD scaling. Data points of W7-AS are highest in absolute value together with those of Heliotron E. They are around the line predicted from previous data of other devices. And also, the confinement in helical devices appears almost the same as those of tokamaks (L-mode scaling) with a similar dimension. The global energy confinement is governed by anomalous transport which dominates at the periphery of the plasma. The causes of the anomalous effects have not been resolved. I'd like to mention another important point. The confinement times are around a line for devices with a wide range of aspect ratio from 5 to 20. The major feature of CHS (Fig.8), our operating device, is the low aspect ratio. It is as low as 5. (The CHS data points are similar to those with high aspect ratio.) This allows a lower aspect ratio device, and therefore a smaller device without sacrificing energy confinement.

Now I'll talk about our future. We expect that the 1990s will be a decade of new significant developments for helical fusion research. Namely, W7-X (Fig.9, Fig.10) in Germany and LHD in Japan will be constructed and they will demonstrate that the helical confinement system is a viable option for a fusion reactor.

Let me talk a little bit about our LHD (the Large

Helical Device) project in the National Institute for Fusion Science (NIFS). NIFS was established in 1989 as the central national institute among the universities in Japan. Since then, Dr. Grieger is a member of the Board of Councilors of our institute and has given us valuable advice. The Institute of Plasma Physics within Nagoya university was abolished, and was reorganized as NIFS, being combined together mainly with the Fusion Theory Center of Hiroshima university and a part of heliotron groups of Kyoto university. The major efforts of NIFS are the LHD project and the simulation study of the fusion plasma. Similarly as W7-X, LHD aims at achieving fusion relevant parameters and employs superconducting magnets. The machine parameters are listed in the Table. This is a schematic overview of the LHD (Fig. 11). LHD contains a built-in divertor. W7-X, I understand, is currently under the European Phase-I examination. The LHD has been already approved by the government in 1990 and construction will be completed in 1997. The NIFS is located at Toki city, Gifu prefecture (about 30km from Nagoya). The first building for the superconducting test facility has already been completed last fall, and R&D test studies for the LHD magnet have begun.

These two projects complement each other since LHD is the optimized device with continuous helical winding coils, and W7X is the optimized device with modular coils (Fig.12). Even though the range of the stellarator configurations is quite wide, these two experiments will reveal the overall physics picture of the confinement mechanisms in helical systems. Technologically, both are large superconducting devices and have a divertor, which will allow steady state operation with respectable plasmas. This inherent stellarator potential advantage, which we have claimed for many years, will become a real advantage in operating the plasma discharges without suffering from the major current disruption. Therefore collaboration between IPP, Garching and NIFS is very important. Here, I would like to show recent numerical simulation results (Fig.13) obtained by Dr. Hayashi one of the theorists who was involved in one of the collaboration studies which already took place.

Finally, in Japan, I guess probably the same as in Germany, the celebration next to the 60th birthday is that of the 70th birthday. I really hope that at the celebration party of Dr. Grieger's 70th birthday, we can congratulate each other on successful W7X and LHD experiments.

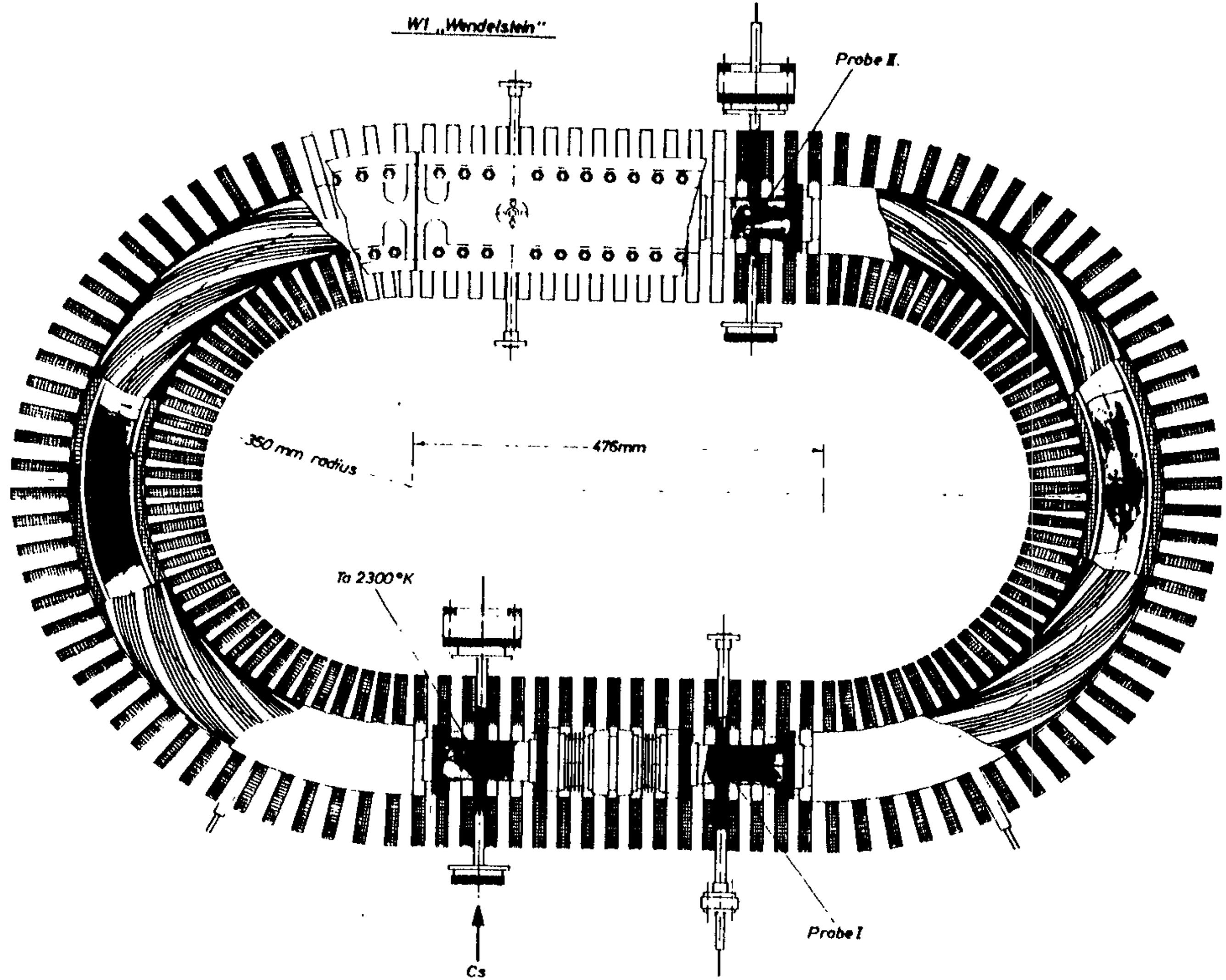


Fig.1 Wendelstein I-A

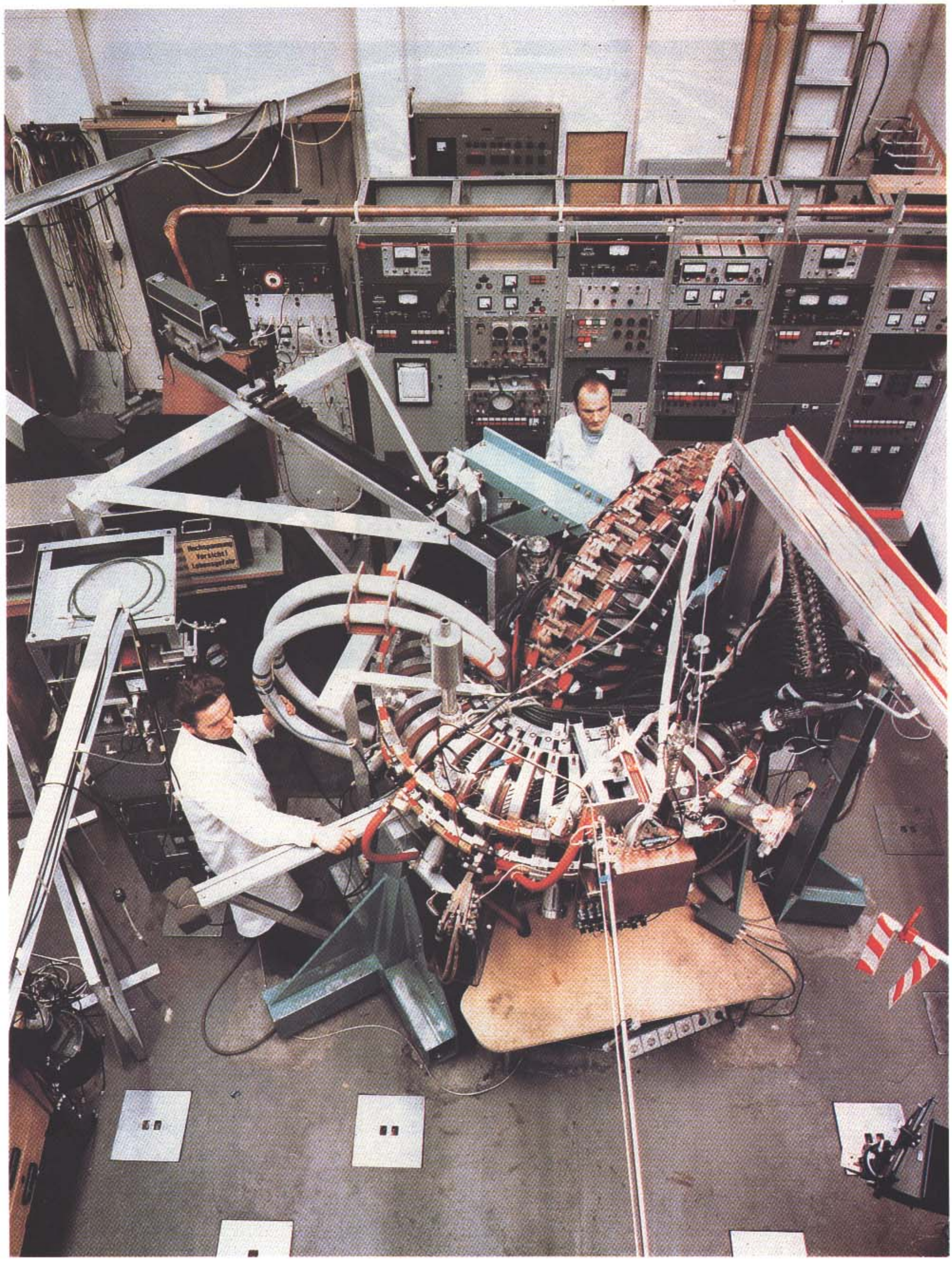


Fig.2 Wendelstein II-B

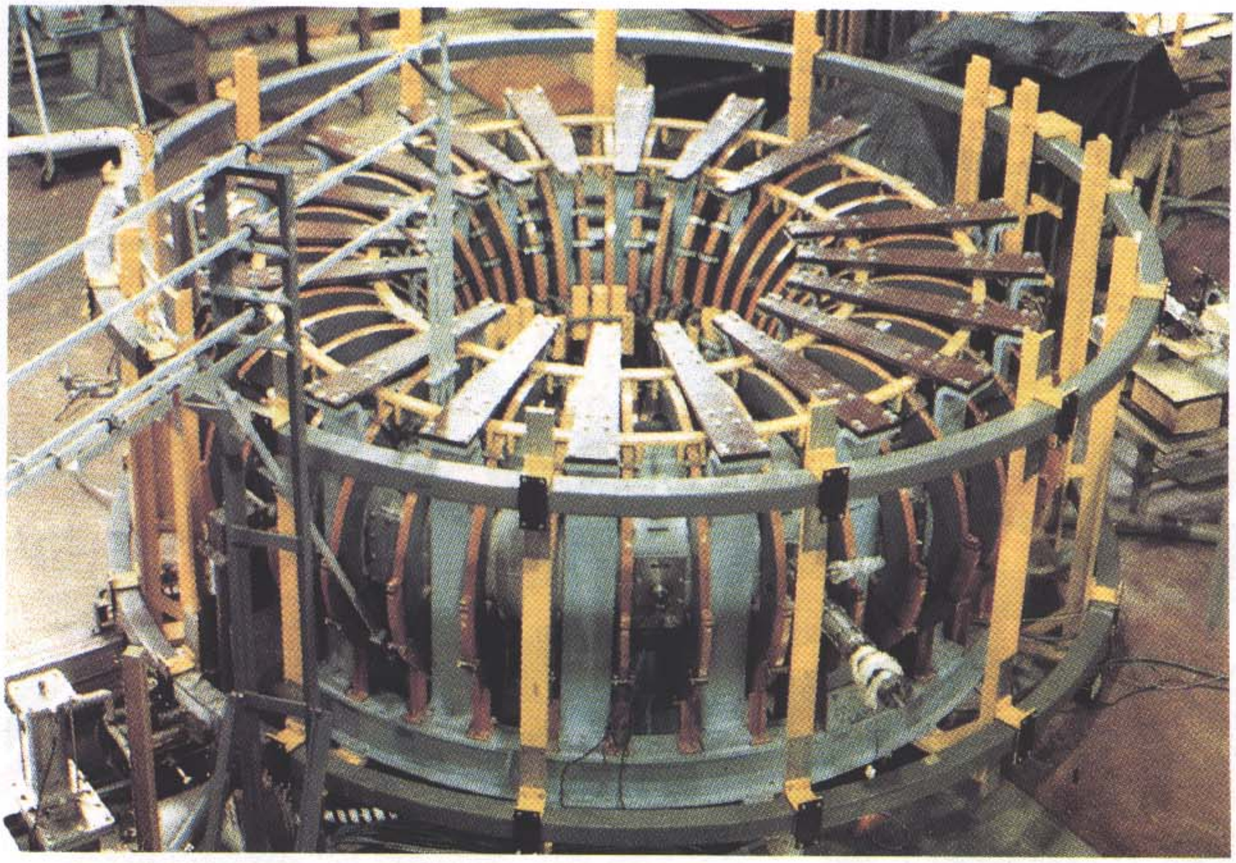


Fig.3 Heliotron D

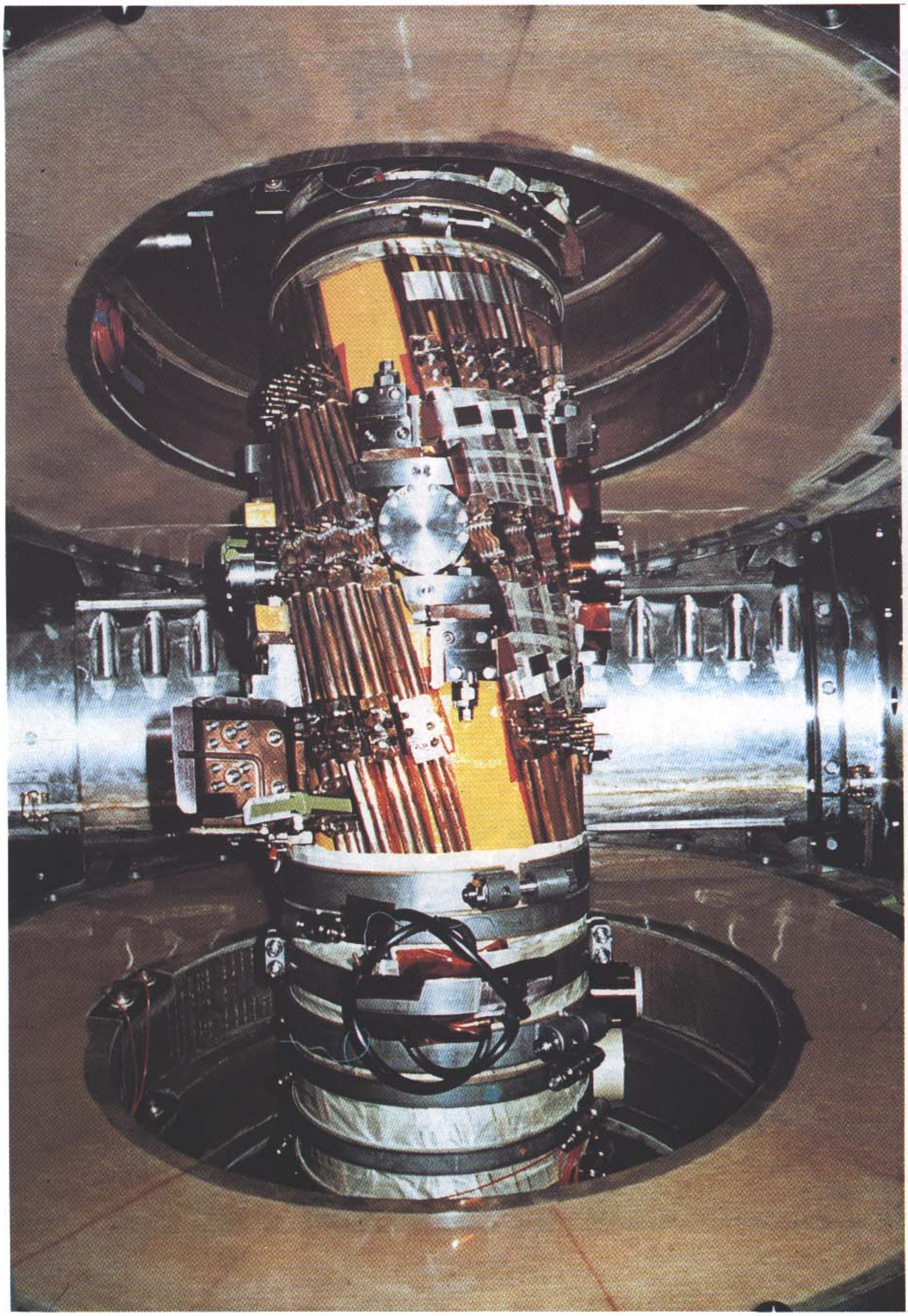


Fig.4 Wendelstein 7-A

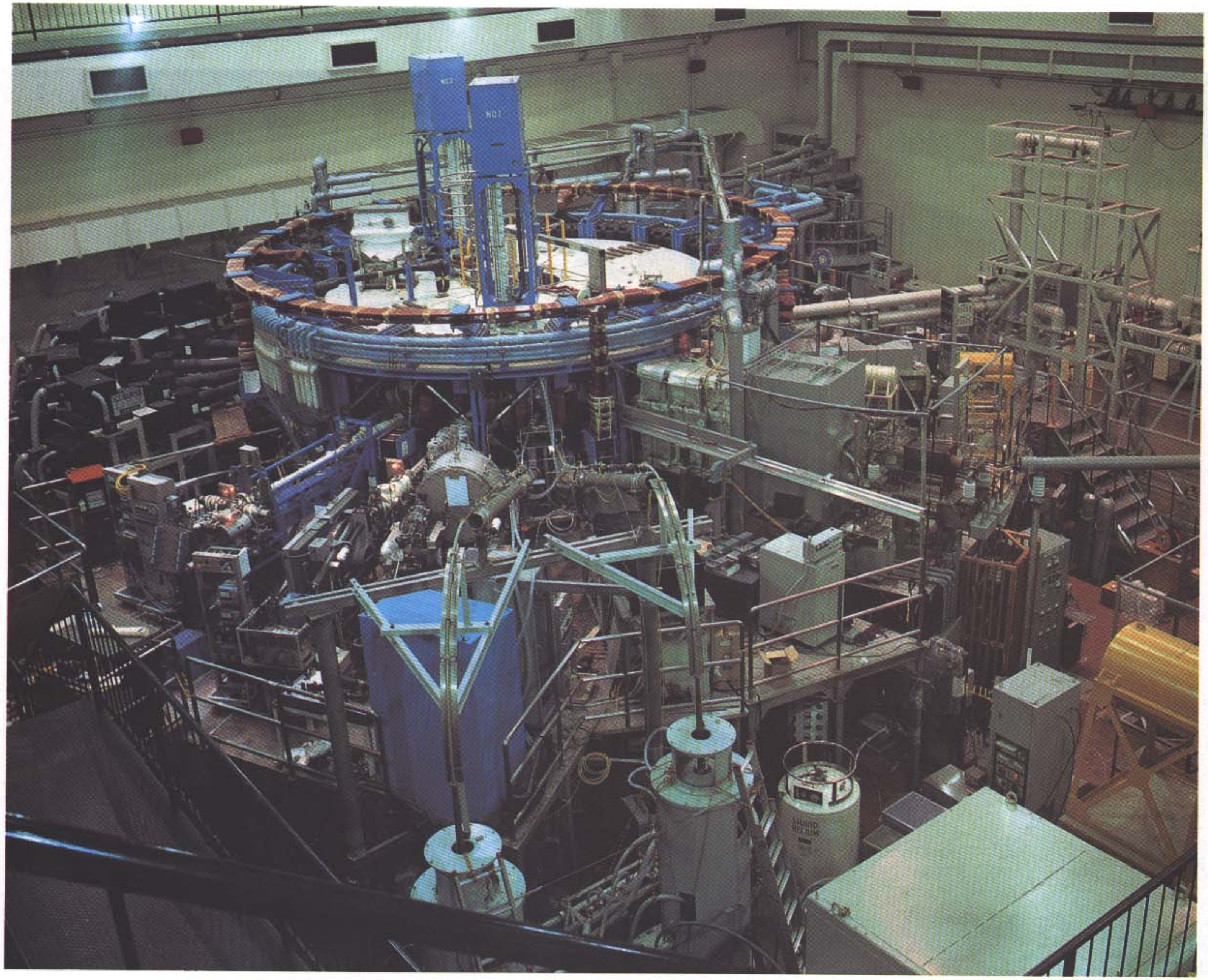


Fig.5 Heliotron E

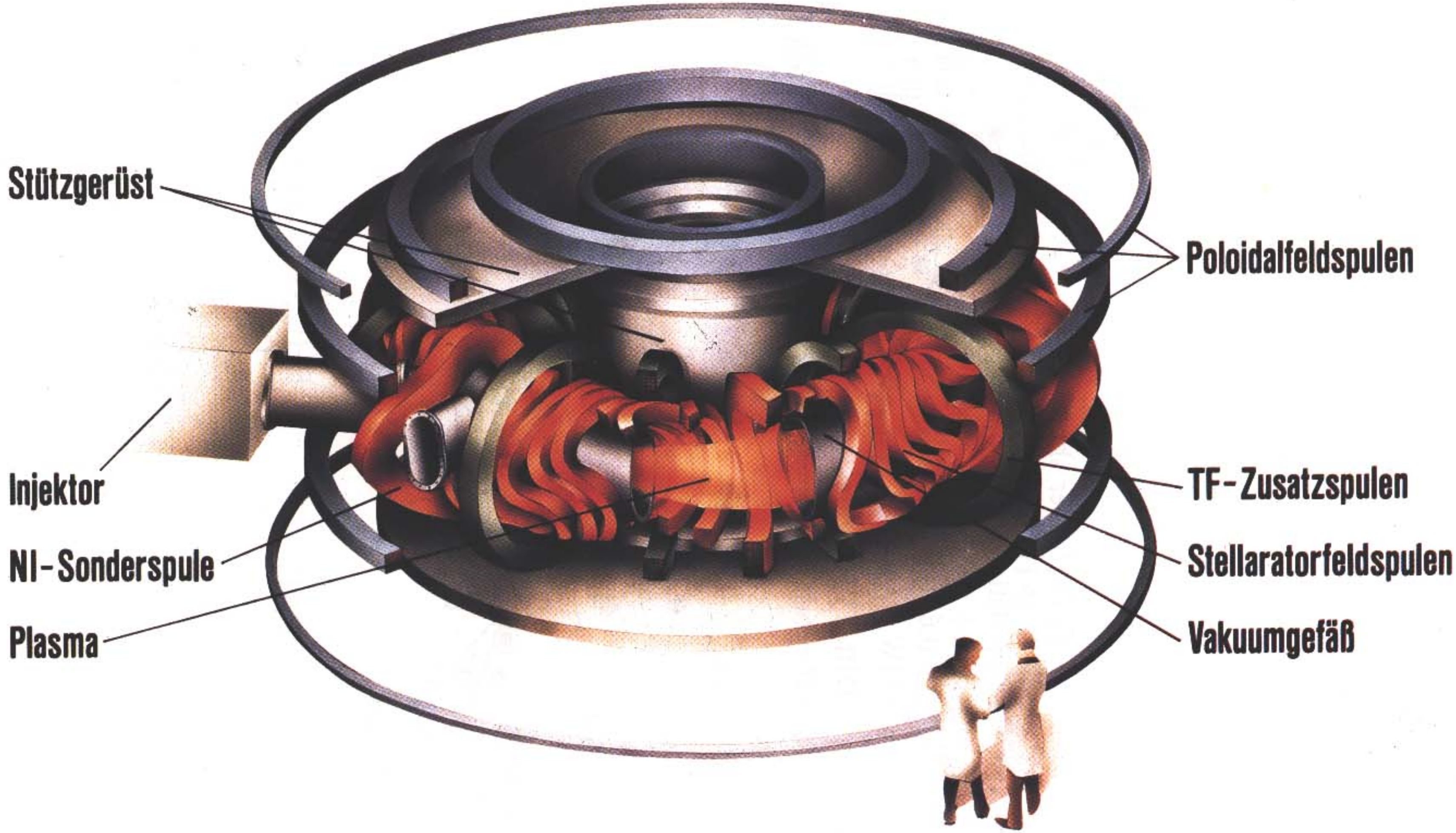
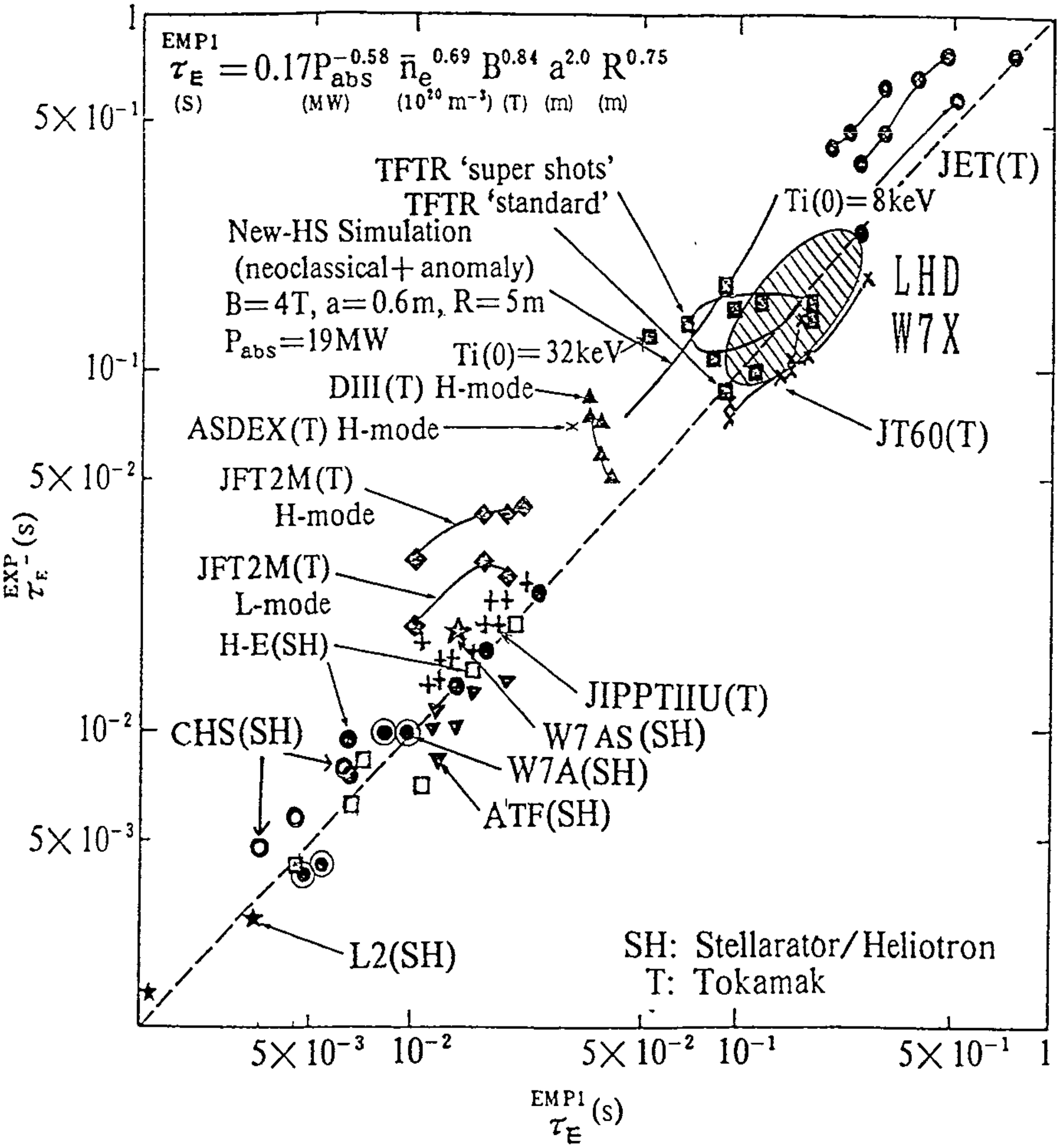


Fig.6 Wendelstein 7AS



$\tau_E^{EMP1} = 0.17 P_{abs}^{-0.58} \bar{n}_e^{0.69} B^{0.84} a^{2.0} R^{0.75}$
(S) (MW) (10^{20} m^{-3}) (T) (m) (m)
TFTR 'super shots'
TFTR 'standard'
New-HS Simulation
(neoclassical+ anomaly)
B=4T, a=0.6m, R=5m
P_{abs}=19MW
Ti(0)=32keV
Ti(0)=8keV
JET(T)
LHD
W7X
DIII(T) H-mode
ASDEX(T) H-mode
JT60(T)
JFT2M(T)
H-mode
JFT2M(T)
L-mode
H-E(SH)
JIPPTIIU(T)
W7 AS(SH)
CHS(SH)
W7A(SH)
ATF(SH)
L2(SH)
SH: Stellarator/Heliotron
T: Tokamak
τ_E^{EXP} (s)
τ_E^{EMP1} (s)

Fig.7

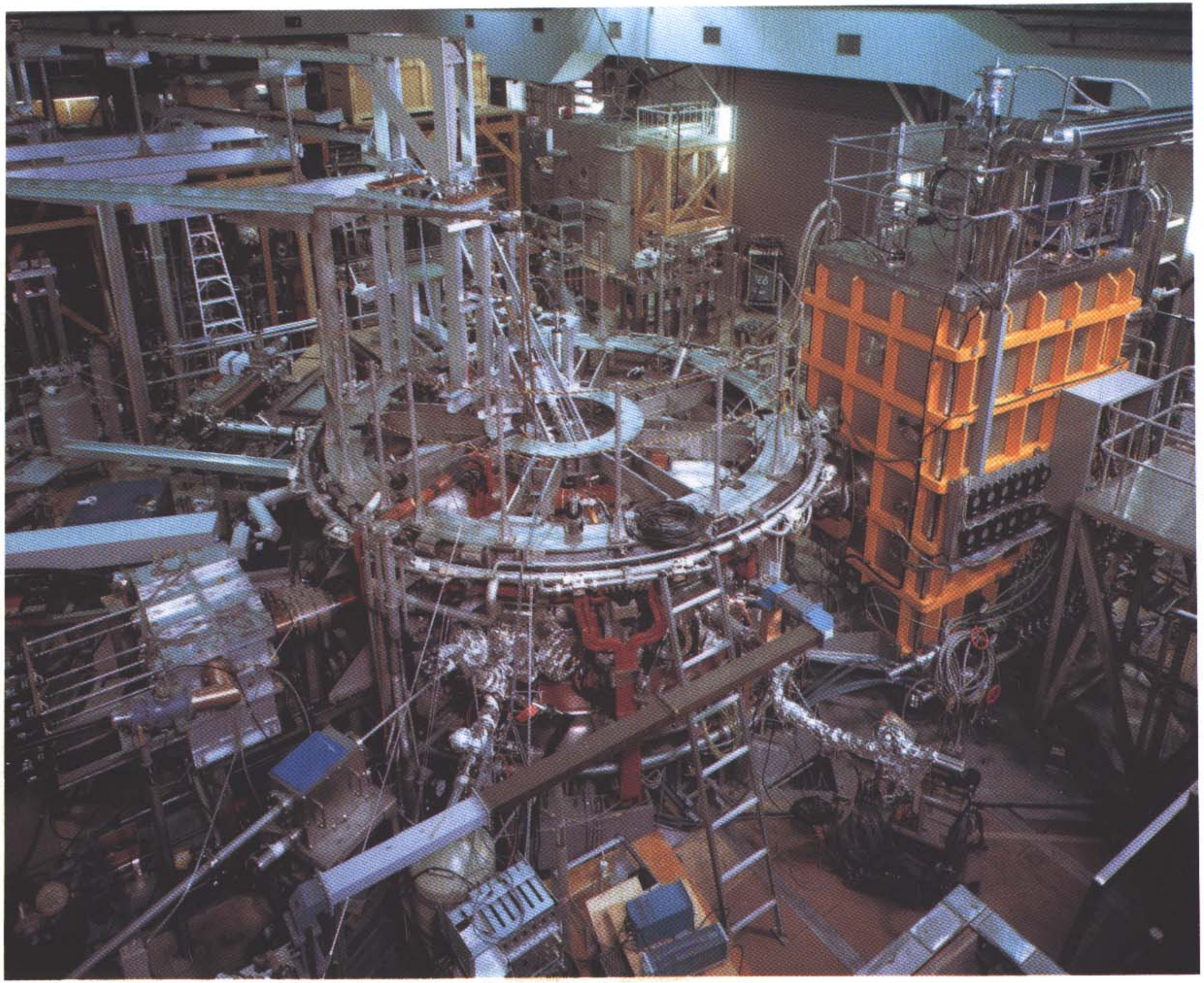
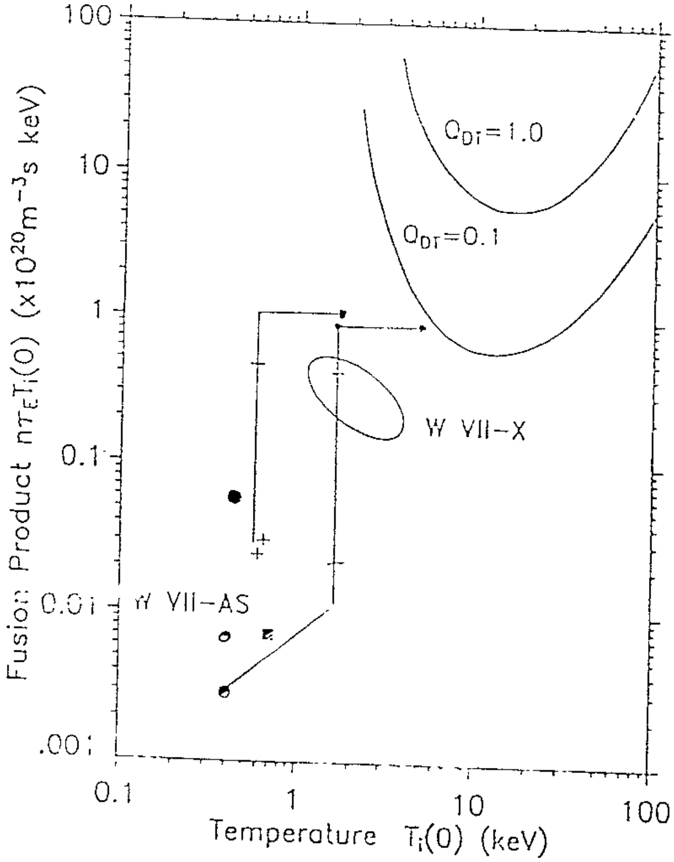


Fig.8 Compact Helical System (CHS)

FUSION DIAGRAM



WENDELSTEIN 7-X

basic coil configuration

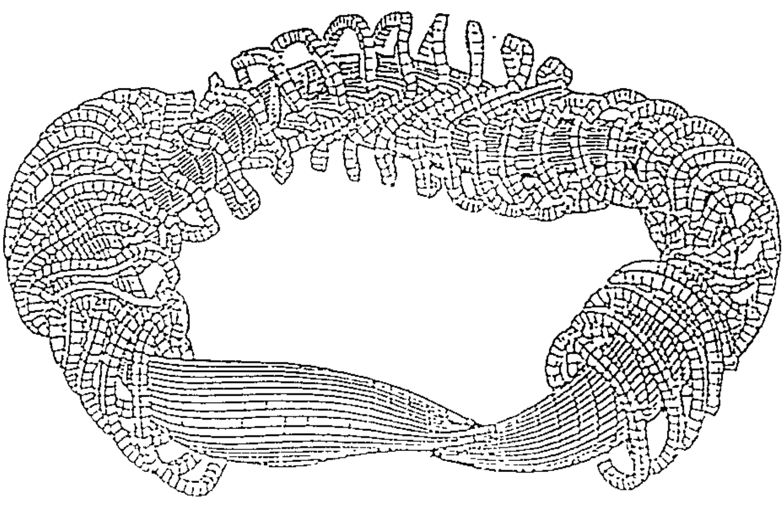


Fig.9 Wendelstein 7-X

Characteristic Nondimensional Data of W 7-X

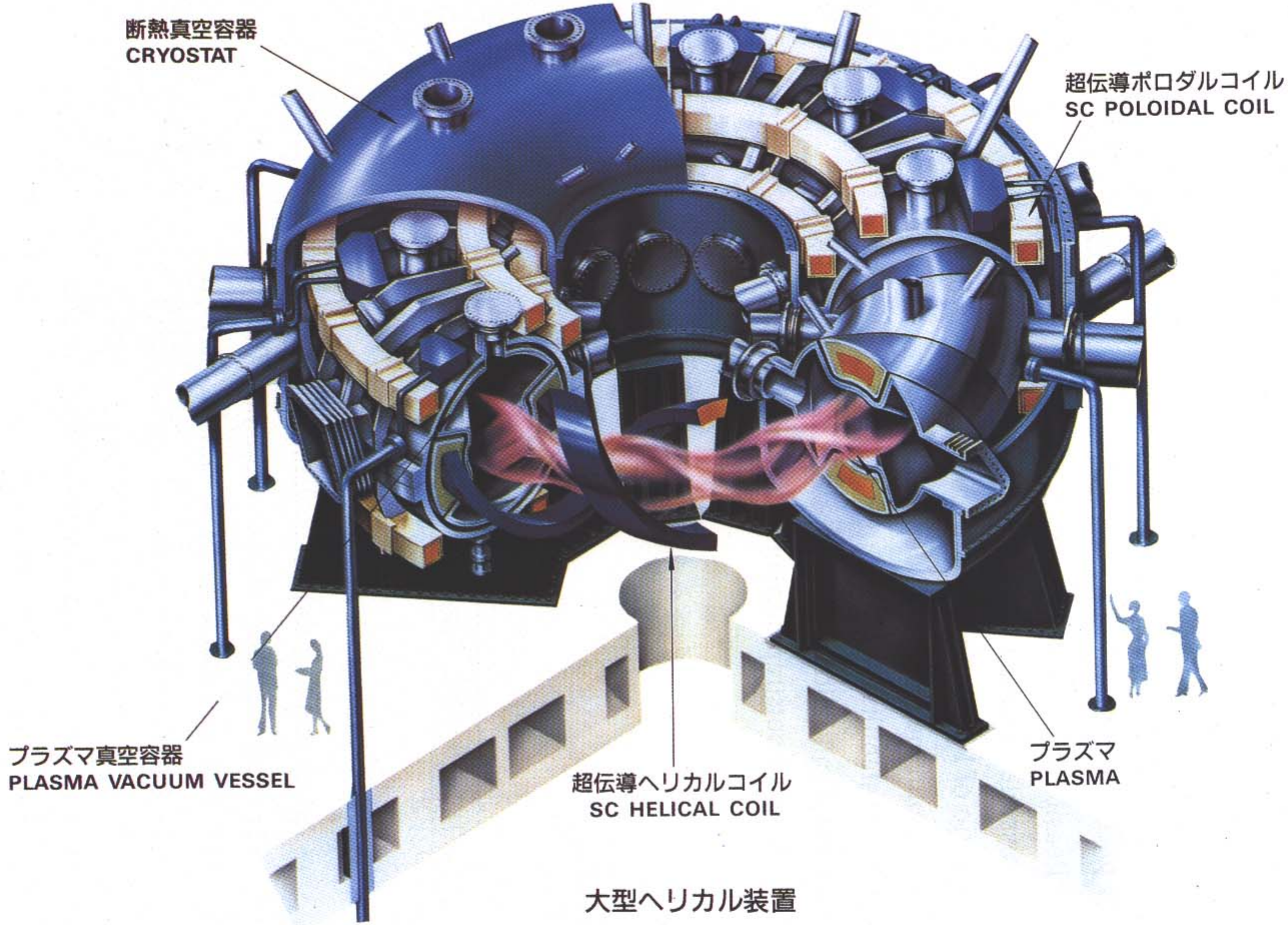
- range of configurational flexibility -

- Twist (1/q) on axis/boundary (shear):......... 0.84/0.99
- Possible variation of twist:......................... ± 0.2
- Possible variation of shear:.......................... ± 0.1
- Possible variation of mirror field:............... 0.1
- Ratio of PS- to diamagnetic currents:.......... 0.7
- Vacuum magnetic well depth:..................... 1 %
- MHD stability limit, <ß>:.. .. .. 4.3 %
- Equivalent field ripple:.............................. 1.5%
- Ratio of bootstrap current to Tokamak one: < 0.1

Characteristic Dimensional Data of W 7-X

- Average major radius:.................... 5.5 m
- Average plasma radius:.................. 0.55 m
- Average coil radius:...................... 1.14 m
- Magnetic field on axis:................. 3.0 T
- Maximum field at coils:................. 6.1 T
- Minimum distance plasma-wall:.... 0.12 m
- Total magnetic energy:.................. 600 MJ
- Maximum net force at one coil:..... 3.6 MN

Fig.10



大型ヘリカル装置
LARGE HELICAL DEVICE

Fig.11

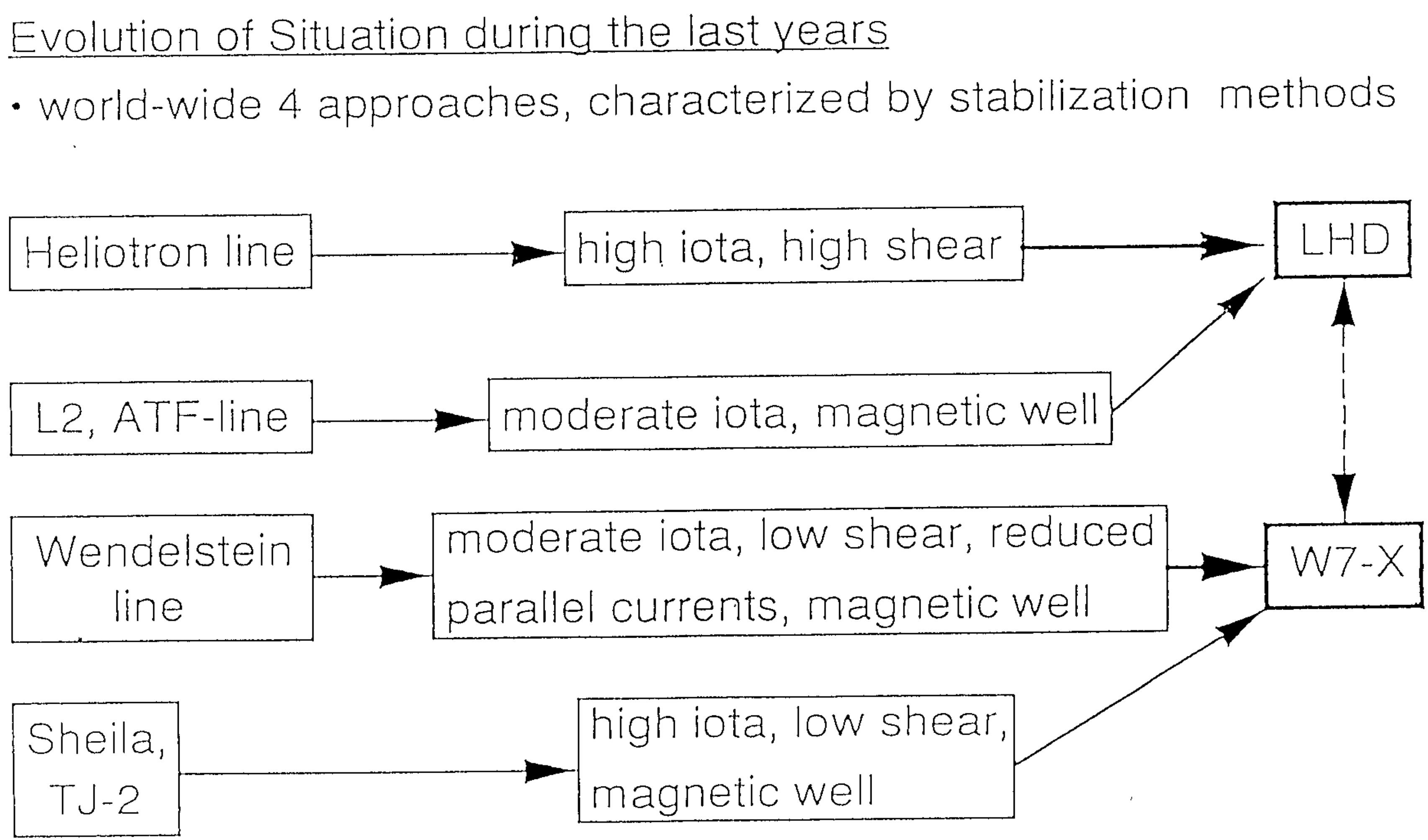

Evolution of Situation during the last years
• world-wide 4 approaches, characterized by stabilization methods
Heliotron line
high iota, high shear
LHD
L2, ATF-line
moderate iota, magnetic well
Wendelstein line
moderate iota, low shear, reduced parallel currents, magnetic well
W7-X
Sheila, TJ-2
high iota, low shear, magnetic well

Fig.12

Fig.13

Table 1 SPECIFICATIONS

	PHASE I	PHASE II
MAJOR RADIUS	3.9 m	←
COIL MINOR RADIUS	0.975 m	←
AVERAGED PLASMA RADIUS	0.5～0.65 m	←
PLASMA ASPECT RATIO	6～7	←
ℓ	2	←
m	10	←
$\gamma_c = m/2 \cdot a_o/R$ (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/mm^2	53.3 A/mm^2
NUMBER OF LAYER	3	←
LHe TEMPERATURE	4.2 K	1.8 K
POLOIDAL COIL CURRENT	STEADY	REAL TIME
INNER VERTICAL	-4.3 MA	←
INNER SHAPING	-4.4 MA	←
OUTER VERTICAL	4.9 MA	←
PLASMA VOLUME	20～30 m^3	←
ROTATIONAL TRANSFORM		
CENTER	＜0.5	←
BOUNDARY	～1	←
HELICAL RIPPLE AT SURFACE	0.2	←
PLASMA DURATION	10 sec	←
REPETITION TIME	5 min	←
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

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