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Near-Term Directions in the World Stellarator Program

J. F. Lyon

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NEAR-TERM DIRECTIONS IN THE WORLD STELLARATOR PROGRAM

J. F. Lyon

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CONTENTS

ABSTRACT 1
I. INTRODUCTION 1
II. STELLARATOR DEVELOPMENT 2
III. STELLARATOR CONFIGURATIONS 5
IV. MAJOR STELLARATOR EXPERIMENTS 10
V. CRITICAL ISSUES FOR STELLARATORS 13
V.A Near-Term Physics Issues 13
V.B Next-Generation Experiments 16
VI. REACTOR STUDIES 17
VII. SUMMARY 17
ACKNOWLEDGMENTS 19
REFERENCES 19

ABSTRACT

Interest in stellarators has increased because of the progress being made in the development of this concept and the inherent advantages of stellarators as candidates for an attractive, steady-state fusion reactor. Three new stellarator experiments started operation in 1988, and three more are scheduled to start in the next few years. In addition, design studies have started on large next-generation stellarator experiments for the mid-1990s. These devices are designed to test four basic approaches to stellarator configuration optimization. This report describes how these devices complement each other in exploring the potential of the stellarator concept and what main issues they will address during the next decade.

I. INTRODUCTION

Interest in the stellarator confinement concept has increased in recent years because of significant advances in stellarator theory, experimental results, construction techniques, and the stellarator reactor embodiment.[1,2] Three major new experiments began operation in 1988, three more are in the final design or construction stage, and design studies have begun on large next-generation experiments, one of which [the ≃$500 million Large Helical Device[3] (LHD)] has recently been approved for construction. These devices encompass a wide range of stellarator geometries with very different coil configurations and physics optimization principles. The near-term experiments are designed to test the principles of stellarator optimization, and the next-generation experiments aim at demonstrating the attractiveness of the stellarator concept with more reactor-relevant parameters. These new devices, plus two older devices that have undergone significant modifications, should provide the main results in stellarator research during the next decade.

This overview of the near-term activities planned for the world stellarator program describes how these devices complement each other in exploring the potential of the stellarator concept and discusses the main issues that they need to address. The state of stellarator development and some of the advantages of this approach are discussed in Sec. II. Some stellarator background and the main features of the different approaches being investigated are briefly reviewed in Sec. III. Better physics understanding and optimization of the magnetic configuration are the primary missions of the new experiments discussed in Sec. IV of this report and in the related papers in *Fusion Technol.* **17** (1) (1989). Of course, much higher performance is obtained from the larger tokamaks, with their higher magnetic fields and higher heating powers, than from present stellarators because confinement scales favorably with device size and magnetic field. Closing this performance gap is the mission of the next-generation stellarators, which are also discussed in Sec. IV. The key physics issues and objectives for the complementary devices described in this issue are discussed at length in Sec. V. Finally, some brief comments on stellarator reactor studies are made in Sec. VI.

II. STELLARATOR DEVELOPMENT

Both stellarators and tokamaks are toroidal confinement devices with toroidally nested magnetic surfaces, and both rely on helical (toroidal plus poloidal) magnetic fields to provide confinement and stability of the plasma. However, the currents producing these fields in stellarators flow in external conductors, allowing a wider range of magnetic configurations and more external control of the fundamental configuration parameters than is possible in tokamaks. This additional flexibility should make it possible to distinguish some of the mechanisms responsible for transport in tokamaks; in this way stellarators can also contribute to the understanding of confinement in their close relative, the tokamak.

However, the main interest in stellarators[4–6] arises from their potential advantages over conventional tokamaks as fusion reactors; in particular, that of ignited, high-beta, steady-state operation without power input to the plasma. The absence of a net plasma current eliminates potentially severe problems that would confront a tokamak reactor: the need to operate in a pulsed mode or to efficiently drive a large plasma current noninductively, the need for compatibility between the current drive scheme and the conditions for adequate confinement and high beta, the severe consequences of current-driven disruptions and large pulsed loads, etc.

Stellarators have grown from tabletop experiments in the 1950s to moderate-size (2-m major radius) devices today, and next-generation experiments two to three times larger are under design. The main device parameters for the new and near-term stellarator experiments are compared with those of some tokamaks in Table I. Two older stellarators, Heliotron E[7] and Uragan-3M,[8] are also included because these devices have undergone recent modifications and will be strong contributors to near-term stellarator research. Because the major radius R_0, the average minor radius $\bar{a}$, the plasma volume V_p, the magnetic field strength B_0 (the on-axis value), the pulse length t_p of the field or of the heating power (the shorter of the two), and the heating power absorbed in the plasma P_{abs} all enter into a measure of the device magnitude, no single parameter characterizes device "size." Here we use a rough estimate of the magnetic field energy inside the plasma volume $[W_{mag} \simeq (\pi a B_0)^2 R_0/\mu_0$ in mks units$]$ and the energy delivered to the plasma in a single pulse ($W_{heating} = P_{abs} t_p$) to characterize the magnitude of the experiment. These quantities are plotted in Fig. 1 for a number of present and near-term stellarator experiments and for the next-generation stellarators projected for the mid-1990s. The equivalent quantities for some tokamak experiments are also shown for comparison. As measured by W_{mag}, present stellarators are comparable in scale to or smaller than some of the smaller tokamaks (JFT-2M, ISX-B), and the next-generation devices are comparable in scale to some of the larger tokamaks (Tore Supra, ASDEX-Upgrade).

Recent improvements in stellarator performance result from the use of high-power (0.2–0.6 MW), high-frequency (28–70 GHz) gyrotrons to initiate currentless target plasmas and from high-power (1–3 MW) auxiliary plasma heating—electron cyclotron heating (ECH), neutral beam injection (NBI), and ion cyclotron heating (ICH). Figure 2 shows the plasma parameters obtained with these techniques for the Wendelstein VII-A stellarator[9]; here OH stands for ohmic heating. The

Table I. Parameters for Some Toroidal Experiments

Experiment	Location	R_0 (m)	$\bar{a}$ (m)	V_p (m^3)	B_0 (T)	t_p (s)	P_{abs} (MW)
			Operating Stellarators				
W VII-AS	Garching, FRG	2.0	0.2	1.58	3.0	3	5.6
ATF	Oak Ridge, U.S.A.	2.1	0.27	3.02	2.0	5	3
Heliotron E	Kyoto, Japan	2.2	0.2	1.74	2.0	0.5	7
CHS	Nagoya, Japan	1.0	0.2	0.79	1.5	2	4
Uragan-3M	Kharkov, U.S.S.R.	1.0	0.11	0.24	2.0	0.5	1.2
			Near-Term Stellarators				
Uragan-2M	Kharkov, U.S.S.R.	1.7	0.22	1.62	2.4	2	5
TJ-II	Madrid, Spain	1.5	0.22	1.43	1.0	0.5	0.4
H-1	Canberra, Australia	1.0	0.22	0.96	1.0	1	0.2
			Next-Generation Stellarators				
W VII-X	Garching, FRG	6.5	0.65	54	3.0	>10	20–30
LHD	Toki, Japan	4.0	0.5	20	4.0	>5	20
ATF-II	Oak Ridge, U.S.A.	2.0	0.52	11	4–5	∞	10–15
			Larger Tokamaks				
JET	Culham, England	2.96	1.62	153	3.5	10–20	25
JT-60	Naka, Japan	3.03	0.95	54	4.5	10	30
TFTR	Princeton, U.S.A.	2.48	0.85	35	5.2	2	27
			Smaller Tokamaks				
JFT-2M	Tokai, Japan	1.31	0.43	4.80	1.4	0.5	7
ISX-B	Oak Ridge, U.S.A.	0.93	0.31	1.74	1.5	0.3	2.5

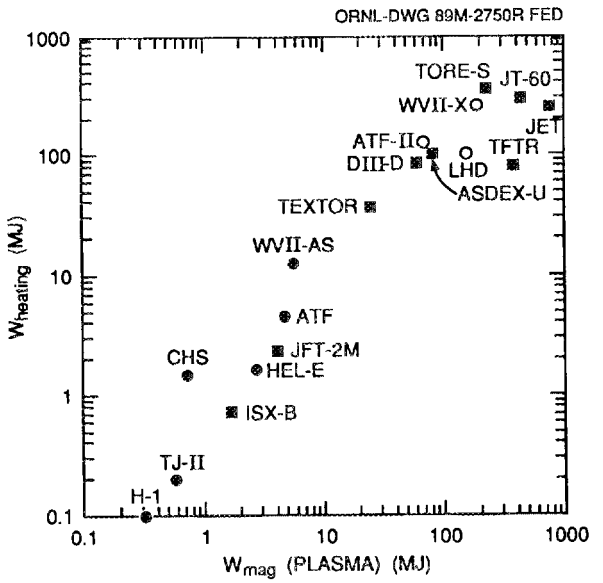

Fig. 1. The magnitude of stellarator experiments as measured by the stored magnetic energy inside the plasma volume W_{mag} and the energy delivered to the plasma in a single shot W_{heating}. The solid circles represent present and near-term stellarators, the open circles represent the large next-generation stellarators under design, and the solid squares represent some comparable tokamaks.

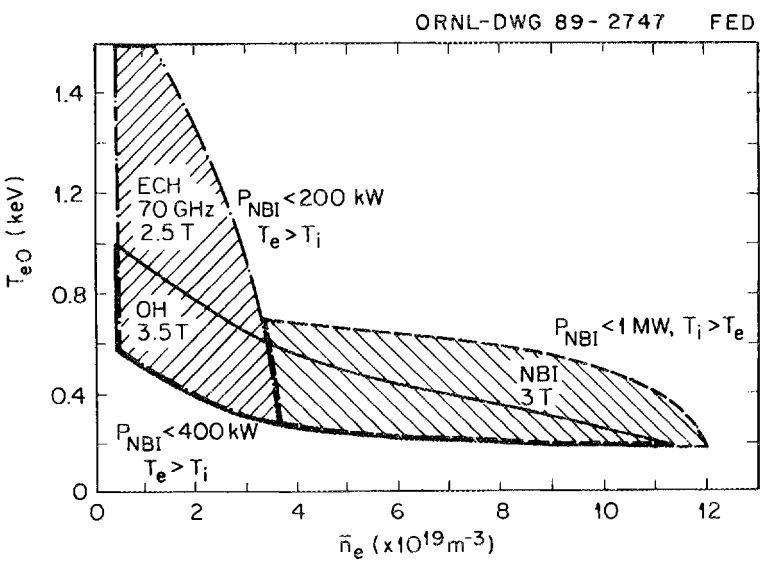

Fig. 2. Range of central electron temperatures T_{e0} and central line-averaged electron densities $\bar{n}_e$ obtained in the W VII-A experiment.

Heliotron E experiment has obtained nearly identical performance.[10] The best individual plasma parameters and the other accompanying plasma parameters for these two experiments are given in Table II. Here T_{i0} is the central ion temperature and $\langle\beta\rangle$ is the volume-averaged beta. The plasma performance attained in stellarators is comparable to or better than that attained in tokamaks of similar scale.

Table II. Highest Plasma Parameters Attained in Currentless Stellarator Plasmas

	Plasma parameters[a]						
Highest value	$\bar{n}_e$ (10^{20} m^{-3})	T_{i0} (keV)	T_{e0} (keV)	$\langle\beta\rangle$(%) at B_0 (T)	$\bar{n}_e\tau_E{}^b$ $\times 10^{18}$ (m^{-3}·s)	P_{abs} (MW), type	Device[c]
T_{e0}	0.1	0.11	<u>2.4</u>	0.1, 2.5	0.04	0.11, ECH	W VII-A
T_{i0}	0.26	<u>1.6</u>	0.66	0.3, 1.9	0.26	<u>3.5</u>, NBI	H-E
Pressure	1.2	1.0	0.7	<u>0.45, 3.2</u>	2.5	0.46, NBI	W VII-A
$\langle\beta\rangle$	0.9	0.41	0.41	<u>2</u>, 0.94	0.63	1.8, NBI	H-E
$\bar{n}_e\tau_E{}^b$	1.4	0.3	0.3	0.47, 1.9	<u>5.0</u>	2, NBI	H-E

[a]Taken from Ref. 1.

[b]Here τ_E is the net energy replacement time corrected for radiation losses.

[c]W VII-A = Wendelstein VII-A, H-E = Heliotron E.

III. STELLARATOR CONFIGURATIONS

The three dominant types of stellarators being studied are torsatrons (or heliotrons), modular stellarators, and heliacs. Examples of their coil configurations are shown in Fig. 3. Torsatrons (or heliotrons) have currents in the same direction in the helical windings and no separate toroidal field (TF) coils. The leading examples are the Advanced Toroidal Facility[11] (ATF), the Compact Helical System[12] (CHS), Heliotron E,[7] and Uragan-2M.[13] Modular stellarators have a toroidal set of nonplanar (toroidally distorted) coils and no separate helical windings. The best examples are Wendelstein VII-AS[14] (W VII-AS) and the small Interchangeable Module Stellarator[15] (IMS). Heliacs have a toroidal set of circular coils whose centers follow a helical path around a linked, central toroidal-ring conductor. The two largest examples are H-1[16] and TJ-II.[17]

A particular coil geometry is chosen for access, flexibility, best match to a particular magnetic geometry, and engineering or facility constraints. The torsatron geometry allows a maximum of access and flexibility, has only outward radial forces, and is best suited for higher-shear configurations. The modular stellarator geometry allows a wider range of magnetic configurations and is best suited for low-shear configurations. The heliac geometry allows much higher rotational transform with low shear and has good access.

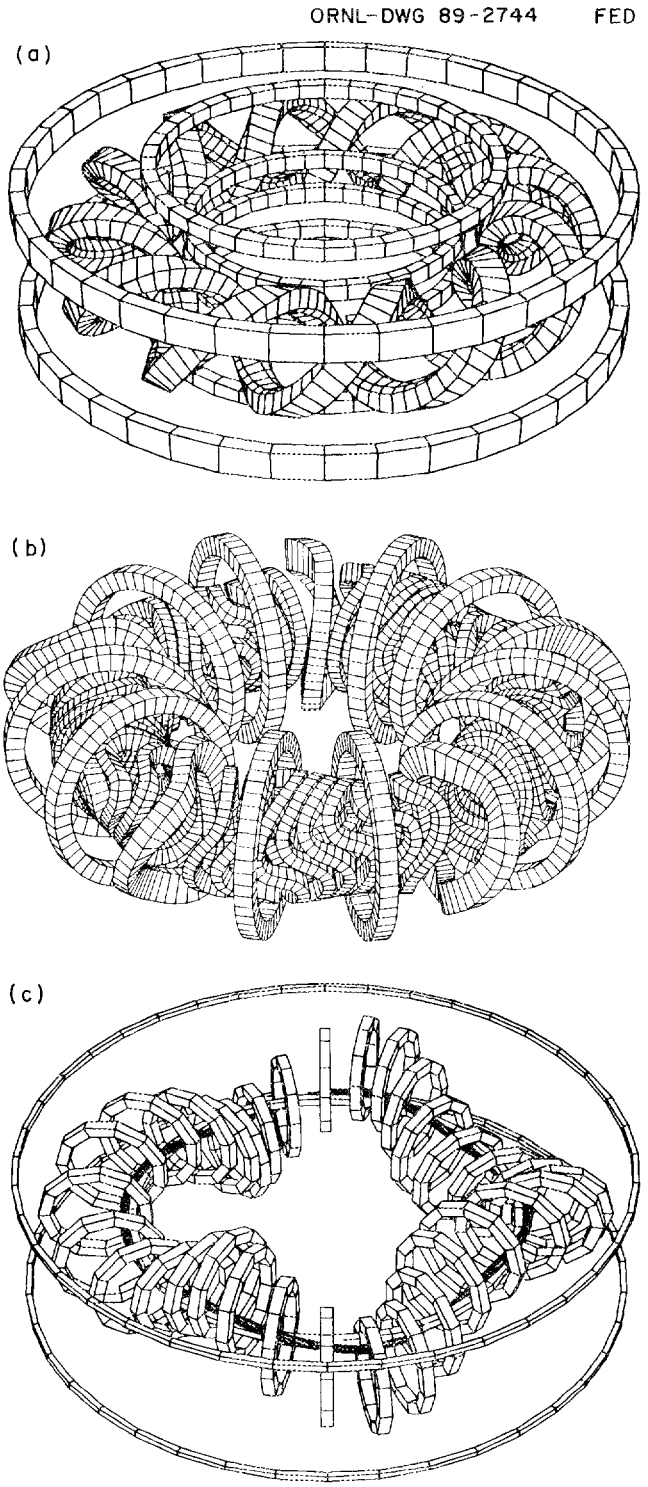

Fig. 3. The main coil configurations used in present-day stellarator experiments: (a) a torsatron, ATF; (b) a modular stellarator, W VII-AS; and (c) a heliac, TJ-II.

The main stellarator approaches can best be described[1] in terms of their magnetic configuration properties and their optimization approach rather than the coil geometry used to realize a desired magnetic configuration. In principle, the coil geometry used to realize a given magnetic configuration (determined uniquely by the shape of the last closed flux surface) is not unique, so the magnetic configuration description is more fundamental.

The stellarator magnetic configuration is inherently nonaxisymmetric. Figure 4 shows the last closed magnetic surface, beyond which the magnetic configuration has ergodic field lines, for a typical stellarator (ATF) and cross sections of the magnetic surfaces at the beginning of, one-fourth of the way through, and halfway through a toroidal field period. The distortion of the magnetic surfaces as they rotate poloidally about the magnetic axis depends on the structure of the magnetic field. Shaping of the flux surfaces gives the desired properties of the magnetic

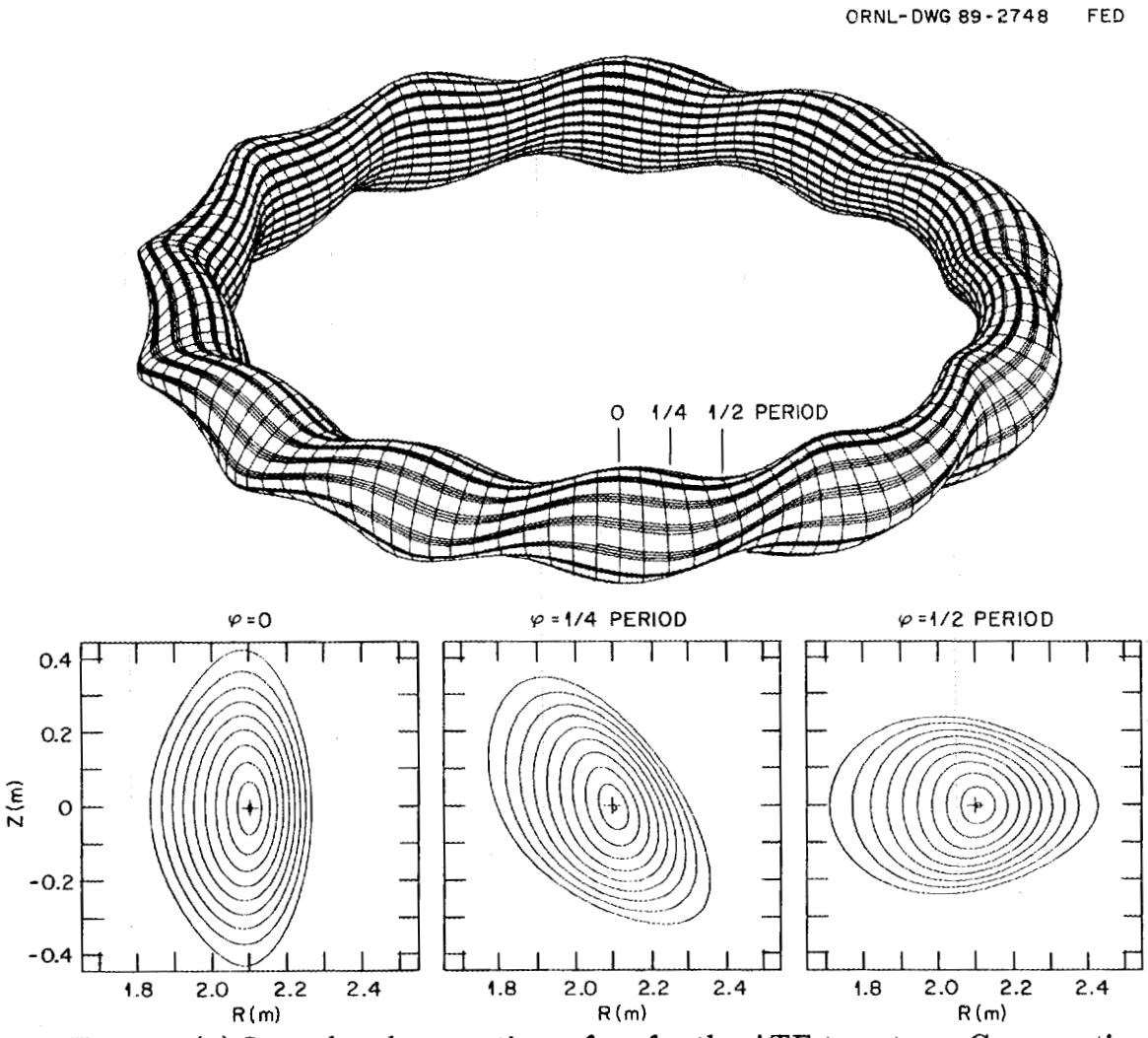

Fig. 4. (a) Last closed magnetic surface for the ATF torsatron. Cross section of magnetic surfaces at (b) $\phi = 0$ deg, (c) $\phi = 7.5$ deg, and (d) $\phi = 15$ deg.

configuration. For example, the Helias[18] surfaces shown in Fig. 5 change from bean-shaped to teardrop-shaped to triangular in one-half of a field period. This complicated plasma shape has a much simpler representation in flux coordinates[19] and can simulate a helically symmetric configuration. This configuration is predicted to have high-beta capability and forms the basis for the large, next-generation Wendelstein VII-X (W VII-X).

The physical meanings of the various parameters that characterize a stellarator magnetic configuration are discussed in detail in Ref. 1. The most important of these parameters are rotational transform, shear, magnetic well depth, helical ripple, aspect ratio, ellipticity, helical axis excursion, and number of field periods. The rotational transform ι gives the number of poloidal revolutions per toroidal revolution that a magnetic field line makes as it transits the torus on a magnetic (or flux) surface. The shear is proportional to ι', where the prime denotes differentiation with respect to ψ, $1/2\pi$ times the toroidal flux within a magnetic surface with volume V. The magnetic well depth is given by $\Delta V'/V'(0)$, where $V' = \int dl/B$, $\Delta V' = V'(\psi_1) - V'(0)$, and 0 to ψ_1 is the region over which $V'' < 0$. The poloidal variation of $\int dl/B$ is related to the parallel equilibrium (Pfirsch-Schlüter) currents, caused by plasma pressure, which determine the magnetic axis shift and the deviation of passing-particle orbits from magnetic surfaces. Another important feature of the spatial structure of the magnetic field is the depth of the helical ripple that traps particles, which then drift radially outward. The plasma aspect ratio $A = R_0/\bar{a}$, and the ellipticity measures the average noncircularity of the plasma cross section. In general, the magnetic axis can be noncircular with a helical axis excursion. The poloidal multipolarity of the helical field structure is given by ℓ, and the number of times the basic magnetic configuration repeats toroidally is M, the number of field periods.

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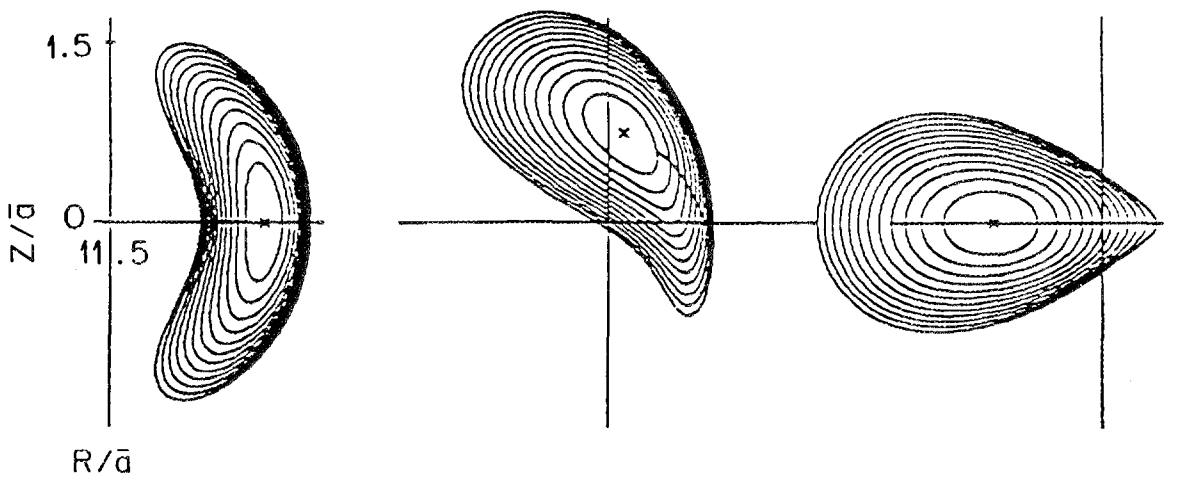

Fig. 5. Cross section of magnetic surfaces for a five-field-period Helias configuration at (a) $\phi = 0$ deg, (b) $\phi = 18$ deg, and (c) $\phi = 36$ deg. A significant helical-axis excursion is present.

As discussed in Ref. 1, there are four basic magnetic configuration types, based on different approaches to magnetic configuration optimization, that are being explored in the new and near-term stellarator experiments. These approaches (and the experiments based on them) are

(1) high ι with strong shear stabilization and a high degree of helical symmetry (Heliotron E);
(2) moderate ι with magnetic-well and shear stabilization to access a second stability region (ATF, CHS, Uragan-2M);
(3) moderate ι with low shear, reduced variation of $\int dl/B$ on flux surfaces, drift-orbit optimization, and magnetic-well stabilization (W VII-AS); and
(4) high ι with low shear and magnetic-well stabilization produced by a helical magnetic axis (H-1, TJ-II).

Together these approaches cover the range of interesting magnetic configurations and complement each other in testing the important configuration optimization principles, which is a major goal of the experiments discussed in Sec. IV. Figure 6 shows the configuration parameter space covered by stellarators and tokamaks for two characteristics of the magnetic geometry, an average rotational transform $\langle\iota\rangle = [\iota(\bar{a}) + \iota(0)]/2$ and a global magnetic shear $\Delta\iota/\langle\iota\rangle$, where $\Delta\iota = \iota(\bar{a}) - \iota(0)$. For tokamaks, $\iota = 1/q$, the tokamak safety factor, and $q(0) = 1$ are assumed. The tokamak family is represented by a single curve. By contrast, the stellarator family covers a large region and has shear of the opposite sign to that in tokamaks.

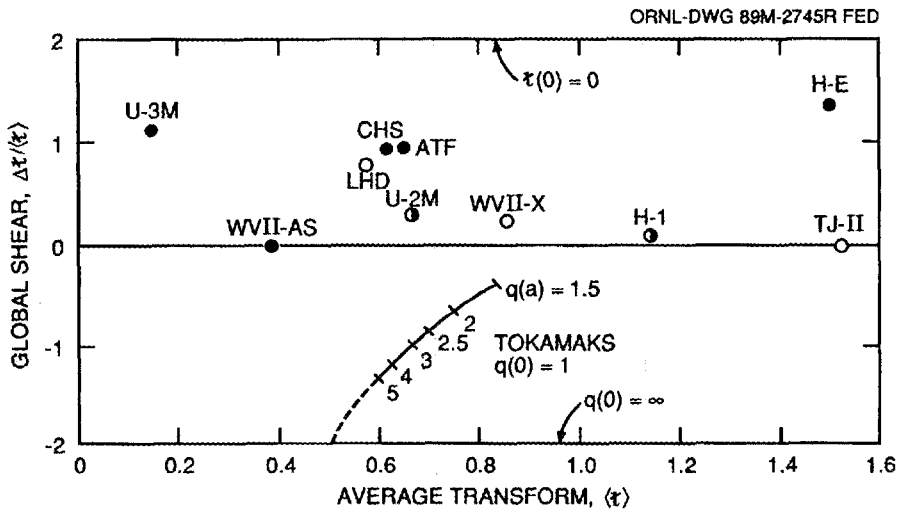

Fig. 6. Magnetic configuration space (global shear vs average rotational transform) for different stellarators and for tokamaks. The solid circles represent existing devices, the half-solid circles represent near-term devices under construction, and the open circles represent next-generation devices in the design study phase.

IV. MAJOR STELLARATOR EXPERIMENTS

The special features of the major near-term stellarators are listed in Table III. The experiments fall into two classes: torsatrons (or heliotrons) with helical windings and significant shear (Heliotron E, ATF, Uragan-2M, CHS, and Uragan-3M) and the modular-coil devices with low shear (W VII-AS, H-1, and TJ-II). These devices cover more than a factor of two in plasma aspect ratio, from $A = 11$ in Heliotron E to $A = 5$ in CHS.

The near-term experiments are well suited for testing their respective optimization principles. Heliotron E has been operating since 1980, so its general behavior is relatively well understood. Its mission now is to explore issues that are important for the design of the LHD experiment: higher-power ICH, the divertor geometry, control of electric fields, possible access to a second stability region, etc. ATF is the largest of the present generation of experiments and has the most versatile sets of coils for exploring a wide range of magnetic configurations and the

Table III. Characteristics of New and Near-Term Experiments

Device	Special Design Features	Special Configuration Features	Key Physics Issues
Heliotron E	Single HF winding, TF coil set	High $\iota(\bar{a})$, high shear, magnetic hill, high helical symmetry	Shear stabilization, divertor, field optimization
W VII-AS	Modular nonplanar coils, TF coil set, enlarged coils for tangential access, high field (3 T)	Reduced secondary currents, very low shear, moderate ι, global magnetic well	Avoidance of resonances, improved transport and stability
ATF	2 jointed HF windings, versatile sets of VF coils, steady state at 1 T, large access ports	Low-order rational ι in well or shear, second stability region, helical axis possible	Improved transport and β limits in second stability region
CHS	2 HF windings on vacuum vessel, low aspect ratio	Like ATF configuration	Transport and beta limits at low $R/\bar{a}$, flux surface fragility
Uragan-3M	$\ell = 3$ helical windings, external vacuum tank	Low $\iota(0)$ and $\iota(\bar{a})$, high shear, $\ell = 3$ configuration, divertor	ICH startup, flux surface fragility
Uragan-2M	Split HF windings, 4 field periods, TF coil set	Intermediate shear, low helical ripple	Improved transport at low collisionality
H-1	$M = 3$ heliac, coils in vacuum tank, small $\ell = 1$ winding on central ring	High ι, low shear, bean-shaped plasmas, large helical axis excursion	Fragility of flux surfaces, beta limit, transport
TJ-II	$M = 4$ heliac, large split $\ell = 1$ winding on central ring, jointed TF coils	Like H-1, additional control on shear, ι	Same as H-1

largest access ports. It will need to demonstrate the ability to avoid unfavorable $\iota(\psi)$ profiles (i.e., islands) as beta increases, access to the second stability region, and adequate confinement of helically trapped particles. CHS will be a good test of an ATF-like configuration at much lower aspect ratio ($A = 5$ vs 7.8 for ATF). It has a vertical field (VF) coil system similar to that of ATF, which should aid in compensating for the fragility of flux surfaces at lower A.

The principal low-shear experiment, W VII-AS, is a significant step beyond the successful W VII-A because it is larger ($\bar{a} = 20$ cm vs 10 cm), has higher heating power (3–4 MW vs $\lesssim$1 MW) with longer pulse length (3–10 s vs $\simeq$0.2 s) and better heating geometry (tangential NBI vs perpendicular NBI), and has better access for diagnostics. W VII-AS will need to demonstrate the advantages of the "Advanced Stellarator" configuration and to avoid shear introduced by plasma currents that could bring low-order resonances into the plasma. H-1 and TJ-II will have modest heating capabilities, which should still be sufficient to test their ability to avoid the flux surface breakup at finite beta that could arise from the combination of toroidal effects breaking the helical symmetry, the low shear, and the high density of rational values for ι.

In addition to these experiments, the older (1975), somewhat smaller ($\bar{a} = 0.11$ m, $V_p \simeq 0.24$ m^3, $B = 1.5$ T) L-2 torsatron[20] and a number of much smaller university experiments will continue to study basic issues of interest to stellarators. Among these smaller experiments are the Heliotron DR experiment, the IMS modular stellarator, the High-Beta Q Machine (HBQM) linear heliac, the SHEILA heliac, the SHATLET-M modular torsatron, and the Asperator H heliac. More information on these experiments can be found in Ref. 21.

Present and near-term stellarators aim at validating the four main stellarator approaches and the optimization criteria on which they are based. However, these experiments are relatively small compared to present-day tokamaks. The main device limitations are small plasma radius ($\bar{a} = 0.2$–0.27 m), modest magnetic field ($B_0 = 2$–3 T), modest plasma heating power (a few megawatts), and modest pulse length (less than a few seconds).

To fully demonstrate the potential of the stellarator approach, next-generation experiments need larger plasma radii ($\bar{a} = 0.5$–0.6 m), higher fields ($B_0 = 4$–5 T), higher plasma heating powers ($P_{abs} = 10$–30 MW), and longer pulse lengths (10 s to steady state). Their plasma parameters also need to be significantly higher than those expected in the near-term experiments [i.e., $T_{i0} = 5$–10 keV, $\langle\beta\rangle > 5\%$, and $n\tau_E > 3 \times 10^{19}$ m^{-3}·s vs T_{i0}, $T_{e0} \sim 1$–2 keV, $\langle\beta\rangle \sim 2$–$3\%$, and $n\tau_E \sim (2$–$3) \times 10^{18}$ m^{-3}·s]. Such experiments are now under design and are listed in Table IV. In general, these devices use superconducting coils for long-pulse or steady-state operation, have fields and sizes at least twice those of the new and near-term experiments, and have an order of magnitude more heating power than present-day stellarator experiments.

The four configuration optimization approaches discussed in Sec. III, which the near-term experiments will test, are converging to two basic approaches for the next-generation experiments. The first is the torsatron with helical windings, moderate ι, a magnetic well at the center, and shear at the edge to simultaneously optimize high-beta capability with reduced ripple transport, energetic orbit losses,

Table IV. Major Device Parameters (Tentative) for Proposed Next-Generation Experiments

	LHD	W VII-X	ATF-II
Location	Toki, Japan	Garching, FRG	Oak Ridge, U.S.A.
Type	$\ell = 2$ heliotron	Modular stellarator	$\ell = 2$ torsatron
Special design features	Superconducting coils, divertor	Modular, non-planar coils; helical axis	Low aspect ratio, steady state
Special physics features	High β, improved confinement	Helias optimization	Second stability region
R_0 (m)	4.0	6.5	2.0
$\bar{a}$ (m)	0.5–0.6	0.65	0.52
B_0 (T)	4.0	3.0	4–5
Pulse length (s)	>5	>10	∞
Aspect ratio A	7–8	10	4.7
M	10	5	9
Heating power (MW)	20	20–30	10–15

and bootstrap current. LHD will test this approach at moderate aspect ratio, and ATF-II would extend it to low aspect ratio. The second approach is the Helias (shown in Fig. 5) with modular coils, moderate ι, low shear, a global magnetic well, a helical axis, and plasma shaping to produce a quasi-helical configuration with an acceptable level of bootstrap current. W VII-X would test this approach.

LHD,[3] approved for the new National Institute for Fusion Science in Toki, Japan, will have a magnetic configuration similar to that of ATF, but with improved confinement properties. The proposed schedule calls for start of engineering design and R&D activities in April 1989, start of construction in April 1991, and facility commissioning in 1995. The goals for LHD are (1) high $\bar{n}\tau_E T_i$ operation with $\bar{T}_i = 3$–4 keV, $\bar{n}_e = 10^{20}$ m^{-3}, and $\tau_E = 0.1$–0.3 s; (2) high T_i operation with $T_{i0} = 10$ keV, $\bar{n}_e = 2 \times 10^{19}$ m^{-3}, and $\tau_E = 0.1$ s; and (3) high-beta operation ($\langle\beta\rangle \leq 5\%$) with $T_{i0} = T_{e0} = 0.6$ keV, $\bar{n}_e = 10^{20}$ m^{-3}, and $\tau_E = 0.03$ s (at $B_0 = 1$ T). Alpha-particle confinement would be simulated by injection of high-energy particles. LHD will have sufficient size, field, and heating power to test the reactor relevance of the moderate-aspect-ratio approach. Key issues for LHD are particle and power handling (optimum divertor geometry and plasma-wall separation) and the optimum configuration for high-beta operation with reduced bootstrap current and good confinement (reduced energetic trapped-particle orbit losses and low thermal conductivity) in the low collisionality regime.

The W VII-X modular stellarator[22] being proposed for the Max-Planck-Institut für Plasmaphysik (Garching, FRG) would have larger aspect ratio and major

radius than LHD, lower magnetic field, and more heating power. The approval process for W VII-X is not as advanced as that for LHD. Plans call for a phase I (design) proposal in 1989, a phase II (construction) proposal for the end of 1990, and completion of construction in 1996. The parameter goals for W VII-X are similar to those of LHD: (1) demonstration of good confinement at reactor-relevant parameters [$\langle T \rangle = 5$–7 keV, $\langle n \rangle = (1.5$–$3) \times 10^{20}$ m^{-3}, and $\langle \beta \rangle = 2\%$ at $B_0 = 3$ T] and (2) operation at high beta ($\langle \beta \rangle = 5.6\%$, $\langle T \rangle = 2.5$ keV, and $\langle n \rangle = 8 \times 10^{19}$ m^{-3} at $B_0 = 2$ T). W VII-X would allow a definitive test of the modular-coil Advanced Stellarator approach.[23] Its large size ($R_0 = 6.5$ m, $\bar{a} = 0.65$ m) would allow the addition of high-power heating (20–30 MW) and would discriminate against anomalous edge transport. Its higher aspect ratio, enhanced by quasi-helically symmetric features (to the extent that they are compatible with a low bootstrap current), should allow high beta limits, low ripple-induced transport, and good confinement of energetic ions.

The ATF-II[24] studies at Oak Ridge National Laboratory aim at a device with $A = 3.5$–5, roughly one-half to one-third that of the other next-generation devices. It is not an approved project, and the design studies are at an earlier stage than the LHD and W VII-X studies. The reference ATF-II case has helical windings, but modular-coil options are also being studied. ATF-II would have approximately the same plasma radius and magnetic field as LHD, but considerably smaller major radius, less heating power, and lower cost. The parameter goals of ATF-II are similar to those of the other devices: steady-state operation with $T_{i0} > 5$ keV, $n\tau_E > 5 \times 10^{19}$ m^{-3}·s, and $\langle \beta \rangle = 5$–$8\%$ (at reduced field). ATF-II would allow a definitive test of the low-aspect-ratio compact torsatron approach.[25] The key issues to be addressed are possible breakup of flux surfaces, increased ripple-induced transport, increased energetic orbit losses, larger bootstrap currents as beta increases, and the configuration optimization that simultaneously minimizes these deleterious effects.

V. CRITICAL ISSUES FOR STELLARATORS

The main task for the near-term and next-generation stellarators is determination of the optimum magnetic configuration. This involves verification and improvement of the sometimes conflicting optimization criteria and determination of the best combination of configuration properties (transform, shear, magnetic well, aspect ratio, variation of $\int dl/B$ on a flux surface, cross-section shape, diverted flux geometry, degree of helical-axis excursion, etc.). The key issues involved in magnetic configuration optimization are basically the same as for tokamaks (beta limits, confinement, and operational control), but more configuration flexibility and external control are available in stellarators.

V.A. Near-Term Physics Issues

A key component in optimization is control of the magnetic configuration. Minimization of field errors is important for stellarators with significant shear (ATF, CHS, Heliotron E, and Uragan-2M) because magnetic islands occur on flux surfaces with low-order rational values of ι if resonant field perturbations are present. These

experiments have versatile poloidal field coil sets to vary the configuration properties and minimize the effect of such islands, should they occur. Low-shear configurations (W VII-AS, H-1, and TJ-II) can, in principle, avoid low-order rational values of ι, and the attendant islands, but then minimization of additional shear caused by finite-beta or heating-produced currents becomes crucial. W VII-AS can vary its shear, with accompanying changes in plasma shape and magnetic axis location, through changes in the vertical field and in the supplementary toroidal field. Two other devices (Uragan-2M and Heliotron E) also have additional TF coils to change the toroidal component of the field and hence the value of ι and its radial profile. H-1 and TJ-II have been specifically designed to study helical-axis plasmas. The degree of indentation of the bean-shaped plasmas and the value of ι are controlled by the ratio of the currents in the TF coils and the central toroidal ring, and both devices have an additional ($\ell = 1$) winding on the central toroidal conductor that allows changing the shear.

Changes in the rotational transform profile can result from net toroidal plasma current produced by heating sources (NBI, ICH, and ECH) and from finite-beta effects (bootstrap and equilibrium currents). Control and compensation of these currents need to be demonstrated. ATF has been designed to allow changing the currents in the VF coil sets so as to preserve the optimized $\iota(\psi)$ profile as beta increases. Control of the ECH $k_{\|}$ spectrum with adjustable mirrors should produce ECH-driven currents in W VII-AS that compensate for other driven currents. Tangential NBI is arranged in opposing pairs (co- plus counter-injection) on both experiments to minimize NBI-driven currents. In ATF, bootstrap currents (typically about one-tenth of those in tokamaks) can be varied over a wide range through changing currents in the VF coil sets and, with the aid of unequal currents in the two independent helical field (HF) windings, can be made zero or reversed in direction. Minimization of bootstrap currents has been taken into account in the design of LHD and W VII-X.

The main issue for stellarators, as for tokamaks, is transport. There is even some similarity in the global scaling laws for the energy confinement time; in the LHD studies, the relation $\tau_E = 0.17 P^{-0.53} n_e^{0.69} B_0^{0.84} \bar{a}^2 R_0^{0.75}$ has been used (where τ_E, B_0, and R_0 are in mks units, P is in megawatts, and n_e is in 10^{20} m^{-3} units), indicating degradation of confinement with increasing power. An improvement in this value by a factor of two to three would be more than adequate for future stellarators. Understanding the mechanisms responsible for transport is more important in stellarators than in tokamaks, where a larger data base exists and large ripple effects are avoided; empirical scaling laws for τ_E will not be adequate for extrapolation to larger devices at lower collisionality. Ion transport appears to be approximately neoclassical, within a factor of three, but electron transport is anomalous. Understanding the edge anomalous transport and its scaling is important for extrapolation to larger devices. Control of the plasma edge (with limiters, gas puffing, edge heating with ECH or ICH, or particle control with a divertor or ergodic regions) may have a large impact on global confinement (as for the tokamak H-mode).

Electric fields play a crucial role in stellarator transport. The resulting $\vec{E}_r \times \vec{B}_T$ poloidal orbit rotation can drastically reduce the direct loss of helically trapped

particles with energies $W \lesssim e\Phi$, where Φ is the potential, and the ripple-induced neoclassical heat conduction at low collisionality. Calculations indicate that these losses can be reduced by more than an order of magnitude. Measurements on W VII-A indicated large radial electric fields in the plasma and corresponding reductions in fast ion losses and ion heat conductivity. Measurement (using probes, Doppler shift of impurity line radiation, and heavy neutral beams or heavy ion beams) and control (with biased plates, electron beams, selective electron heating with ECH, or ion heating with ICH) of the electric field in the plasma will be important in the low collisionality regime. The next-generation experiments, which should reach high temperatures at low density (low collisionality), will contribute most in this area.

The primary issue in the magnetohydrodynamics (MHD) area is determination of beta limits and their scaling with configuration properties. Of most interest are demonstration of direct access to a high-beta second stability region in ATF, CHS, and Uragan-2M; reduction of the equilibrium-limiting outward axis shift through reduction of secondary currents in W VII-AS; and avoidance of flux surface breakup in the nearly shearless H-1 and TJ-II configurations. In addition to beta limits, experimental interest will focus on magnetic islands, fragility and repair of flux surfaces, ergodic regions, and instability mode characteristics and on their relation to enhanced (anomalous) transport. Although demonstration of high-beta operation is needed, this is less of a concern than in the past. Three-dimensional (3-D) MHD equilibrium and stability theory is well developed, and there is already some degree of confirmation by experiment.

There are a number of issues associated with plasma heating. Perpendicular NBI was used in Heliotron E and in W VII-A. The new experiments will use tangential NBI, so beam-driven currents and effects from toroidal momentum input may be important. High-power, long-pulse NBI will also provide a particle source, so control of the plasma density will be an issue, especially at low density. ECH has been used for currentless plasma startup and study of electron heat conductivity. In the new experiments, which have higher-power ECH (0.4–1 MW) with more control over the modes used, ECH will be used to control T_e and the plasma potential, to study electron heat conductivity at lower collisionality, and to study control of impurities in the presence of NBI and ICH. Relatively low-power ICH has been studied with mixed results. The new experiments will use higher-power ($\leq$3 MW), long-pulse (3–30 s) ICH with fast-wave and ion-Bernstein-wave antennas better matched to the plasma shape. Some of these antennas are radially movable for optimizing their coupling to the plasma and can operate over a wide frequency range. The key issues here are determination of the best antenna orientation, optimum heating frequency, adequate coupling to the noncircular plasma as it shifts outward with beta, control of impurities, and loss of energetic trapped particles.

Particle control is another key need for stellarator research, especially for steady-state operation. Pellet injection is the preferred fueling technique because of the central fueling and the observed increase in τ_E that it provides. However, unless an equal particle current is removed at the plasma edge with pump limiters or through the diverted flux layers outside the last closed flux surface, the plasma density will increase uncontrollably. The recycling coefficient is unity in steady state; the walls

do not pump, and gettering cannot be used as in present experiments. W VII-AS has a series of islands at the plasma boundary and no clear diverted flux layer, so it will rely on pump limiters for particle control. Torsatrons (like ATF, CHS, and Uragan-2M) have a helical diverted flux bundle outside the last closed flux surface. The particle current and power density in this layer are very high, so it can be used to channel the outgoing particle flux into an external collector, similar to a tokamak divertor.

Impurity control is also an issue because accumulation of impurities can lead to radiative collapse of the plasma at high density. Active means of impurity expulsion need to be developed for the plasma interior, and the impurity influx to the plasma must be minimized. Increasing the plasma-wall separation and using divertors may help to reduce the plasma-wall interaction. However, both are constrained by the fast radial decay of the necessary higher-order field components in stellarators and by the proximity of the bean-shaped plasma to the central toroidal ring in heliacs.

V.B. Next-Generation Experiments

Next-generation experiments need to address the key issues for the viability of stellarator reactors: beta limits, ripple-induced losses and transport at low collisionality, and long-pulse/steady-state operation. In addition to optimization of the magnetic configuration, these devices must also demonstrate the other ingredients needed for an attractive reactor candidate: effective steady-state heating and heat removal (first wall cooling), effective fuel control, validation of wall materials and in-vessel components, and incorporation of superconducting coils in an experiment.

A major objective for the next-generation experiments will be to demonstrate the potential for true steady-state operation at reactor-relevant parameters. Controlling impurities and particles and maintaining a steady-state, high-beta equilibrium are key issues. Impurities could accumulate in the plasma interior to the point where the associated radiative loss either limits the plasma energy content or causes radiative collapse of the plasma. Equilibrium between erosion of the wall material and redeposition of this material will need to be studied, and the best operating scenario for plasma edge control will need to be developed. There are a number of potential remedies, but they must be demonstrated, especially for application to eventual ash removal in a reactor. Although near-term experiments will have sufficient power to test beta limits at the lower values ($\langle\beta\rangle \lesssim 2\%$), the higher power of the next-generation experiments will probably be required to reach the highest beta values projected ($\langle\beta\rangle = 5$–8%).

For true steady-state operation, steady-state heating and heat removal techniques must be developed. In present-day experiments the duty cycle is $\sim 10^{-3}$, so megawatt heating produces only kilowatt-level average power loading on the walls. In continuous operation, megawatt levels of heat must be continuously removed from the walls. Some of the power will be concentrated on divertor targets or limiters, but a sizable fraction ($\sim 30\%$) will be lost more or less uniformly to the walls and ports by radiation and charge exchange. Erosion of walls and in-vessel components (antennas, limiters, etc.) may be severe in steady-state operation.

Coil construction is another important issue for next-generation stellarators. The main experiments to date have used copper coils and have not had to face issues connected with complex-shaped, high-accuracy, large superconducting coils. The problems of structural support, cooling manifold geometry, and crossovers must be solved. LHD will address these issues for large helical windings, and W VII-X will do the same for modular coils. The coils must be close to the plasma edge in torsatrons to provide high shear at the plasma edge and in modular stellarators to restrict the toroidal excursions and complexity of the modular coils. On the other hand, control of the plasma density with collectors in the diverted flux bundle or in stationary islands at the plasma edge requires adequate space between the last closed flux surface and the vacuum wall. This conflict has caused the W VII-X group to increase R_0 from 5 m to 6.5 m and reduce B_0 from 4 T to 3 T, the LHD group to choose $M = 10$ over $M = 14$, and the ATF-II group to favor thinner, high-current-density windings. Development and validation are needed for force-flow, high-current-density, high-field, stable, NbTi/Cu cable-in-conduit conductors.[26]

In the longer term, issues associated with deuterium-tritium (D-T) operation will become important: alpha-particle losses, alpha-particle effects on MHD stability and transport, control of the burning plasma, and helium ash removal. Some information will be gained on these issues in the next-generation experiments, but they can only be fully addressed in a later D-T device.

VI. REACTOR STUDIES

Studies of reactors are not as developed as in the tokamak area. In addition to the much larger resources devoted to tokamaks, there is the expectation that the qualitative similarity of stellarators and tokamaks will allow adoption of many of the generic system features and component designs developed for tokamaks. Other factors are the wider range of coil designs available for stellarators and the lack of a verified "optimum" design. A major goal of the large next-generation stellarators described in Sec. V is to provide sufficient information to determine an optimum configuration on which to base the design of a follow-on D-T experiment and more realistic reactor studies. In the meantime, point designs and scoping studies have been done for the Wendelstein,[27] Heliotron,[28] ATF,[29] and Uragan[30] approaches in order to understand the consequences of the differences in these approaches.

VII. SUMMARY

The large variety of possible stellarator configurations is both an asset and a liability. It is an asset because the large variety allows selection of an optimum mix of magnetic configuration properties that is not dependent on an internal, driven plasma current and is under external control for steady-state operation. However, the large variety of configurations and their inherently 3-D nature make it more difficult to determine an optimum. This is the reason for the four more or less orthogonal stellarator approaches that are being followed and for the large degree of cooperation and complementarity in the world stellarator program.

The present set of medium-size ($\bar{a}$ = 0.2–0.27 m, P = 2–4 MW) copper-coil experiments aims at demonstrating the principles of configuration optimization for the four approaches and at extending stellarator plasma parameters to the higher values of T_i (a few keV) and beta ($\langle\beta\rangle$ = 2–5%) where the key issues of low collisionality ripple transport, impurity control, and MHD beta limits can be explored. The next-generation devices now under study would be larger ($\bar{a} \simeq 0.5$ m, P = 10–30 MW) superconducting-coil devices whose aim would be to demonstrate optimized steady-state magnetic configurations at more reactor-relevant plasma parameters (T_i = 5–10 keV, $\langle\beta\rangle > 5\%$).

The pursuit of a few complementary approaches gives the best chance for the understanding needed for success. If this combined approach is successful, then stellarators offer a fusion reactor option without the drawbacks of tokamaks (driven current, pulsed operation, disruptions, etc.), but with many of their advantages (good confinement based on toroidally nested, closed flux surfaces; high beta limits; and low to moderate aspect ratio, hence reasonable values for wall power loading and power output). The next decade of stellarator research on the facilities described in this issue will determine whether the vision foreseen in the 1981 cartoon[31] in Fig. 7 will be realized.

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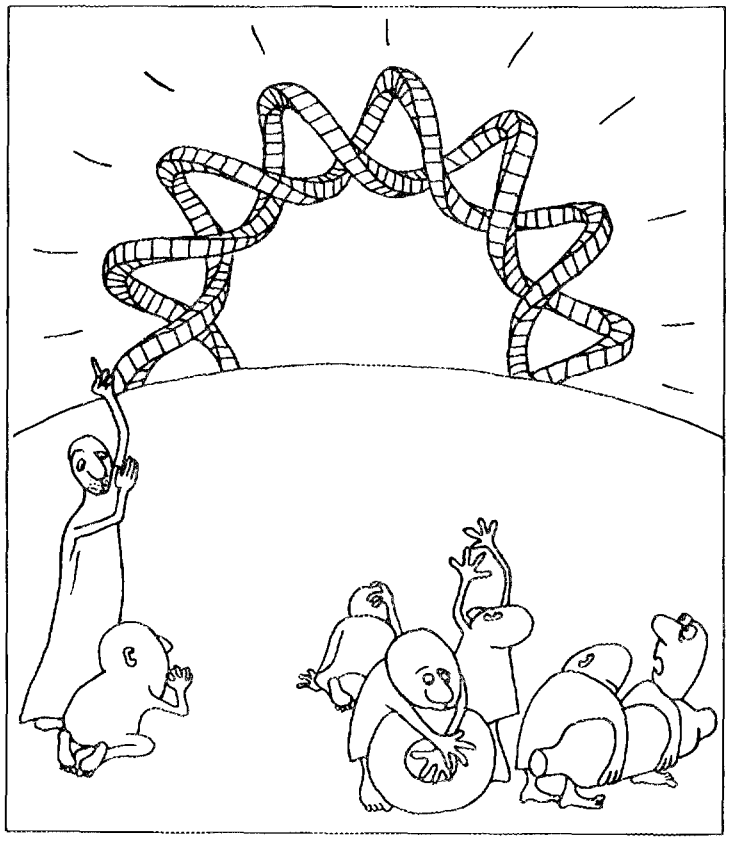

Fig. 7. A vision of the potential of the stellarator concept.

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