U.S. STELLARATOR PROGRAM PLAN

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U.S. STELLARATOR PROGRAM PLAN

The U.S. stellarator community proposes a significant expansion in the national stellarator program to capitalize on recent innovations in the stellarator concept that could lead to a more attractive fusion power plant, fill a serious gap in the world stellarator program, and offer unique opportunities for fusion science studies. The goal of the expanded program is to develop the knowledge base needed for compact, high-beta, good-confinement stellarators.

Stellarators are an Essential Part of an Innovative U.S. Fusion Science Program. Stellarators have the potential for an attractive reactor featuring inherently steady-state, disruption-free operation; low recirculating power; and good confinement and beta. Because of this, stellarators are a large part of the world fusion program with large experimental investment and substantial performance. Stellarators have a magnetic topology similar to that of tokamaks. However, while tokamak configurations have two-dimensional symmetry, stellarator configurations are fully three dimensional (3-D). The extra dimension makes available a richly diverse set of configurations, providing significant additional design freedom that can be used to optimize for fusion performance or for studies of particular plasma physics properties at minimum cost. Control of the q-profile, the bootstrap current, and the radial electric field is possible using external coils. These capabilities are complementary to the advances of the axisymmetric tokamak program and allow novel solutions to some of the problems of developing advanced toroidal configurations, particularly disruptions and current drive. An expanded stellarator program provides the opportunity for synergy, combining the physics understanding developed in the tokamak program with the demonstrated control and design advantages of stellarators, potentially shortening the reactor development path and broadening the U.S. Fusion Energy Science Program.

As part of an innovative U.S. program to improve the attractiveness of fusion reactors and decrease development costs, the U.S. stellarator program is focused on compact configurations with reduced transport and with beta limits at least as high as those in the advanced ARIES tokamak reactor studies.

The International Stellarator Program is Focused on Large-Aspect-Ratio, Currentless, Nonsymmetric Configurations. The largest new fusion facilities are stellarators: the Large Helical Device (LHD) now operating in Japan and the Wendelstein 7-X (W7-X) under construction in Germany are $0.5- to $1-billion facilities that feature superconducting coils. These facilities are designed to demonstrate steady-state disruption-free stellarator operation and a level of performance that allows extrapolation to devices capable of burning plasma operation. The large world stellarator program has contributed to development of the computational techniques for configuration optimization and has overcome earlier concerns about coil complexity. However, there are gaps in the world program that present opportunities for innovation. The non-US stellarators, including smaller experiments, have plasma aspect ratios ranging from 5 to 11 and extrapolate to very large reactors; low aspect ratios (<5) are unexplored. The W7-X experiment was explicitly designed to minimize the bootstrap current, while LHD is expected to have bootstrap current smaller than that of a comparable tokamak because of its magnetic structure. None of the non-US stellarator devices takes advantage of the bootstrap current, magnetic symmetry, or drift-orbit omnigeneity in its design strategy. The large non-US stellarator programs will extend stellarator research to new levels of size and performance, but will not cover the full range of issues important for compact stellarator development.

Opportunities and Key Issues for a U.S. Stellarator Program. Recent development of two new confinement-optimized configurations holds the promise that a low-aspect-ratio stellarator, which would allow a more compact stellarator reactor, can be developed with good confinement and high beta. These configurations make use of the self-generated bootstrap current, which allows potentially higher equilibrium and stability beta limits than can be generated otherwise in low-aspect-ratio stellarators, while relaxing some of the constraints on the external coils. However, these configurations require investigation of the helical fields required from external coils to prevent the kink instabilities and disruptions observed in tokamaks. In addition, the theoretically predicted beta limits, the reduction of neoclassical transport through magnetic symmetry or omnigeneity, the role of higher plasma flow shear in reducing anomalous transport, compatibility of the bootstrap current with required profiles, startup, and power and particle handling must be demonstrated in these new configurations. In short, promising compact stellarator concepts have been developed to the point of readiness for experimental testing.

The United States Should Invest More Resources in Stellarator Research. The United States. should undertake a ten-year proof-of-principle program to develop the knowledge base needed for compact, high-beta, good-confinement stellarators. To do so would fill the gap in the world program and would further key aims of the U.S. fusion program: confinement concept innovation, fusion science understanding, and plasma physics advancement. These aims can be advanced at modest cost; the immediate needs can be met with investments far less than those of LHD and W7-X.

The proposed National Stellarator Research Program. An integrated program of experiment, theory, and systems studies is planned. It will consist of well-coordinated research drawing on several elements: (1) a flexible, reconfigurable proof-of-principle facility; (2) a new concept exploration experiment; (3) the present Helically Symmetric Experiment (HSX) at the University of Wisconsin; (4) experimental collaboration with the international stellarator program in specific areas; (5) theory focusing on concept optimization and key stellarator issues; and (6) systems studies to guide the concept optimization tradeoffs. The six program elements, which cross-link with each other to provide a coherent, well-integrated program, are depicted schematically in Fig. 1. The proof-of-principle facility must be reconfigurable to ensure that experimental tests of the new developments coming out of the program can be conducted expeditiously.

At present, two promising transport optimization strategies for compact stellarator design have been developed theoretically: quasi-axisymmetry and quasi-omnigeneity. Both make use of the bootstrap current, but to different degrees, to make a more compact configuration than the currentless W7-X. Both look attractive for compact stellarator reactors, but each has distinct complementary advantages. Both must be developed experimentally to maintain the broadest possible scientific base for the program’s ultimate success. A determination of which is the better strategy will be one of the program’s goals. The new proof-of-principle facility will provide sufficient plasma performance and machine capability for integrated testing of a compact stellarator configuration with high beta and bootstrap currents that can form the basis for extrapolation to more reactor-relevant performance. In order to minimize cost, it is desirable to take advantage of an existing facility, the PBX-M tokamak, and modify it for the proof-of-principle stellarator tests. The quasi-axisymmetric concept is more compatible with the PBX-M constraints, so this configuration will be chosen for the initial tests. The facility will be modified to test improved configurations as they are developed by the program. A new concept exploration facility will be constructed to test the basic optimization principles of quasi-omnigeneity. The HSX, nearing operation, will be the first test of improved neoclassical transport and reduced parallel viscosity in quasi-symmetry. The HSX will also investigate high effective transform and very low plasma currents, features not covered in the compact stellarator PoP program. Other small-scale supporting experiments are needed to investigate specific scientific and technical issues in support of the compact stellarator PoP program.

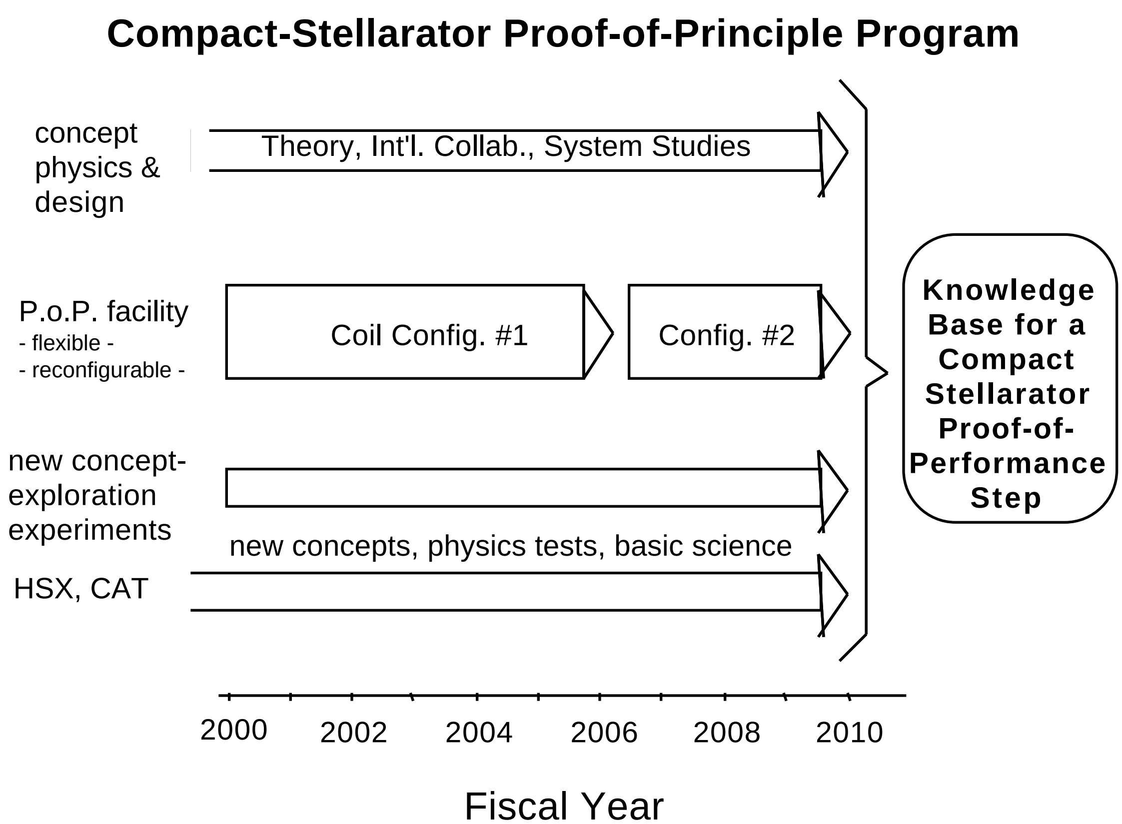

Fig. 1. Road map for the proposed stellarator proof-of-principle program.

The program will include collaboration with the international program in focused areas both to understand key issues for extrapolation of stellarator performance and to test optimization features at higher plasma parameters. A robust theory and concept optimization effort will continue to develop new configurations, incorporate physics advances from other parts of the program, and develop understanding of experimental results from US and international experiments. Systems studies will guide the fusion optimization tradeoffs in concept development, e.g., cost/benefit trade-offs between aspect ratio, beta limit, and confinement improvement; limits on acceptable energetic-particle orbit losses; and integration of physics optimization with reactor optimization considerations and constraints.

Anticipated program costs are summarized as follows: (1) proof-of-principle facility, in the range of $35 million to construct and $20 million/year to operate; (2) HSX, $1.6 million/year; (3) new concept exploration experiments, $2 million/year; (4) international collaboration, $1.5 million/year; (5) theory, $3.5 million/year; and (6) system studies, averaging $1 million/year The program budget will reach a plateau level of $30 million/year All elements of the program are necessary to adequately develop the concept and ensure proper balance.

This program will build on the substantial data base in stellarators and tokamaks and will make important contributions to the world fusion program. If the proposed plan is carried out, in ten years the resulting knowledge base will be sufficient to permit comparisons with steady-state tokamak-based power-plant designs and will provide a basis for a decision on proceeding to the next step, a proof-of-performance program to study more reactor-relevant plasmas in a compact stellarator configuration, possibly with D-T capability.

The proposed program is an exciting opportunity for the U.S. to develop the compact stellarator concept as part of the innovative Fusion Energy Science Program.

I. STELLARATORS ARE AN ESSENTIAL PART OF AN INNOVATIVE U.S. FUSION SCIENCE PROGRAM

Stellarators confine plasma within nested toroidal magnetic surfaces composed of helical magnetic field lines. In that respect, the topology of stellarators is similar to that of tokamaks and toroidal pinches. However, their physical geometry is quite different. In tokamaks the poloidal field coils are symmetric in the toroidal direction, while stellarators use external coils that are non-uniformly deformed or displaced as the toroidal angle varies to generate the confining poloidal magnetic field. While tokamaks are two-dimensional, stellarator configurations are three dimensional (3-D). The extra dimension makes available a richly diverse set of configurations for varying the interactions between the plasma and the confining magnetic field. This provides significant additional design freedom that can be used to optimize the configuration for attaining desired fusion performance or for studying particular plasma physics properties at minimum cost.

These advantages potentially provide ready solutions to some of the most challenging problems facing tokamaks: operating in steady state and eliminating disruptions at high β and high density.

I.A. Reactor Advantages

The various stellarator configurations available have distinct advantages over other magnetic confinement concepts and offer low-risk solutions to many of the challenging problems of tokamak reactors.

1. Stellarators eliminate or greatly reduce the need for externally driven plasma current, reducing the recirculating power in a reactor. Plasma control is simplified because the confining fields are controlled with external coils instead of current-drive techniques that rely on coupling power inside the plasma itself. The maintenance of the confining fields in a stellarator is based on well-tested physics and provides a clear path to steady-state operation.

2. Disruptive discharge termination and quench of the plasma current are not observed in stellarator experiments, even at the theoretical β limit. Even in stellarators experiments with large plasma currents, the external fields stabilize the configuration preventing disruptions when the externally generated transform is at least 15% of the total [1]. This allows stellarators to robustly access their full β limit in steady state, reducing the required magnetic field and plasma size.

3. Empirically, the density in stellarators is limited only by power balance, not by disruptions or edge instabilities (e.g. MARFEs) as found in tokamaks. For equivalent configurations, the empirical density limit is higher in stellarators [2], allowing an optimal reactor burn-point. Since edge current drive is not required, operation with high edge density can be used to simplify divertor design.

4. The external magnetic field for a stellarator can be created by a single set of modular non-interlocking coils, simplifying the reactor construction and maintenance. Such coils have been developed and utilized on existing experiments [3], and will be used on the superconducting W7-X [4].

5. A wide range of aspect ratios is theoretically available, allowing the configuration to be optimized to match the plasma power density to the available edge power handling technology.

6. Stellarator designs offer high β values, similar to the aggressive ARIES tokamak power plant designs ($\beta \geq 5\%$) [5]. Due to their lack of disruptions, stellarators offer the potential for reliable operation at high β in compact reactors.

Due to these advantages, including established solutions to many problems challenging tokamak reactors, stellarator configurations offer the possibility of a reduced development cost, reduced development risk path for fusion energy. This is why stellarators are an important part of the world program to develop confinement systems for an attractive fusion power source.

I.B. Earlier Stellarator Issues Have Been Largely Overcome

Earlier stellarator designs and experiments led to significant concerns about confinement at low collisionality, the achievable beta limit, and coil complexity. Through patient and continuing research, these earlier issues have been understood and largely resolved!

Development of new theoretical tools has led to stellarator designs with good neoclassical confinement. Earlier designs had large helical-ripple-induced neoclassical losses at low collisionality. These losses are greatly reduced (in some cases to below those in a comparable tokamak) in newer stellarator designs such as the Helias approach used in W7-X [4], the quasi-helically-symmetric approach used in HSX [6], and the quasi-axisymmetric [7] and nonsymmetric quasi-omnigeneous configurations [8] being developed for low-aspect-ratio stellarator candidates in the U.S. In addition, anomalous transport has been reduced in present stellarator experiments (improved confinement modes) and techniques developed for tokamaks are being applied to stellarators. The newer designs optimized to reduce neoclassical transport should be particularly suited for confinement improvement due to electric field ($E \times B$) flow shear.

Present stellarators exceed simple estimates of beta limits, and novel configurations show promise of higher beta. The simple equilibrium beta limit criterion based on the shift δ of the magnetic axis equal to half the average plasma radius a_p has been exceeded in Wendelstein 7-AS (W7-AS) where $\delta/a_p = 2/3$ was obtained without a significant change in the plasma behavior, even when the outer 1/3 of the plasma was theoretically resistively unstable [9]. In addition, the Mercier instability criterion has been exceeded over most of the plasma radius in the Compact Helical System (CHS) experiment at the highest beta values achieved in stellarators ($\beta = 2.1\%$) without a significant change in plasma confinement [10]. In both cases the achievable beta is limited by the available heating power and transport rather than by an observed stability limit. New designs with significant self-generated bootstrap currents are predicted to allow higher equilibrium and stability beta limits.

New computational techniques and modern numerically controlled machines have made fabrication of complex stellarator coils routine, as demonstrated by successful construction of the modular-coil stellarators HSX (R_0 = 1.2 m, B_0 = 1.4 T) and W7-AS (R_0 = 2 m, B_0 = 3 T). The accuracy achieved in W7-AS was a few parts in 10^4 [3]. These techniques are now being used for the large superconducting-coil W7-X stellarator (R_0 = 5.5 m, B_0 = 3 T). Similar accuracy has been achieved in construction of the Large Helical Device (R_0 = 3.9 m, B_0 = 4 T) designed for 1.6 GJ of stored magnetic energy [11]. The complex W7-AS vacuum vessel was also fabricated from flat plates using similar techniques.

The remaining concern, that stellarator power plants may be too large, could be addressed in the U.S. program. The most developed confinement-optimized stellarator reactor concept, HSR [12], has R_0 = 22-24 m. U.S. reactor designers would to reduce R_0 for an economical fusion power plant by a factor of ~4. Recent developments in the U.S. of new confinement-optimized configurations hold the prospect that a low-aspect-ratio can be developed with good confinement and higher beta. This could result in a reduced development cost and a more compact stellarator reactor. Given the complementary nature of the world stellarator program and the large resources being devoted to that program, the U.S. could have the leading role in development of more compact stellarator configurations.

I.C. Design Flexibility for Concept Innovation and Toroidal Physics Studies

Stellarators offer a much broader range of plasma configurations than do tokamaks. This flexibility is an advantage for solving design problems and focusing experiments on specific issues. As discussed in Section III, a quasi-axisymmetric stellarator has tokamak-like plasmas but with a significant fraction of the rotational transform produced by external coils. A true tokamak is the limiting case of zero fraction. What fraction is required to eliminate disruptions at high β? Non-symmetric omnigeneous stellarators, like W7-X, can have the bulk of the trapped particles located in a region of good magnetic curvature. What effect does this have on the anomalous transport? Local shear is known to be an important parameter in the theory of anomalous transport, ideal magnetohydrodynamic (MHD) instabilities, and neoclassical tearing modes. Stellarator configurations offer a much broader range of local shear profiles than tokamaks, allowing ‘reversed’ shear across the entire profile and external coil control of the shear profile.

Two basic features, configuration control and access, make stellarators favorable for studying the implications of the magnetic configuration and for improving the attractiveness of fusion power. These features imply that stellarator experiments of various scales are needed: (a) integrated tests of steady-state plasma performance (e.g., LHD and W7-X), (b) integrated tests of plasma performance in attractive configurations (so called proof-of-principle experiments), (c) tests of specific stellarator issues and concepts (e.g., HSX), and (d) tests of fundamental physics issues. The flexibility and recent rapid advancement of the stellarator concept also implies the need for an ongoing optimization effort. As our understanding of toroidal plasma physics improves, the flexibility of the stellarator allows that knowledge to be exploited to produce configurations that are even better adapted to fusion applications. A strong theory program is required to interpret, integrate, and extrapolate the increased knowledge of toroidal plasmas that will come from the stellarator program.

I.D. Synergy with the Tokamak Program

Stellarators share much of the physics basis of other toroidal systems, such as tokamaks, and can build upon and contribute to the developments in the larger tokamak program. Many of the instabilities and configuration issues (e.g. transport control, divertors) expected in stellarators have been extensively studied in tokamaks. This shared physics basis aids the comprehensive analysis and evaluation of stellarator configurations, building upon the knowledge gained in both configurations. It also gives confidence that the flexibility of stellarators can be optimized to provide novel solutions to the challenging problems that have arisen in the study of the tokamak, such as high-β disruption control and the need for current drive. Finally, the three-dimensional flexibility available with stellarators allows the design of experimental configurations for testing elements of toroidal plasma physics that are not accessible in tokamak configurations, such as locating the trapped particles in good curvature regions.

I.E. Contributions to 3-D Plasma Science

Uniquely among toroidal confinement systems, stellarators are intrinsically three dimensional as are the magnetosphere, free-electron lasers, accelerator transport lattices, and perturbed axisymmetric tokamaks. These fields have developed synergistically, with stellarators driving the development of 3-D plasma physics. Methods for reducing field line stochasticity in stellarators have been adapted to reduce chaos and improve performance in storage rings [13]. Particle chaos due to crossing the current sheet in the magnetotail has been analyzed by methods developed to study transitioning orbits in stellarators [14]. Recent methods for optimizing stellarators may find application in helical-wiggler free-electron lasers. Electron orbits in astrophysical systems are studied using the magnetic coordinate and drift Hamiltonian techniques developed for stellarators [16]. Within fusion, resistive wall modes and field error effects in tokamaks are 3-D equilibrium problems. 3-D effects provide fundamental limits on the performance of nominally axisymmetric devices like tokamaks and reversed field pinches (RFP's). Transport and particle losses due to symmetry breaking had a natural development in the context of stellarators. The further development and understanding of stellarators will continue to contribute to, and benefit from, the understanding of other 3-D plasma systems.

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II. THE FOCUS OF THE WORLD STELLARATOR PROGRAM

Development of the stellarator concept is being actively pursued in several countries. The largest fusion facilities now under construction are stellarators: LHD in Japan and W7-X in Germany are $0.5-1 billion facilities with superconducting coil systems. These experiments are supplemented by more moderate-size (under $100 million) research facilities presently in operation in Japan (CHS and Heliotron E), Germany (W7-AS), Spain (TJ-II), Australia (H-1), etc. The proposed U.S. stellarator proof-of-principle program would complement the existing world stellarator program by adding an important element – research on compact stellarator configurations – that will be of value to all countries.

The main parameters for the present stellarator experiments are given in Table 1. Here R_0 is the major radius, a_p is the average plasma radius, V_p is the plasma volume inside the last closed flux surface (LCFS), B_0 is the magnetic field on axis, P is the plasma heating power, and t_{exp} is the pulse length.

Table 1. Major Device Parameters for Operating and Near-Term Stellarators

Experiment	Location	R_0 (m)	a_p (m)	V_p (m^3)	B_0 (T)	P (MW)	t_{exp} (s)
Large-Next Generation Experiments							
LHD (1998)	Japan	3.9	0.5–0.65	30	3 (4)	30	10 – ∞
W 7–X (2005)	Germany	5.5	0.52	30	3	30	10 – ∞
Medium-Size Foreign Experiments							
W 7–AS	Germany	2.0	0.2	1.6	2.5	4	3
CHS	Japan	1.0	0.2	0.8	2	2	1
Heliotron E	Japan	2.2	0.2	1.7	2	4	0.2
TJ–II	Spain	1.5	0.22	1.4	1	1	0.5
H–1	Australia	1	0.21	0.9	1	0.2	0.2
U.S. Experiments							
HSX (1998)	U. Wisc.	1.2	0.15	0.53	1	0.2	0.1
CAT	Auburn	0.53	0.1	0.11	0.2	0.007	120

II.A. The International Stellarator Program Is Focused on Large-Aspect-Ratio, Minimal-Plasma-Current, Nonsymmetric Configurations

In Japan, the focus is on helical-coil systems optimized to have a magnetic well that deepens with increasing beta, small net plasma current, and aspect ratio half that of W7-AS or W7-X. The medium-size CHS [1], which will cease operation in 1999, is focusing on transport, MHD, divertor studies, and diagnostic development for the next step, LHD [2], which started operation in March 1998. Figure 1 shows the upper half of the structural shell being lowered onto the LHD vacuum chamber with its two large helical coils. The lower poloidal field coils are below the vacuum vessel in the bottom half of the cryostat, and the upper poloidal field coils are installed on top of the structural shell. The outer diameter of the vacuum vessel is 11.5 m; the cryostat into which the vacuum vessel and superconducting coil sets are inserted has a diameter of 13.5 m. Part of the middle section of the cryostat is shown to the right of the vacuum vessel. The two large helical windings, the three sets of poloidal field coils for plasma positioning and shaping, and the plasma surface (LCFS) for the ten-field-period LHD are illustrated in Fig. 2. The cross section of the non-circular, nonaxisymmetric LHD plasma is basically an ellipse that rotates toroidally around a planar circular magnetic axis. The magnetic configuration has moderate shear, with $\iota\!\!\!-(0) \approx 0.3$ and $\iota\!\!\!-(a_p) \approx 0.9$ where $\iota\!\!\!-$ (= $1/q$) is the rotational transform and q is the tokamak safety factor. The CHS magnetic configuration is similar to that of LHD, but has eight field periods toroidally instead of ten, has somewhat smaller plasma aspect ratio ($A = R/a_p = 5$, vs 6 for LHD), and is not as well optimized for energetic orbit confinement in the plasma core as LHD.

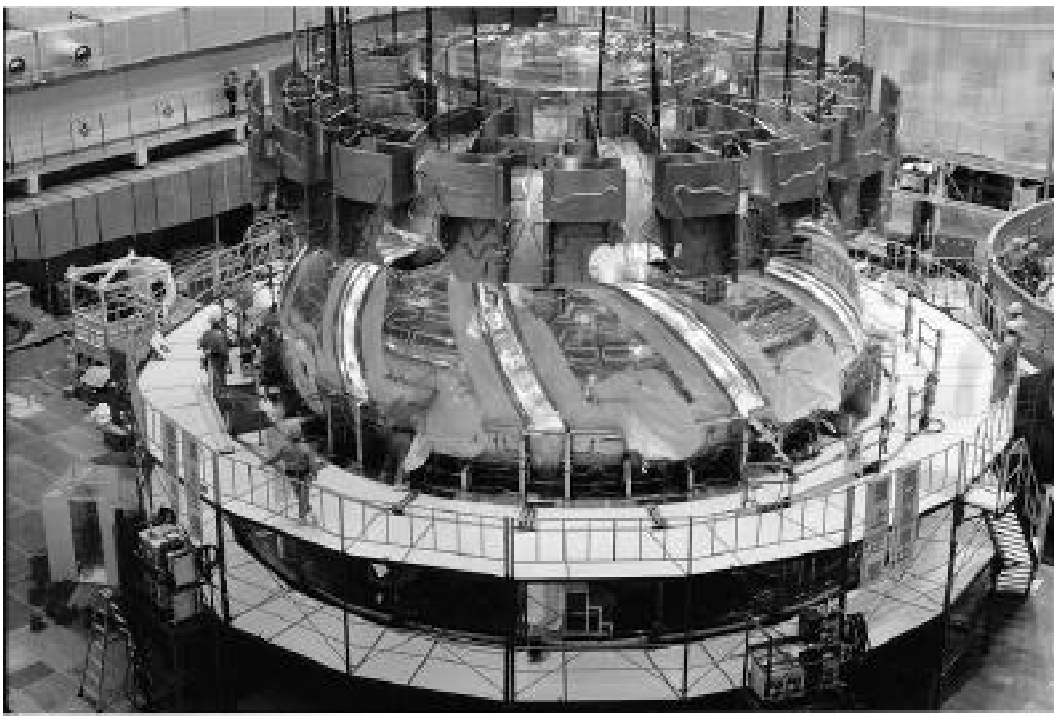

Fig. 1. The large superconducting helical device LHD under construction (1997).

In Germany, the focus is on the modular-coil “Helias” approach [3] in which the magnetic field is carefully designed to satisfy a particular set of physics optimization criteria, including a beta limit of 5%, low neoclassical transport, and low bootstrap current. The large W7-X [4], scheduled to begin operation in 2005, will be the key test of this approach. Figure 3 shows the modular non-planar coils and the LCFS for W7-X. The plasma cross section changes from elliptical to a tear-drop shape to triangular as the plasma rotates toroidally around a helical magnetic axis. W7-AS [5], which will operate through 2001, is testing some of the Helias physics elements and developing techniques for W7-X. The W7-AS configuration differs from that of W7-X in that W7-AS has a planar (but noncircular) magnetic axis, a different shape for the LCFS, and lower rotational transform: $\iota\!\!\!-(0) \approx \iota\!\!\!-(a_p) = 0.4\text{-}0.5$. Both W7-AS and W7-X are nearly currentless and have large plasma aspect ratio ($A = 10\text{-}11$).

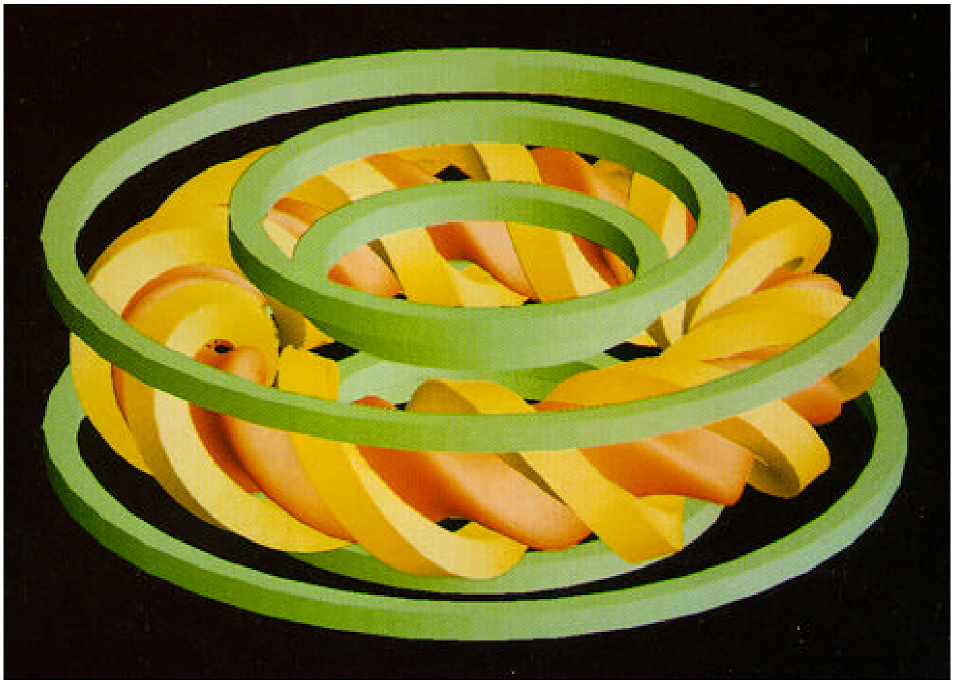

Fig. 2. The last closed flux surface, the helical windings, and the poloidal field coils for LHD.

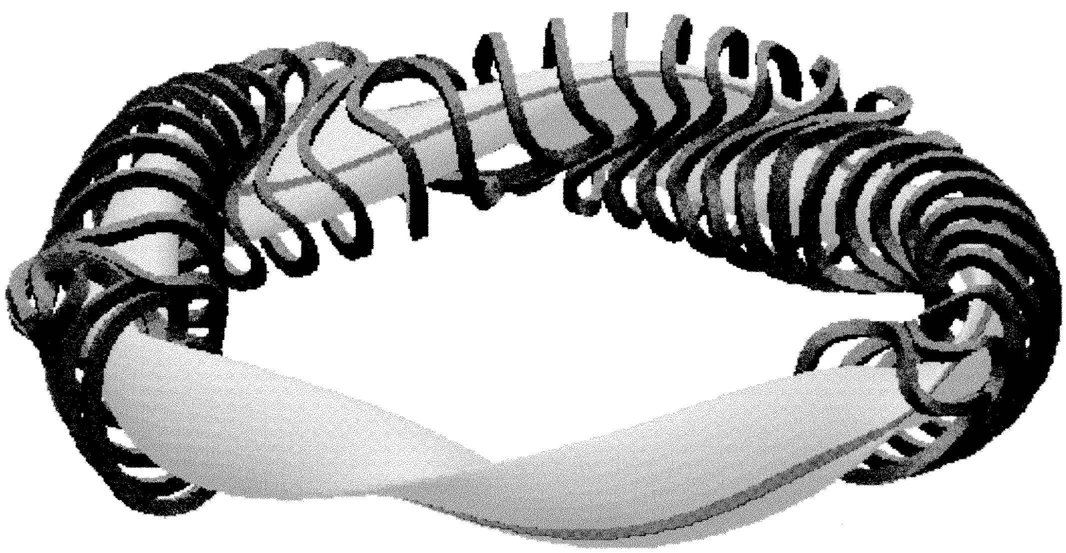

Fig. 3. The modular nonplanar coils and the last closed flux surface for W7-X.

In Spain and Australia, the focus is on the "flexible heliac" being studied in the TJ-II [6] and H-1 [7] experiments, respectively. In this concept, a set of planar circular coils whose centers follow a helical path about a linked central conductor creates a bean-shaped plasma that rotates toroidally around a helical magnetic axis, as shown in Fig. 4. These configurations have low shear but higher rotational transform: $\iota(0) \approx \iota(a_p) = 1.2\text{-}1.6$ for TJ-II. The TJ-II and H-1 programs emphasize transport and beta limits for this configuration.

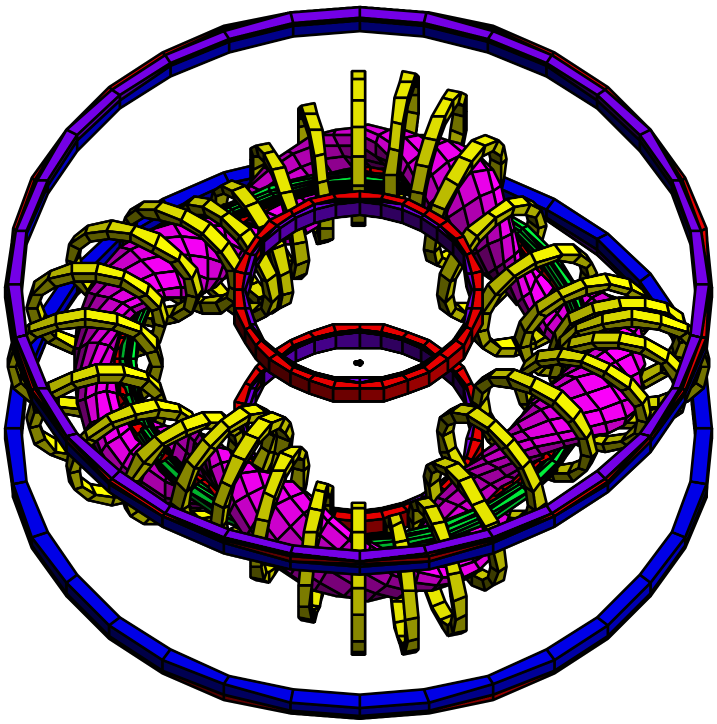

Fig. 4. The coil sets and the last closed flux surface for the TJ-II flexible heliac.

II.B. The High-Aspect-Ratio Long-Pulse LHD and W7-X Devices Will Focus on High Performance at Low Plasma Current

LHD and W7-X are designed to demonstrate steady-state disruption-free stellarator operation and a level of performance (volume-average beta $\beta \geq 5\%$, ion temperature $T_i \sim 10$ keV, energy confinement time τ_E of hundreds of ms, etc.) that allows extrapolation to devices capable of burning plasma operation. The large stored magnetic energy (up to 1.6 GJ) and energy input during a discharge (~300 MJ for 10-s operation, ~3.6 GJ for 20-minute operation) in LHD indicate the scope of these experiments and the technological difficulties to be addressed in these programs. LHD and W7-X will address issues that are crucial to demonstrating the viability of the stellarator confinement concept and extrapolation of stellarator performance. LHD and W7-X will provide unique data on both helical and magnetic-island-based divertors, high-power plasma heating, and superconducting-coil operation that will be relevant to all design approaches. Collaboration on these experiments will therefore be an important element of the U.S. stellarator program.

Both LHD and W7-X can also provide basic tests of physics and optimization principles at high plasma parameters that can be used for stellarator concept development in the U.S. aimed at a more compact, high-beta disruption-free reactor concept. However, LHD and W7-X have plasma aspect ratios of 6 and 10.5, respectively, and extrapolate to very large reactors. In addition, W7-X was explicitly designed to minimize the bootstrap current, while LHD is expected to have bootstrap current much smaller than a comparable tokamak because of the combination of toroidal and helical curvature. Incorporation of the bootstrap current is an important element in the optimization of the low-aspect-ratio stellarator configurations proposed for the U.S. program.

II.C. The Medium-Scale Foreign Stellarators Focus on Complementary Physics Studies in Stellarators with a High Degree of Nonaxisymmetry

Because of their different missions, there is a large range in capabilities for the world stellarators. Although the medium-scale experiments have much smaller magnetic stored energies (by a factor of ~300) and energy input during a discharge (by a factor of ~30-1000), they nevertheless allow relevant physics studies with interesting plasma parameters. For example, W7-AS obtains T_e = 5 keV, T_i = 1.6 keV, β = 1.8%, and τ_E > 50 ms in different operating regimes. The wide range of stellarator configurations accessible on CHS, Heliotron E [8], W7-AS, TJ-II, H-1, etc. allows study of the role of aspect ratio (from 5 to 11), helical axis excursion, magnetic-island-based divertors, and the consequences of a net plasma current, elements that are being incorporated in the low-aspect-ratio stellarator concepts under consideration in the U.S. program. These experiments will make significant contributions to understanding of the stellarator concept and provide information of importance to the U.S. program. However, none of the non-US stellarator experiments incorporates magnetic symmetry or plasma current in their design strategies, so there are important opportunities available for innovative stellarator configurations in the U.S. program.

II. D. Three Important Gaps in the World Stellarator Program

From the preceding summary of the world stellarator program, there are three areas in which there is an important gap in the world stellarator program:

1. smaller aspect ratio for a more compact stellarator configuration;
2. incorporation of the bootstrap current in the optimization; and
3. quasi-symmetric magnetic fields or a magnetic field configuration in which the particle orbits exhibit omnigeneity.

Aspect ratio

Stellarators tend to have large aspect ratio. W7-AS, Heliotron E, and W7-X have an aspect ratio of ≈11. At the lower end of the spectrum, the aspect ratios are ≈7.5 for TJ-II, ≈6 for LHD, and ≈5 for CHS and H-1.

From a simple point of view it can be understood why stellarators historically have had large aspect ratio. The equilibrium beta limit, for example, scales as $\beta \propto \iota^2/A$. Because ι tends to scale with A, the equilibrium beta limit scales with aspect ratio as well. At high aspect ratio, the symmetry-breaking terms due to toroidal curvature are small, hence the neoclassical losses are reduced. In the low-collisionality $1/\nu$ regime, for example, the thermal conductivity scales as $1/A^2$. Also at low collisionality, large symmetry-breaking terms can lead to direct orbit loss; trapped particles leave the confinement region in a time shorter than the collision time. Finally, flux surfaces for conventional stellarators at higher aspect ratio have tended to be much more robust than at lower aspect ratio. Typically, the symmetry-breaking terms at lower aspect ratio destroy the outer flux surfaces as the aspect ratio of the coils is decreased.

This simple picture however is not necessarily an accurate one. The W7-AS experiment (with $A \approx 10$) has an equilibrium beta limit of 4.5% compared to the old W7-A device (with $A \approx 20$) which had a beta limit of 2%. Appropriate shaping of the plasma boundary allows for a reduction in the

Pfirsch-Schlüter currents and the resulting Shafranov shift in W7-AS. The decreased shift of the magnetic axis with beta has been demonstrated experimentally [9].

Recent theoretical and computational breakthroughs have further shown the simple picture to be incomplete. A good example of this is the quasi-toroidally symmetric MHH2 [10] configuration which has an aspect ratio of 3.5, an edge rotational transform of 0.4, an equilibrium beta limit of about 5%, and neoclassical transport considerably lower than a conventional stellarator at this aspect ratio. Other low-aspect-ratio configurations that allow for a self-consistent bootstrap current (QAS [11] and SMARTH [12]) have demonstrated theoretically how the rotational transform due to the plasma current can add to the external transform and increase the equilibrium beta above 5%. Very large equilibrium betas can be achieved in quasi-helical stellarators like HSX [13], which has an effective rotational transform $\iota_{\text{effective}} = m\iota - N \sim 3$ and an equilibrium beta limit of 35-50%. Here N is the toroidal mode number and m is the poloidal mode number. However, quasi-helical stellarators have not been developed at aspect ratios less than about 5-6, where the symmetry-breaking toroidal curvature becomes strong.

Plasma current

Up until the early 1980's, stellarators routinely operated with ohmic current. At the time, stellarator researchers discovered what was called drift-parameter scaling; the confinement degraded with increasing ratio of electron drift velocity to thermal speed. Although it was not understood then, this result was a manifestation of L-mode scaling and not an inherent advantage over tokamaks. What was clear, however, was that stellarators with significant external transform did not disrupt. With the widespread use of gyrotrons, currentless stellarators became the norm. The ultimate route of the currentless stellarator lies in the direction of W7-X, which has been optimized to have minimal bootstrap current (among other constraints). The rationale behind this optimization principle is that the bootstrap current can be dangerous in a low-shear system, possibly giving rise to large low-order island structures that can devastate confinement.

It is time, after about 15 years of relative inactivity in this area, to take a second look at plasma current in stellarators. The advantage of the plasma current in low-aspect-ratio stellarators is that it provides additional rotational transform that contributes to higher equilibrium beta limits. This in turn relaxes some of the constraints on the external coils with regard to the current density and torsion. More importantly, bootstrap currents are an intrinsic attribute of quasi-symmetric configurations; it is the combination of helical and toroidal curvature in W7-X that minimizes the bootstrap current on each flux surface. For a quasi-axisymmetric configuration, the magnitude of the bootstrap current is naturally on the order of that in a tokamak and flows in a direction such as to increase the transform. For a quasi-helical configuration, the bootstrap current is roughly an order of magnitude lower (due to the very high effective transform), but flows in a direction such as to lower the vacuum transform. The bootstrap current in quasi-omnigeneous configurations can be small or lie somewhere in the middle, depending on the magnetic field spectrum.

Based on recent tokamak results and theoretical calculations of MHD stability and neoclassical tearing modes, there are additional reasons for considering plasma current. Theoretical calculations by Cooper [14] indicate that the hollow bootstrap currents generated by peaked plasma profiles in stellarators can increase stability to infinite-*n* ballooning modes. Recent calculations have shown that ballooning stability, which tends to be in the range of 2% in most conventional stellarators (up to 5% maximum expected in W7-X) can be in the range of 6-7%, possibly higher, in a low-aspect-ratio device with a large hollow current profile [15]. Another advantage of allowing for a large bootstrap current is its effect on neoclassical tearing modes and vacuum magnetic islands. For a transform that increases with minor radius, the bootstrap current can decrease the size of magnetic islands and help overcome problems with flux surface fragility at low aspect ratio. This line of reasoning is counter to the optimization principles on which W7-X is based. However, experimental results in tokamaks with negative central shear (which have good ballooning and neoclassical tearing stability) are a good basis for justifying plasma current as an important element of a U.S. program that does not overlap with non-US stellarator programs. Whether stellarator plasmas with large bootstrap currents do not disrupt will be an important issue for the U.S. stellarator program to investigate.

Magnetic symmetry and quasi-omnigeneity

Improving neoclassical transport, especially at low aspect ratio, is a key element of the U.S. stellarator program. Just a few years ago, it could have been argued that it does not matter how good the neoclassical transport is in a stellarator, as long as it is below the anomalous level. Recent tokamak results with core transport barriers, in which the ion thermal conductivity can drop even below the neoclassical level, highlights the importance in stellarators of having as low a neoclassical transport as possible. The challenge to achieve good confinement at low aspect ratio is exemplified by the poor confinement of high energy ions during ion cyclotron range of frequency (ICRF) heating in CHS (which has the smallest aspect ratio, along with H-1, of any operating stellarator). Significant losses of beam ions were also observed in NBI-heated plasmas in CHS [16].

The U.S. response to that challenge is to open up the paths to tokamak-like neoclassical confinement, while still retaining the intrinsic disruption-free, steady-state attributes of the stellarator. The quasi-axisymmetric approach intrinsically has neoclassical transport comparable to a tokamak. The quasi-helically symmetric approach can have transport even better than a tokamak because the high effective transform can lead to very small poloidal gyroradii. The quasi-omnigeneous approach can potentially lead to transport as good as the quasi-toroidally symmetric approach, without the additional constraint of magnetic field symmetry.

Of all the stellarators in the non-US program, only W7-X has the potential for having low neoclassical transport. However, the principle under which W7-X was optimized is closer to the concept of quasi-omnigeneity than that to quasi-symmetry because of the additional design constraint of minimal bootstrap current. None of the non-US stellarators have magnetic field configurations that are quasi-symmetric, that is to say, there is no direction or angle on a flux surface in which the magnetic field does not vary.

There is an important consequence of this lack of symmetry that has been observed experimentally: stellarator plasmas by and large do not rotate as freely as tokamaks. Although there has been evidence of a poloidal spin-up during H-mode transitions in W7-AS, there has been a uniform observation so far in stellarators that there is very little flow in the toroidal direction. This can be understood from the fact that the parallel viscous damping in the toroidal direction scales as a function of $N/\iota\!\!\!-$, whereas the damping in the poloidal direction scales as m. Typically, in a stellarator $N/\iota\!\!\!- >> m$. Experiments with tangential neutral beam injection in CHS demonstrated that the outer portion of the plasma showed very little toroidal rotation due to the large toroidal viscosity [17]. Comparisons of the damping with neoclassical theory showed good agreement. Only towards the plasma core where the ripple is fairly small was an anomalous perpendicular viscosity required to explain the data.

The importance of magnetic symmetry and the ability of the plasma to rotate is related to the $E \times B$ velocity shear that is necessary to quench turbulence and to the formation of transport barriers and enhanced confinement regimes. The radial force balance equation is given by:

$$E_r = (Z_i e n_i)^{-1} \quad p_i - v_{\theta i} B_\varphi + v_{\varphi i} B_\theta,$$

where $v_{\theta i}$ and $v_{\varphi i}$ are the poloidal and toroidal ion flows, respectively. The required $E \times B$ velocity shear can arise due to the pressure gradient and the two directions of flow on a flux surface. As discussed by Burrell [18], the toroidal flow, for example, can play a major role in both the VH mode as well as the core transport barrier in tokamaks.

Some of the enhanced confinement regimes in stellarators, whether they are the high ion temperature mode in Heliotron E [19] or the bifurcation observed in H-1 [20], show that the $E \times B$ shear can be only due to the pressure gradient, not necessarily the shear in the plasma flow. Having another degree of freedom in which to allow the $E \times B$ shear to quench turbulence could be an important avenue to explore for the quasi-axisymmetric stellarators. Recent results on W7-AS have demonstrated that significant $E \times B$ shear can arise from the constraint that the non-intrinsically ambipolar electron and ion fluxes must be equal. As a consequence, the particle fluxes and thermal conductivities were found to be neoclassical out to r/a_p = 0.7-0.8 [21]. Under these conditions, confinement in W7-AS is a factor of 2.5 times greater than the ISS95 scaling value [22]. These results might be applicable to the search for enhanced confinement regimes in quasi-omnigeneous configurations in which ambipolar radial electric fields develop spontaneously. The associated $E \times B$ orbit rotation can substantially improve confinement beyond that obtained from the magnetic configuration by itself.

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III. KEY PHYSICS ISSUES FOR A U.S. STELLARATOR PROGRAM

One can identify certain physics issues which must be resolved in order for a compact stellarator concept to succeed. These will be the focus of the proposed stellarator research program.

1) Development of confinement-optimized configurations (using the quasi-symmetry or quasi-omnigeneity approaches) and experimental verification of these confinement optimization principles

When the neoclassical theory of transport was developed in the late 1960's, the existing stellarator designs were found to have unacceptably large losses. Although the large losses were originally interpreted as a fatal flaw of stellarators, two types of stellarator designs have been developed that avoid large neoclassical losses: quasi-symmetric stellarators and non-symmetric omnigeneous stellarators. Recently, both of these approaches have been developed at low aspect ratio.

Quasi-symmetric stellarators [1] conserve a component of the canonical momentum (as do tokamaks) and have neoclassical transport properties that are tokamak-like. Although the geometry of a quasi-symmetric stellarator is fully three dimensional, the field strength has a continuous symmetry, either toroidal or helical. Exact quasi-symmetry is not possible in a finite-aspect-ratio stellarator, but quasi-symmetry can be approximated with sufficient accuracy to insure excellent neoclassical confinement. The HSX stellarator [2], which is under construction at the University of Wisconsin, will be the first experimental test of quasi-symmetry and is of the quasi-helical type. The

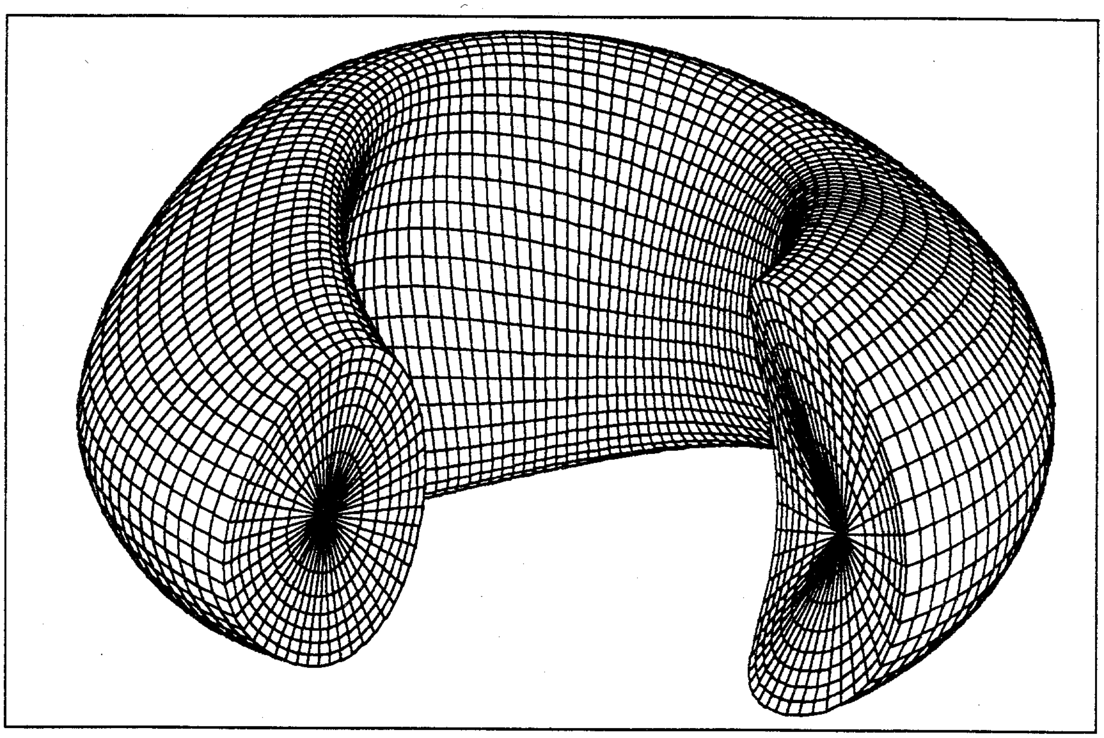

Fig. 5. The LCFS for a quasi-axisymmetric stellarator configuration in which 60% of the rotational transform is produced by the bootstrap current.

quasi-helically symmetric (QHS) approach can have transport even better than a tokamak because the high effective transform can lead to very small poloidal gyroradii. Quasi-toroidally symmetric stellarators can have aspect ratios and bootstrap currents typical of tokamaks, so they resemble tokamak-stellarator hybrids. Like tokamaks, they can have a deep magnetic well and high beta limits for ballooning, even at a low aspect ratio. Figure 5 shows the plasma surface (LCFS) for a quasi-axisymmetric stellarator (QAS) configuration in which 60% of the rotational transform is provided by the bootstrap current. Although the LCFS appears non-axisymmetric in real space, the Fourier spectrum of $|B|$ in magnetic coordinates (Boozer coordinates [3]), upon which the particle drift orbits and neoclassical transport depend, has a dominant axisymmetric component with non-axisymmetric components of only a few percent at the plasma edge. QAS configurations with a higher bootstrap current contribution to the total rotational transform appear more toroidally symmetric in real space. QAS configurations differ from QHS configurations in that QAS configurations have

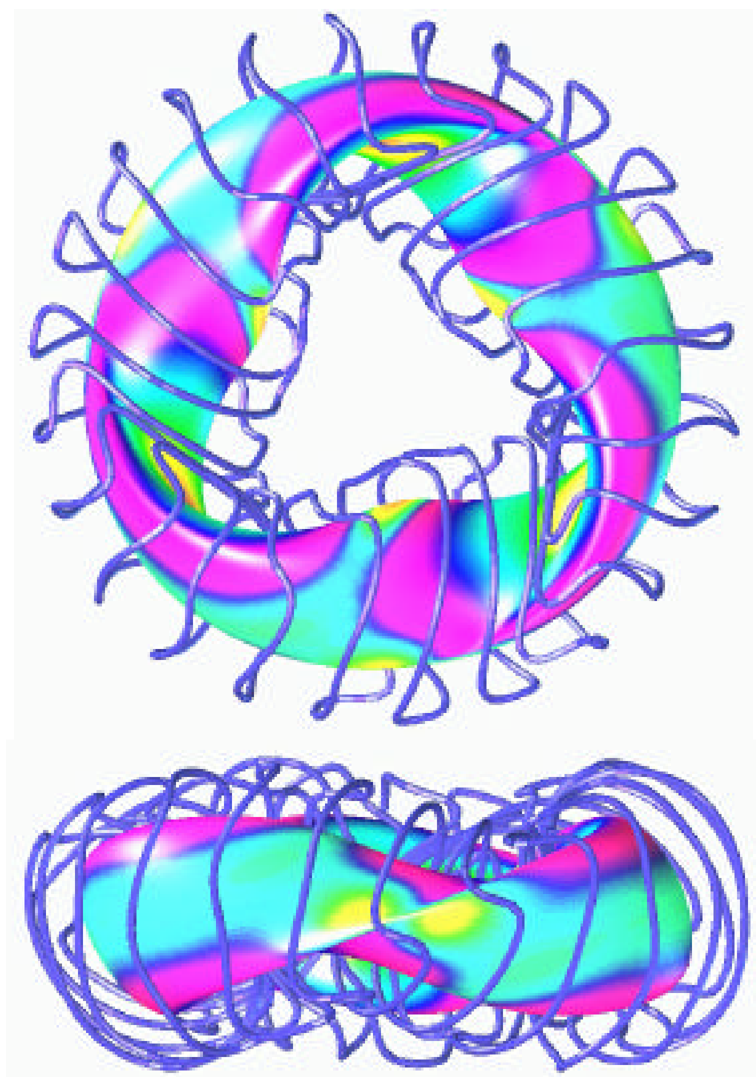

Fig. 6. The LCFS and modular coils for a quasi-omnigeneous stellarator configuration in which 10% of the rotational transform is produced by the bootstrap current.

smaller plasma aspect ratio (typically ~3), larger bootstrap current, and the rotational transform produced by the bootstrap current is in the same direction as that produced by the external coils in the QAS configuration at low collisionality, but in opposing directions in the QHS configuration.

Non-symmetric omnigeneous stellarators [4], or quasi-omnigeneous configurations (QOC), achieve acceptable neoclassical losses by aligning the surfaces on which the second adiabatic invariant J* is constant with the magnetic surfaces. The variation of the field strength within a magnetic surface can be more complicated than in quasi-symmetric configurations since no particular symmetry is imposed. However, aligning the J* surfaces with flux surfaces imposes different constraints, for example, the minima of the field strength within a magnetic surface need to depend weakly on the poloidal angle. Examples of stellarator designs of this type are W7-X [5] (shown in Fig. 3) with a large aspect ratio, and SMARTH [6] with a small aspect ratio. Figure 6 shows the LCFS for a quasi-omnigeneous stellarator configuration in which 20% of the rotational transform is provided by the bootstrap current. The large non-axisymmetric terms in the |B| spectrum and the lower fraction of bootstrap current distinguish QOC stellarators from QAS devices

2) Development of anomalous transport reduction techniques

If the neoclassical losses are sufficiently reduced, the actual losses in an experiment are likely to be anomalous. Global energy confinement in present-day stellarators follows ISS95 scaling [7], which provides a rough estimate of the anomalous losses to be expected in devices under design. A key physics issue for future experiments will be to understand transport barriers and to develop techniques to reduce the anomalous losses, as has been done in tokamaks. It will be essential, for example, to determine whether the external control over the radial electric field afforded by the flow shear in the direction of symmetry in quasi-symmetric stellarators and by flow shear or sheared ambipolar fields in quasi-omnigeneous stellarators can serve the same beneficial role in turbulence suppression that it does in tokamaks.

3) Understanding and improvement of equilibrium and stability β limits

The equilibrium beta limit has in the past been estimated as that value of beta at which the magnetic axis shift is equal to one half of the minor radius across the midplane. Introducing net plasma current, both quasi-omnigeneous [8] or quasi-symmetric configurations [9] are found to have acceptable equilibria at β in excess of 10%. However the conventional axis shift estimate of the equilibrium limit does not address the fundamental mechanism of loss of flux surfaces with increasing beta – formation of islands. In recent years, equilibrium codes that do not assume the existence of flux surfaces have been developed [10], and these codes can be used to calculate the equilibrium beta limit. A complication is the fact that perturbed bootstrap currents are likely to have a strong influence on magnetic island formation, and this physics has not yet been incorporated in the equilibrium codes. Perturbed bootstrap currents are predicted to either strongly magnify or suppress magnetic islands, depending on the relative sign of the shear and the bootstrap current [11].

The stability beta limit is a particular issue for compact stellarators because stellarators typically suffer deterioration in stability as the aspect ratio is decreased. Recent designs for both quasi-omnigeneous and quasi-axisymmetric compact stellarators have predicted ballooning beta limits that are high relative to that presently achieved in stellarators [12,13]. Stability to pressure-driven external kink modes can be studied using ideal MHD stability codes developed in Europe in recent years [14], but those studies are just beginning. Tokamak experiments in recent years have highlighted the particular importance of neoclassical tearing modes in setting beta limits. Stability to these modes depends on the relative sign of the shear and the bootstrap current, which can be designed and controlled in stellarators. While stellarators have recently operated at their predicted beta limits, they have not seriously challenged their beta limits, thus far. A medium-size, short-pulse, strongly-heated U.S. experiment could make a valuable contribution to the world stellarator program in experimentally investigating the MHD beta limit.

4) Understanding of the disruption and stability properties of low-aspect-ratio stellarator configurations having substantial plasma current (tokamak/stellarator hybrids)

Significant parallel current can destabilize MHD modes, including neoclassical and resistive tearing and kink modes, and thus may reintroduce the disruptions observed in tokamaks. While the linear stability of these modes can be calculated in stellarators, the non-linear saturation and consequences have not been quantitatively characterized and experimentally verified. Experiments on W7-A showed immunity from disruptions with modest amounts of externally generated rotational transform (~0.14) in addition to that from the Ohmic current [15]. Similar results were observed on CLEO [16]. Experiments including significant parallel current must investigate the role of these instabilities and the empirical range of disruption-free operation at low aspect ratio and high β.

5) Understanding of the compatibility of the bootstrap current with the required plasma current and current profiles

To avoid the economic disadvantages associated with large current-drive systems, all or almost all of the required current must be provided by the bootstrap effect. In stellarators, the helical curvature of the magnetic field produces a bootstrap current with opposite sign to that arising from the toroidal curvature of the field. Thus, depending on the magnetic design of the stellarator, the bootstrap current can be of either sign or can be carefully balanced to be approximately zero, as was demonstrated in the Advanced Toroidal Facility (ATF) [17] and incorporated in the W7-X design [5]. The bootstrap current varies inversely with the rotational transform and is driven by the pressure gradient. Since confinement is provided by the rotational transform and also determines the current-driven part of the transform, equilibria with high bootstrap currents are very nonlinear. For configurations with significant bootstrap current, self-consistent equilibria must be developed and the capability to control nonlinear equilibria at high beta must be investigated.

6) Divertor – power and particle handling

In addition to the fundamental role of power and particle handling systems for exhaust, tokamak experience has shown that these systems are critical in creation and control of enhanced confinement regimes. There is much less experience with divertors in stellarators as compared with tokamaks, although there are concepts being tested in the world stellarator program such as the Local Island Divertor [18] being developed for LHD. It is not known to what extent divertor concepts for truly axisymmetric devices such as tokamaks, or for higher-aspect-ratio stellarators, can be applied to the compact stellarator devices. Divertors will be a particular challenge if the magnetic configuration changes significantly between the vacuum or startup plasma and the full-current, full-beta operation.

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IV. THE PROPOSED NATIONAL STELLARATOR RESEARCH PROGRAM

The Ten-Year Goal. The U.S. fusion program should embark on a 10-year proof-of-principle program that develops the knowledge base needed for compact, high-beta, good-confinement stellarators. The program should consist of well-coordinated research in several areas: (1) proof-of-principle (PoP) experiments in a flexible, reconfigurable facility; (2) the present HSX experiment; (3) new concept exploration experiments; (4) experimental collaboration with the international stellarator program in specific areas; (5) theory focusing on concept optimization and key stellarator physics issues; and (6) reactor systems studies to guide concept optimization tradeoffs.

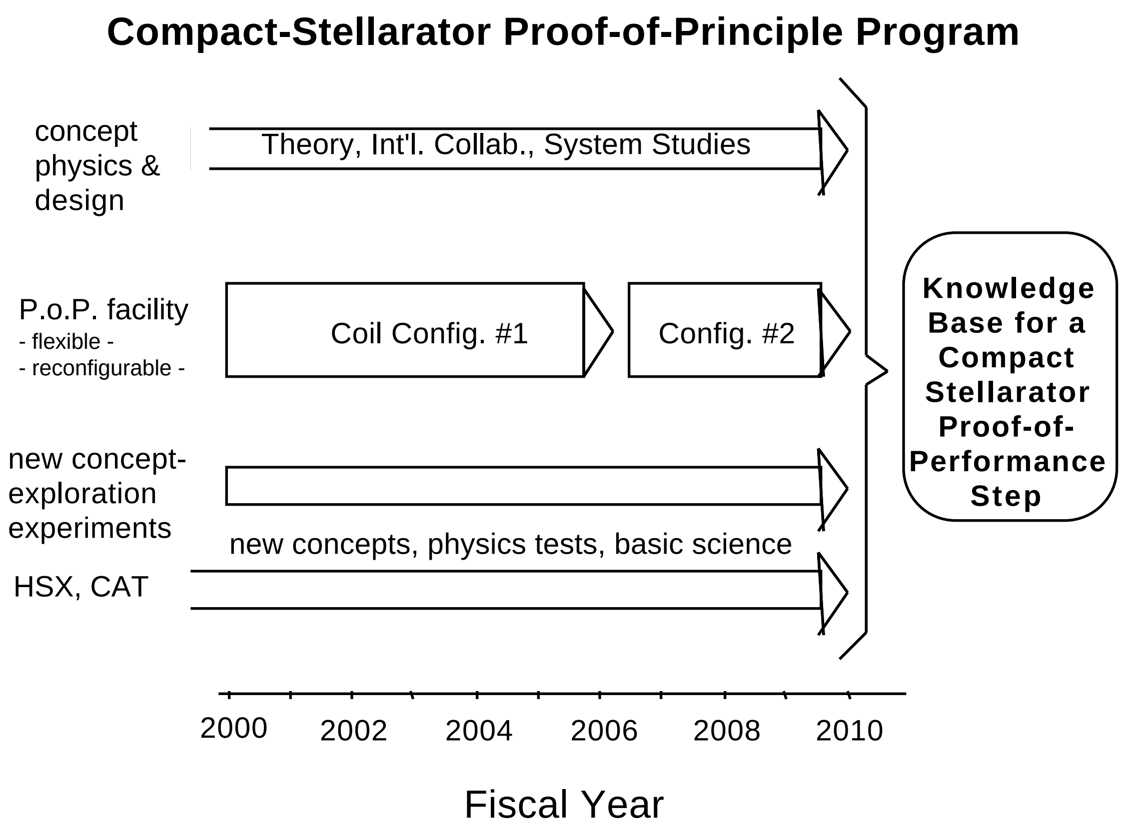

Fig. 7. Road map for the proposed stellarator PoP program. The PoP experiment would be reconfigured after a few years of operation to test improved configurations emerging from the research program that integrates the different concept improvement elements.

The six PoP program elements, which cross-link with each other to provide a coherent, well-integrated program, are depicted schematically in Fig. 7. The PoP facility would be the focus of the PoP program. It would be reconfigurable to ensure that experimental tests of the new developments arising from the total stellarator PoP program can be conducted expeditiously. HSX and the new concept exploration experiments will allow tests of new optimized configurations, basic concept optimization physics studies, and extension of 3-D plasma science. The results from these experiments and from non-US collaboration, theory, and system studies will help determine the configuration to be developed in later phases of the PoP facility experiments.

This program will build on the substantial data base in stellarators and tokamaks and will make important contributions to the world fusion program. If this plan is carried out, the resulting knowledge base will be sufficient to permit comparisons with steady-state tokamak-based power plant designs and will provide a basis for proceeding to the following step, a proof-of-performance program to study more reactor-relevant plasmas in a compact stellarator configuration.

IV.A. Proof-of-Principle Experiments

The new compact stellarator configurations discussed in Section III offer neoclassical particle confinement similar to tokamaks, stability beta limits similar to the advanced ARIES tokamak reactor designs, possible freedom from disruptions and the need for external current drive, and control of turbulent transport through flow shear and the magnetic configuration. These are shared goals across the fusion program for achieving an attractive magnetic confinement scheme, but have not been achieved. The principles on which the new stellarator configurations are based must be tested and validated experimentally at the proof-of-principle scale to demonstrate that their promising characteristics can be simultaneously attained and can form the basis for further development of fusion reactors.

The mission of these experiments will be to test, develop, and demonstrate understanding of the key issues for compact configurations, described in Section III. Scientific questions to be resolved include:

- *Can a high-beta configuration, including bootstrap currents and external transform, avoid disruptions?*
- *What are the beta limits and limiting mechanisms?*
- *Can neoclassical transport be reduced by proper configuration design?*
- *Can turbulent transport be controlled, leading to enhanced global confinement? (for example, by flow shear or magnetic configuration)*
- *Can transport and stability be controlled through external magnetic configuration control?*
- *Are neoclassical islands and tearing modes stabilized with bootstrap current and the proper choice of shear?*

In addition, empirical optimization of the plasma performance will be studied, contributing to the further development of the compact stellarator concept. This has been remarkably effective in the development of tokamaks.

The plasmas used in these experiments should be in a physical regime similar to that expected for a reactor, to ensure that the physics understanding developed can be extrapolated. Thus, the PoP plasma must have low collisionality, good shielding from edge-neutral particles, and equilibrated inductive currents. Such a facility: should be similar in size to the tokamak experiments at the same stage of development (e.g. ASDEX, D-III, PDX/PBX); will require multi-megawatt auxiliary heating, moderate pulse lengths, and detailed diagnostics; and should build on the substantial developments of the tokamak program. Flexibility in the magnetic configuration is essential, so it can be varied to study control of plasma transport and stability.

The PoP facility must also be reconfigurable to allow modification of the plasma configuration. Results from the other elements of the proof-of-principle program – theory, system studies, small experiments, and non-US experiments – as well as from the PoP experiment itself, will lead to a better understanding of, and improvements in, compact-stellarator plasma configurations. A reconfigurable PoP facility will permit experimental tests of new configurations to be conducted in a timely fashion as they emerge from the compact stellarator PoP program. This is illustrated schematically in Fig. 7.

At present, both quasi-axisymmetric and quasi-omnigeneous configurations look attractive for compact stellarator reactors, but each has distinct complementary advantages. Both must be developed experimentally to maintain the broadest possible scientific base for the program's ultimate success. A determination of which is the better strategy will be one of the program's goals. The new proof-of-principle facility will provide sufficient plasma performance and machine capability for integrated testing of a compact stellarator configuration with high beta and bootstrap currents that can form the basis for extrapolation to more reactor-relevant performance. In order to minimize cost, it is desirable to take advantage of an existing facility, the PBX-M tokamak, and modify it for the proof-of-principle tests. The quasi-axisymmetric concept is more compatible with the PBX-M constraints, so this configuration will be chosen for the initial tests. The PoP facility will be modified to test improved configurations as they are developed by the program. A new concept exploration facility will be constructed to test the basic optimization principles of quasi-omnigeneity.

These experiments would be a major element in the U.S. fusion program and would be expected to lay the scientific and operational foundation for a later proof-of-performance experiment to test extrapolatability of the compact stellarator concept. Concept refinement and definition studies are presently underway to establish configuration and machine requirements for the PoP facility. It is anticipated that it could operate as early as 2002, depending on the availability of funding. It would be similar in scale and approach to the National Spherical Torus Experiment (NSTX), a proof-of-principle experiment now under construction. The cost of the stellarator PoP experiment, the National Compact Stellarator Experiment (NCSX), would be about $35 million. Operating costs would be about $20 million per year.

IV.B. The Helically Symmetric Experiment (HSX)

The Helically Symmetric Experiment (HSX) [1,2] is the principal element in the U.S. stellarator program at the present time. Construction is nearly complete with operations to commence in FY 1998. The goal of the HSX experimental program is to test the improved confinement properties in quasi-symmetric configurations and to exploit its unique geometry to elucidate outstanding issues in toroidal confinement. The symmetry in the |B| assures neoclassical transport analogous to the tokamak and reduced by nearly two orders of magnitude from the conventional stellarator in the collisionless regime. The primary objectives of the physics program are:

- Verify reduction of neoclassical transport for quasi-symmetric configurations; quantify levels of symmetry necessary to achieve full benefits
- Demonstrate a reduction in the direct loss of deeply trapped particles
- Show that restoration of a direction of symmetry leads to lower viscous damping of plasma rotation on a flux surface

HSX is a quasi-helically symmetric [3] (QHS) device, and the only device of this type in the world program. The physical parameters of HSX are shown in Table 4.1. The symmetry is obtained by reducing the toroidal curvature term in the magnetic field spectrum through appropriate shaping of the plasma. HSX has the toroidal curvature of an aspect ratio 400 conventional device, while being a fully toroidal system of aspect ratio 8. The spectrum thus possesses a single dominant helical harmonic, with symmetry breaking terms well under 1%. QHS configurations have also been identified [4] numerically at aspect ratio 6 with only minimal increases in symmetry-breaking terms from those in HSX.

QHS configurations have an effective transform given by the number of field periods minus the actual transform, $|N - \iota|$; for HSX with four field periods and near unity transform, $\iota_{\mathrm{eff}} \sim 3$. Thus, HSX will have neoclassical transport analogous to a $q = 1/3$ tokamak. The high effective transform has multiple benefits, which factor into the elements of the experimental program:

- Reduction of Pfirsch-Schlüter and bootstrap currents; small finite-beta effects on the magnetic field spectra and equilibrium
- Smaller banana widths with accompanying improved confinement of high-energy particles; HSX can fit as many banana widths within its 15 cm minor radius as a stellarator or tokamak with a much larger plasma cross-section.
- Anomalous transport should be reduced, based on data from L-2 and ISS95 scaling, which scale inversely with transform.

HSX is an extremely flexible device. A sketch of HSX is given in Fig. 8. The QHS field is produced by a set of 48 modular coils. A set of 48 planar, non-circular, auxiliary coils provides for variation in rotational transform, magnetic well depth, and spectral content. One configuration of the auxiliary coils (mirror-mode) breaks the quasi-symmetry and increases the transport back to the level of a conventional stellarator, with minimal effect on the plasma stability. In an alternate mode (well-mode), the plasma stability limit to Mercier and ballooning modes [5] can be varied by a factor of 3 (Mercier: 0.4% to 1.3% , ballooning: 0.7% to 1.7%), with only small changes in the neoclassical transport. In the mirror mode, direct losses are dramatically increased and the neoclassical electron thermal conductivity increases by 2-3 orders of magnitude. Finally, the parallel viscosity can be altered by 1-2 orders of magnitude to examine how changes in the plasma rotation and radial electric field affect confinement.

H-mode confinement characteristics are not necessarily predicted in the HSX physics program. The experimental program will, however, provide insight as to the mechanisms by which the $E \times B$ shear necessary for the quenching of anomalous transport can be maintained. Through use of the auxiliary coils, HSX can span the space between having a large plasma flow contribute to the radial electric field (in QHS mode) or having the non-intrinsically ambipolar electric field arise naturally when the symmetry is broken. Additionally, HSX will investigate the relationship between anomalous transport, effective transform, and the level of symmetry. Initial flow measurements will be made using passive spectroscopy utilizing the C^{+4} and other impurity lines. We are implementing a CHERS system over the next three years for more detailed flow measurements. Density profiles will be measured in HSX using a 9-chord interferometer being set up in collaboration with David Brower of UCLA. We are also working with Brower and Tony Peebles to set up a reflectometer for density fluctuations as we move more into the study of anomalous transport. In addition to our 8-point Thomson scattering system, we have an ongoing collaboration with Neville

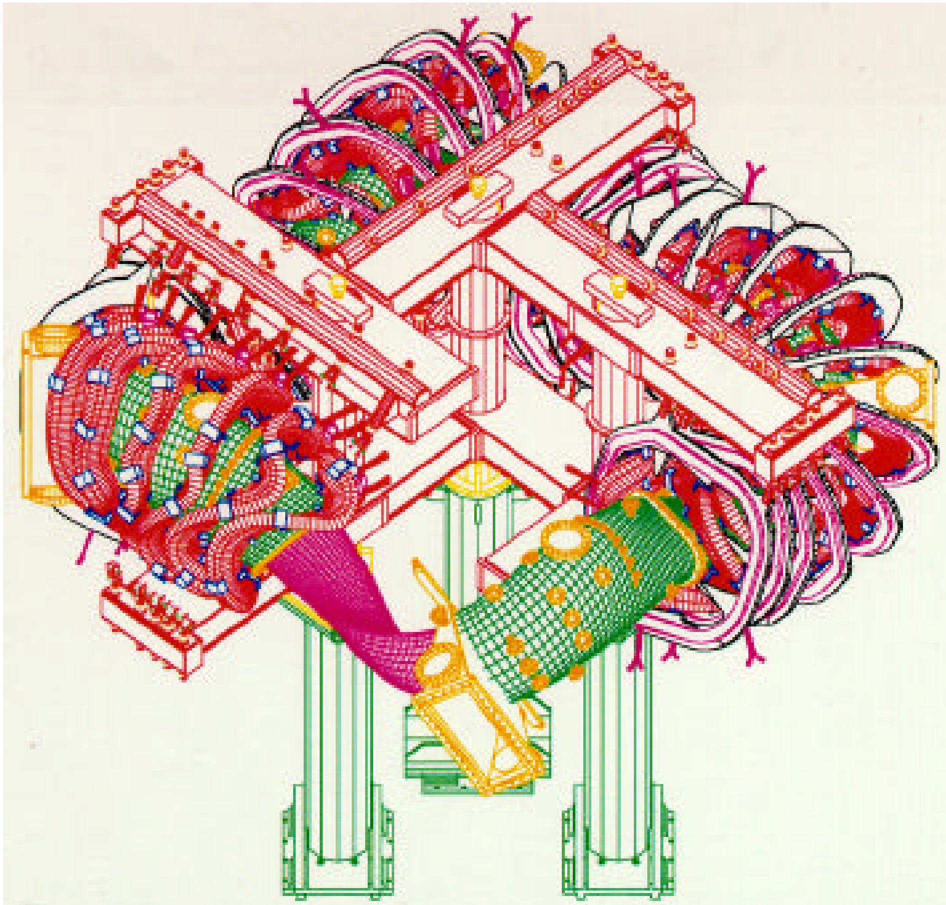

Fig. 8. The LCFS, main nonplanar modular coils, trim toroidal field coils, and the support structure for the HSX experiment.

Luhmann's group at UC-Davis to implement a 2-D ECE imaging system on HSX for electron temperature profiles; this diagnostic will also provide electron temperature fluctuations for the anomalous transport studies.

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HSX will use a 28-GHz gyrotron with a power output of 200 kW to heat the plasma electrons into the collisionless regime for this first part of the HSX program. This has been shown to be effective in other stellarators of similar size and field strength, such as L-2, and is ideal for carrying out the above investigations. The 28-GHz heating does restrict HSX to operation at 1 T or 0.5 T and to densities less than 10^{13} cm^{-3}. Heating in the ion cyclotron range of frequencies in stellarators has typically suffered from impurity problems. Poor confinement of the high-energy ions in the rippled magnetic field is suspected as the cause of the unsuccessful results. Experiments performed on CHS with ICRF heating [6] of ECH target plasmas has shown strong electron heating. The loss of high-energy trapped ions produced in an ion heating regime degraded the performance and limited the duration of the discharge. The high effective transform and good trapped-particle confinement in HSX should permit effective ion or electron heating in the QHS mode with ICRF, although we would need to operate at somewhat higher densities than CHS for ion heating to reduce central charge-exchange energy loss. One megawatt of ICRF on HSX would provide a wider range in the density-field operating space as well as provide direct data on effects of orbit confinement and impurity production with ICRF. Additionally, the ICRF in combination with the ECRH and auxiliary coils could allow tailoring of the radial electric field through differential loss mechanisms over the plasma cross section. The addition of the ICRF is envisioned as an extension of the main program after the primary goals above are accomplished.

In order to obtain maximum scientific benefit from the HSX experiment, tight coupling with the theory and computation effort (Chapter 5) is needed in the following areas: effects of small non-symmetric fields (how much symmetry is enough?), ballooning mode stability limits (can they be made low enough to test?), Fokker-Planck and delta-f modeling of ECH in HSX, and turbulence and anomalous transport predictions for HSX-type plasmas.

The initial resources needed for this effort are $1.6 million per year (the current level); further resources will be required as the program evolves.

The HSX Device		Estimated Parameters	
Major radius:	1.2 m	Heating power (28 GHz ECRH)	200 kW
Average plasma minor radius:	0.15 m	Pulse length	100 ms
Field Periods	4	**Electron density**	$< 10^{13}$
Rotational transform: axis	1.05	Central electron temperature (with 100 kW absorbed)	~ 1 keV
Edge	1.12	Energy confinement time (LHD)	2 ms
Magnetic well depth	0.6%	Plasma electron β	0.3%
Magnetic field strength	1.37 T	ν^*_e	<0.1
Magnet flat-top (full field)	0.2 s		

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IV.C. New Concept Exploration Experiments

In addition to HSX, exploring quasi-symmetry, other exploratory small-scale experimental investigations are vital to the successful implementation of the proposed U.S. stellarator program to develop compact 3-D configurations with good confinement properties at high beta. The range of innovative concepts described in Sec. III necessitates experiments to test key issues of the physics and optimization to help guide the PoP program and later phases of the PoP facility. In particular, there is a vital need for an experiment exploring the nonsymmetric quasi-omnigeneous optimization of neoclassical transport, for potential use in a second phase of the PoP facility. Small-scale experiments, including modifications of existing experiments such as the Compact Auburn Torsatron (CAT), can also play an important role in testing technologies (for example, RF coupling and diagnostics) as well as in confinement physics studies. Exploratory experiments should be optimized for a particular goal, for example, confinement or stability, but usually would not explore as wide a range of simultaneous issues as in the PoP experiment.

There are a number of fundamental issues on stellarator transport, heating, and optimization that can and need to be addressed in smaller scale exploratory experiments.

(1) *Demonstration of the reduction of neoclassical transport via nonsymmetric quasi-omnigeneity, the role of anomalous transport, and the level of rotation damping.*

(2) *Exploration and development of efficient methods to produce sheared E × B flow, including methods to switch between the ion and electron transport roots (where the radial electric field confines the ions or electrons, respectively).*

(3) *Determination of whether high edge shear is required to avoid disruptions (from kinks) with significant current and the dependence on the current profile and edge magnetic transform.*

(4) *Tests of the predicted dependence of the size of equilibrium islands on shear, the bootstrap current direction, and beta.*

(5) *Determination of the specific techniques of RF heating and flow control that will be most successful in quasi-axisymmetric and non-symmetric quasi-omnigeneous configurations.*

(6) *Development and tests of methods for particle and energy-exhaust control that are compatible with optimized stellarator configurations.*

Exploratory concept stellarators are extremely useful in testing the new physics principles emerging from recent stellarator theory as well as making comparisons with issues previously confronted in tokamak research. Moreover, because of the largely unexplored configurations envisioned for the U.S. stellarator program, small-scale experiments are essential guideposts in selecting the optimized physics to be incorporated in later stages of the PoP facility. With targeted investigations such as those described, smaller experiments can make valuable contributions to the development of an attractive compact stellarator concept.

The resources needed for this effort are $2.5 million per year.

IV.D. Collaboration with the International Experimental Program

The international stellarator program features billion-dollar-class facilities under construction in Japan (LHD, March 1998) and Germany (W7-X, 2005) supplemented by medium-size ($30-100 million scale) experiments in Japan (CHS), Germany (W7-AS), Spain (TJ-II), etc. Given the very substantial investment by Europe and Japan in stellarators, the U.S. program can benefit greatly from access to the non-US experiments. The draft DOE OFES *Strategic Plan for International Collaborations in Fusion Science and Technology Research* recommends that the U.S. "pursue opportunities for collaboration on stellarators through participation in the Large Helical Device program in Japan and the Wendelstein program in Germany" as an effective way to achieve U.S. fusion program goals using unique worldwide fusion facilities.

The wide range of stellarator configurations accessible on LHD, W7-AS, CHS, and TJ-II allow study of the role of aspect ratio, helical axis excursion, magnetic-island-based divertors, and the consequences of a net plasma current, elements that are being incorporated in the low-aspect-ratio stellarator concepts under study in the U.S. program. Areas of particular importance are ion heating and transport, neoclassical transport, role of electric fields in confinement improvement, enhanced confinement modes, beta limits, practical particle and power handling, profile and configuration optimizations, and steady-state performance. An additional benefit is broadening our understanding of toroidal confinement (e.g., steady-state transport barriers) through comparisons with the related tokamak concept.

LHD. LHD allows study of stellarator physics at more reactor-relevant parameters ($\beta \geq 5\%$, T_i ~ 10 keV, τ_E hundreds of ms, etc.) The order of magnitude increases in plasma volume, heating power, and pulse length of LHD over that in existing stellarator facilities will allow size scaling studies for a confinement concept that is second only to the tokamak in development. The superconducting coil system, divertor, and steady-state multi-MW heating power allow comparison with steady-state component development in tokamaks (particularly Tore Supra). The U.S. can participate in the LHD program through measurement and analysis of the energetic ion distribution for study of ion heating and transport, plasma-material interactions in the divertor chamber, and specialized diagnostics for study of high-β, high-T_i, and high-τ_E stellarator plasmas.

W7-AS. Confinement improvement, configuration optimization, and divertors are being studied in W7-AS in magnetic configurations complementary to that of LHD. The U.S. can capitalize on the U.S. pellet injector on W7-AS, analysis of the consequences of a net plasma current in W7-AS (a key element of the low-aspect-ratio stellarator concepts under study in the U.S. program), and analysis of a magnetic-island-based divertor system applicable to modular-coil stellarators.

CHS. CHS allows study of stellarator optimization physics, especially transport and beta limits, at plasma aspect ratios as low as 4.5 in a well-diagnosed experiment. The U.S. has contributed strongly to the Heavy Ion Beam Probe measurements of the electric field, ICRF heating, and the local island divertor studies on CHS, areas that should be developed further on LHD.

TJ-II. The U.S. can also take advantage of the U.S.-supplied neutral beam heating system on the TJ-II flexible heliac to understand beta limits and transport in a stellarator with a large helical axis excursion, another important ingredient in U.S. stellarator configuration optimization.

H-1. The Australian H-1 flexible heliac allows studies of a stellarator with a large helical axis excursion in a configuration complementary to that of the Spanish TJ-II.

The resources needed for this effort are $1.5 million per year.

IV.E. Theory Focusing on Concept Optimization and Key Stellarator Issues

The changing world perspective on stellarators is associated with three areas of advance in the theory of intrinsically three-dimensional systems: (1) new concepts that simplify the description of the physics; (2) new computational tools that allow the calculation of equilibrium, stability, and neoclassical transport properties; and (3) new magnetic configurations that address the deficiencies of earlier stellarator designs. The U.S. theory program has played a major role in all three areas, but the U.S. stellarator theory effort must be significantly expanded in size and scope to attain the goals of the proposed U.S. proof-of-principle stellarator program.

Stellarator theory should address six areas: (1) MHD equilibrium, (2) MHD stability, (3) neoclassical transport and drift orbits, (4) microstability and anomalous transport, (5) divertor and edge physics, and (6) optimization of magnetic configurations. To build and maintain a healthy research effort, each area requires the equivalent of 2-3 theorists, approximately 16 in total, plus graduate students and postdoctoral fellows.

Equilibrium limits in a stellarator are largely determined by the breakup of magnetic surfaces, an issue that does not arise in axisymmetric equilibria. Nevertheless, physical effects well known in tokamaks such as neoclassical tearing modes and rotational healing of islands can change, and in many cases improve, the quality of stellarator magnetic surfaces. More efficient algorithms, particularly for free-boundary equilibria, would allow a more rapid advance of stellarator research. Many of the physics and numerical issues that arise in equilibrium theory also appear in stability theory. An additional area of stability theory that requires attention is whether the standard ballooning mode analysis gives pessimistic answers for the stability of the plasma to localized perturbations. There have long been analytic reasons for believing this might be true and recently there have been numerical indications.

The constraint of adequate confinement of the particle drift orbits is not trivially satisfied in a stellarator. This is in contrast to the situation in an axisymmetric tokamak. Two fundamentally different concepts for achieving good orbits in stellarators are known: quasi-symmetry and nonsymmetric omnigeneity (quasi-omnigeneity). Further research is required to insure that stellarators can be optimized for higher beta while maintaining good orbits. A number of neoclassical transport issues remain to be addressed. An example is the development of transport theory for nonsymmetric omnigeneous systems. Such systems differ from both quasi-symmetric and classical stellarators in that trajectories are fully three-dimensional, yet the trajectories remain close to a flux surface, and there are no direct losses.

The theory and computer codes that have been developed for studying the microstability and anomalous transport in tokamaks should be adapted to stellarators. Such codes are needed to address whether the far broader range of magnetic configurations that arise in stellarators allows a significant reduction in the predicted transport rates.

Concepts for stellarator divertors exist and some computational studies have been made. However, divertor theory that is appropriate for stellarator applications requires significant development. Codes that model tokamak divertors do not address important issues for stellarator divertors.

At the core of an innovative stellarator program must be a significant effort on the development of new, better optimized, stellarator concepts. The effort must have two parts: the development of the optimization criteria and the exploration of configurations that address these criteria. Both the plasma configuration and the coil design are important areas for optimization. In guiding optimization efforts, a number of subsidiary studies are required. An example is the tradeoff between beta and aspect ratio in a stellarator. Theoretically one can achieve volume averaged betas in stellarators above 30% at infinite aspect ratio, but how does this beta limit change as the aspect ratio is made smaller?

A stellarator theory program of ~$3.5 million per year would allow the United States to address the theoretical issues that are critical to innovative development of the stellarator concept and would make the United States the a strong contributor to innovative three-dimensional plasma physics.

IV.F. System Studies to Guide Concept Optimization Tradeoffs

Integrated physics and engineering systems studies allow assessing the reactor potential of innovative stellarators and setting criteria that a concept should meet to be an attractive reactor candidate as well as optimizing a candidate configuration for an experiment. These capabilities have been developed in U.S. stellarator reactor studies [1,2] and the ARIES tokamak reactor studies [3]. The most recent example is the U.S. Stellarator Power Plant Study (SPPS) [4], a "scoping study" at a smaller scale than the typical ARIES study. A byproduct of the SPPS work was development of the Modular Helias-like Heliac (MHH) configuration on which the SPPS was based. This four-field-period coil configuration (MHH4) has physics properties similar to the Helias configuration on which W7-X is based, but with lower aspect ratio, which allowed reducing the reactor size from R_0 = 22-24 m (for the W7-X-based HSR [5]) to R_0 = 14 m for the SPPS reactor. The possibility of further significant reductions in reactor size is a major motivation of the proposed U.S. stellarator program; the goal would be another factor of 2 reduction in major radius. The SPPS configuration extrapolated to a reactor power plant that was economically competitive with the second-stability ARIES-IV tokamak reactor [6] assuming the same unit costs for components with complicated geometry and the same availability for both devices. A more detailed study is needed to assess the potential advantages and design issues for quasi-axisymmetric and quasi-omnigeneous configurations as fusion power plants relative to conventional stellarators and tokamaks. An in-depth study (similar in size to the ARIES studies) would clarify the trade-offs on more issues than were possible in the SPPS, and would greatly assist optimization of these new stellarator configurations.

The areas that need to be explored for compact stellarator configurations include:

- cost/benefit tradeoffs for aspect ratio, beta limit, and confinement enhancement to guide the physics optimization efforts and the targeting of experiments;
- limits on acceptable orbit losses for α-particles and other energetic ions to help bound how much optimization of orbit losses is required;
- consequences of practical particle (including impurities) and power handling to help integrate the divertor geometry with the coil geometry;
- cost/benefit tradeoffs for plasma-coil spacing, access between coils, maximum field on the coils, degree of nonaxisymmetry, etc. to guide optimization of the coil design; and

- integration of reactor systems optimization with stellarator physics and configuration optimization to guide the development of self-consistent attractive reactor configurations.

Both coil design and physics constraints are important in setting the minimum size of a fusion power plant [1]. Adequate space Δ between the edge of the plasma and the center of the coils is needed in a reactor for the plasma scrapeoff/divertor region, the first wall, the thick (≈2 m) blanket and shield assembly, the superconducting windings and assembly gaps, etc. For a given stellarator coil configuration, Δ/R_0 is a constant and relatively small; the coils normally have to be close to the plasma in a stellarator because the higher order multipole components that produce the desired field configuration decay away rapidly from the coils. More compact stellarator reactor designs might be obtained if the physics properties of a more compact stellarator are not compromised too much in the process of increasing Δ/R_0. Such stellarators could have a significant impact on the viability of the stellarator reactor concept.

Other nations (most notably Japan and Germany) are continuing to study their stellarator variant for its reactor potential. In addition to the studies outlined above, the United States could effectively participate in the world effort on power plant studies in selected areas where it has special expertise (systems studies, costing algorithms, blankets and shields, concept innovation, etc.) at a relatively modest level.

The resources needed, averaging $1 million per year and alternating between 2 years of scoping studies at the $0.3 million per year level and 2 years of the more detailed ARIES-type studies with a specific QA or QO coil configuration at $1.7 million per year, should be funded as part of the ARIES program.

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V. THE U.S. SHOULD INVEST MORE RESOURCES IN STELLARATOR RESEARCH

Stellarators have the potential for disruption-free operation at high beta and configuration maintenance with minimal recirculating power. Because of these advantages, stellarator research constitutes a major fraction of the worldwide program to develop an attractive fusion power source. However, there is a critical gap in the world program, namely the need for development of compact stellarator designs. The U.S. can lead the world in this area by increasing its investment in stellarator research and undertaking the proof-of-principle program described herein. To do so would benefit the world program and would further key aims of the U.S. fusion program: confinement concept innovation, fusion science understanding, and plasma physics advancement. An expanded U.S. stellarator program is timely now because of a unique combination of opportunities: the existence of a well developed and expanding knowledge base, a critical unmet need for compact stellarator development, the existence of promising concepts ready to move to experimentation, and the opportunity to advance these aims at modest cost.

A Well-Developed and Expanding Knowledge Base. Stellarator research is well advanced. This has provided a powerful design capability without which the concept improvements we have discussed would not be possible. The range and variety of stellarator configurations affords considerable design freedom in which to optimize. These capabilities will grow with the rapid expansion in stellarator knowledge expected in the next few years. The combination of powerful tools and wide design freedom gives stellarators their great potential for producing even more attractive designs.

A Critical Unmet Need: Compact Stellarator Development. The two large facilities now under construction abroad are designed to demonstrate steady-state operation and a level of performance that allows extrapolation to devices capable of burning plasma operation. The aspect ratios of these and other, smaller non-US stellarators range from 5 to 11 and extrapolate to very large reactors. Stellarator reactors projected from current knowledge tend to be relatively large, with major radii of 14 m to 24 m. Concepts with aspect ratios in the 2-4 range that have been studied theoretically show promise of shrinking the projected size of stellarator reactors. These concepts and the physics strategies on which they are based must be further developed theoretically and tested experimentally. Reactor studies taking advantage of compact designs, and guiding the experimental and theoretical research, must be carried out. These needs will go unmet unless more of the expertise and available technical resources of the U.S. fusion program are applied to stellarator research.

Promising Concepts Are Ready for Experimentation. The new confinement-optimized configurations recently developed in the U.S. hold the promise that a low-aspect-ratio disruption-free stellarator with good confinement and high beta can be developed, and would lead to a more compact stellarator reactor. These configurations, quasi-axisymmetric and quasi-omnigeneous stellarators, are ready for experiments to confirm their predicted transport and stability properties, to determine the conditions required for avoiding large-scale instabilities and discharge termination events (disruptions), and to demonstrate reduced anomalous transport. Integrated experimental testing of a high-beta, good-confinement compact configuration and focused experimental tests of transport optimization strategies are needed.

Opportunity to Lead in Compact Stellarators at Modest Cost. The immediate needs of compact stellarator development require investments far smaller than those of LHD and W7-X, which cost \$0.5-1.0 billion. The largest element, a proof-of-principle experiment for integrated concept testing, will cost in the range of ~\$35 million for construction and ~\$20 million annually for operation and research. Other needed elements (and the required annual budgets) include HSX (\$1.6 million), a focused concept exploration experiment (~\$2.5 million), collaboration with foreign programs (\$1.5 million), theory and concept optimization (\$3.5 million), and system studies (\$1 million). Expenditures at these levels are affordable by the U.S. fusion program, yet would be sufficient for the U.S. to make a critical contribution to world fusion research and be the leader in the development of compact stellarators.

The United States should capitalize on this extraordinary opportunity through a significantly expanded stellarator research program, with a corresponding increase in annual expenditures for stellarators.

APPENDIX. STATUS OF STELLARATOR RESEARCH

Plasma Parameters. Stellarators have attained significant plasma parameters that are second only to those in tokamaks. Below are the highest values achieved for ion temperature, electron temperature, average beta, confinement time, density, and pulse length. At the present time, there does not appear to be any fundamental limit to any of these parameters. Many of these parameters were obtained only recently due to the extension of available heating sources, as well as progress in wall conditioning, the control of recycling and the use of boronization. Most of the data shown below were obtained on W7-AS. The highest average beta value of 2.1% was recorded on CHS. W7-AS itself has achieved an average beta only slightly less than this value, 1.8%. Both experiments have nominal minor radii of 20 cm, substantially smaller than present-day tokamaks. The longest pulse length, 1 hour and 17 minutes, was produced on ATF. What the achievement of these parameters tells us about the physics of stellarators is described further in the rest of this Appendix.

	T_i (keV)	T_e (keV)	$n_e(0)$ (10^{20} m^{-3})	B (T)	P (MW)
Highest T_i, low ν_i* $T_i = 1.6$ keV [1]	1.6	1.8	0.5	2.5	1.25
Highest T_e, low ν_e* $T_e = 5.0$ keV [2]	0.2	5.0	0.2	2.5	1.4
Highest Beta $\beta = 2.1\%$ [3]	- -	- -	≥1	0.57	1.8
Longest Energy Confinement Time $\tau_E = 55$ ms [4]	0.8	1.0	1.1	2.5	0.35
Highest Density: $= 3 \times 10^{20}$ m^{-3} [5]	- -	0.3	≥3.0	2.5	1.5
Longest Discharge $T_{pulse} = 4667$ s [6]	- -	0.03	≥0.02	0.5	0.07

Neoclassical Transport. The combination of toroidal and helical ripple in a stellarator can lead to large neoclassical transport in the low collisionality regimes and to direct orbit losses. Since the early 1980's, it has been relatively easy for stellarator experiments to test confinement of electrons in the low collisionality regimes using ECH to heat the plasma. Only fairly recently has wall conditioning and recycling control progressed to the point where the ions have entered the low collisionality regime as well on Heliotron E [7] and W7-AS [1]. In general, stellarator transport tends to be neoclassical towards the plasma center and anomalous towards the outside. Efforts to suppress the anomalous transport at the plasma edge, using the neoclassical ambipolar electric field, will be discussed in the next section.

The stellarator community has pursued two methods experimentally to overcome the limitations of neoclassical transport in stellarators: (1) tailor the magnetic field to minimize the neoclassical losses, and (2) rely on the ambipolar electric field to decrease the neoclassical losses. Both of these mechanisms have been shown experimentally to be effective. One of the earliest models to improve the neoclassical transport was developed by Mynick [8] who showed analytically and numerically in the early 1980's that localizing the helical ripple to the inside of the torus could lower the transport by an order of magnitude. Experimentally this was verified in Heliotron E [9], as well as in other stellarators, by shifting the magnetic axis inward with a vertical magnetic field. The improvement in the confinement was on the order of 50 % even though the global confinement was dominated by anomalous transport. The 'σ-optimization' approach developed by Mynick (σ refers to the modulation of the helical ripple), corresponds in practice to having roughly constant minima in the magnitude of B along a field line. This approach to optimizing the magnetic field spectrum finds its ultimate application in the quasi-symmetric and quasi-omnigeneous configurations described earlier in this document.

The radial electric field in a stellarator that satisfies the ambipolarity constraint can have multiple values when the particle fluxes have a nonlinear dependence on the electric field. Theoretically it has been known that the neoclassical transport can be greatly reduced with a large positive electric field, the so-called 'electron root' of the ambipolarity condition. This root has been somewhat elusive in the past; work done by Maassberg analyzing transport in W7-A, L-2 and W7-AS showed that the predicted improvement in the neoclassical transport due to the electron root, considering only thermal particle fluxes, was not supported by the experimental evidence [10]. More recent results however have shown that it can be achieved in at least two experiments, but only with a significant nonthermal electron population present. On CHS the transition from the ion root to the electron root was observed during off-axis second harmonic electron cyclotron heating on a neutral-beam-heated plasma [11]. The off-axis heating in the low field side of the torus gave rise to a trapped electron population that supported an enhanced electron flux. The threshold for the transition between the two roots was qualitatively in agreement with the theoretical model. Also, fast changes (60 μs) between positive and negative radial electric fields were measured in CHS using a heavy ion beam probe [12]. The nonlinear relation between the radial electric field and radial current was thought to be responsible for the bifurcation phenomenon.

In W7-AS clear evidence of the electron root of the ambipolarity condition was observed in discharges where the electron temperature was the highest of any stellarator (see plasma parameters above), T_e = 4 keV [13]. However, it was observed that the electron root (electric fields up to +600 V/cm) and the high electron temperature occurred only with second harmonic ECH at 140 GHz in a magnetic configuration in which there was a substantial ripple on the magnetic axis. Under these conditions a substantial fraction of the ECH power was absorbed by ripple-trapped suprathermal particles that contribute to the ambipolarity condition. The presence of these electrons was supported by Monte Carlo calculations. The experimental electron thermal conductivity was at least an order of magnitude lower with the large positive electric field than if the electric field was assumed to be zero, although somewhat larger than calculated for the electron root. Without the ripple, or at lower ECH power, the positive electric field and the high temperature were not observed.

In W7-AS during combined ECH and NBI, the ion root of the ambipolarity condition has been instrumental in obtaining the highest ion temperature of any stellarator, T_i = 1.6 keV [1]. Electric fields towards the plasma center, on the order of –40 V/cm were sufficient to reduce the neoclassical ion heat diffusivity by more than an order of magnitude and allow access to the $\nu^{1/2}$ regime. Even stronger radial electric fields towards the plasma edge, up to –200 V/cm to –700 V/cm, have been measured in the gradient region in W7-AS [14]. These high radial electric fields (and correspondingly high gradients in E_r), also in agreement with the neoclassical ambipolarity constraint, have been effective in reducing the edge turbulence and have led to energy confinement times that are 2.5 times greater than standard stellarator scaling (see Confinement Scaling and Improvement below). In contrast, the electron root has not been as effective in improving the energy confinement time yet because it is localized towards the plasma center where the transport also tends to be close to neoclassical.

The agreement observed in W7-AS between the measured electric field and the theoretical calculation derived from the ambipolarity constraint, even in regions where anomalous transport dominates, is an indication that the anomalous particle fluxes are intrinsically ambipolar [15]. This result also agrees with measurements made in Heliotron E [7]. The agreement is not quite as good on CHS [16], possibly because of the effect of direct ion orbit losses in this low-aspect-ratio experiment [17]. Measurements of electric fields in stellarators are typically based on flow velocity measurements, with some experimental results obtained with a heavy ion beam probe. Because of the large variation in the magnetic field in the toroidal direction, there is little rotation in this direction for most stellarators including CHS [18], W7-AS [19], IMS [20], and H-1 [21]. The very high toroidal component of the parallel viscosity in stellarators forces the rotation damping in this direction to be neoclassical. This is in contrast with tokamaks where the parallel viscosity in this direction vanishes because of the symmetry and the residual toroidal damping is anomalous. Experiments in IMS demonstrated the competition between damping due to neutrals and damping due to parallel viscosity to determine the radial electric field, momentum decay rate, and plasma flows [20].

Confinement Scaling and Improvement. Anomalous transport plays a dominant role in confinement scaling in stellarators. Although lacking in ohmic current and faced with the possibility of large neoclassical losses at low collisionality due to the lack of symmetry, stellarators have typically shown a magnitude and scaling of transport similar to that of L-mode scaling in tokamaks. In addition, a variety of scaling laws including LHD scaling, gyro-reduced Bohm, and Lackner-Gottardi have been shown to fit the data. The most comprehensive data set including data from ATF, CHS, Heliotron E, W7-A, and W7-AS was used to derive the 1995 International Stellarator Scaling, or ISS95 [22]:

$$\tau_E^{ISS95} = 0.079\, a^{2.21}\, R^{0.65}\, P^{-0.59}\, n^{0.51}\, B^{0.83}\, ɫ^{0.4}$$

where the rotational transform ɫ is the value at r/a = 2/3. Anomalous transport in stellarators is not well understood. Experiments on ATF demonstrated that the energy confinement time was dependent on the radial extent of the magnetic well, suggesting that resistive interchange modes played an important role in determining confinement [23]. Also on ATF, the dissipative trapped electron mode was identified but was observed not to affect the energy confinement time [23].

Methods to improve confinement in stellarators have taken varied approaches. The high-T_i mode was reported in Heliotron E, whereby the ion temperature roughly doubled from 0.4 to 0.8 keV

[24]. In this mode, the density profile was peaked, fueled by neutral beams with low wall recycling. The ion thermal conductivity dropped at the plasma core due to shear in the radial electric field due to the pressure gradient; the poloidal rotation velocity was relatively unaffected. The confinement improvement was on the order of 40%. A similar high-T_i mode was more recently found on CHS as well [25].

The L-H transition has been observed in three stellarators, with characteristics similar to that observed in tokamaks, reduction in H_α emission, increase in plasma density, increase in edge poloidal rotation, and decrease in edge turbulence. However, only marginal improvements in the energy confinement time were observed: 15% in CHS [26] and 30% in W7-AS [27]. In H-1 factors of two increases in ion temperature and plasma density were observed, although the actual increase in energy confinement time is unclear [21,28]. In both W7-AS and CHS, the transition was observed to occur only in a narrow range of rotational transform profiles. This was shown in W7-AS to correspond to local minima in the poloidal viscosity [29]. In CHS, a small ohmic current was used to control the transform. Outside the parameters of the L-H transition, magnetic shear that was controlled with an ohmic current was also found to be important in W7-AS. With low shear and high-order resonances present in the plasma boundary, the electron thermal conductivity was observed to be anomalous over the whole plasma cross-section. With increasing shear, the level of anomalous transport dropped until it was neoclassical out to $r/a = 0.7$ [30].

The best results demonstrating confinement improvement in a stellarator were in W7-AS in discharges that resembled transport barrier formation in tokamaks. This culminated in the highest energy confinement time observed in a stellarator, 55 ms, a factor of 2.5 greater than the ISS95 scaling [4]. Density control with low recycling during moderate levels of neutral beam heating and high ion temperature and plasma gradients were the key to the improved confinement regimes, somewhat similar to the high-T_i modes described above. A strong gradient in the radial electric field, consistent with the theoretical neoclassical ambipolar value, was observed to increase towards the edge region, corresponding to a minimum in the thermal conductivity. The increase in the electric field leveled off when the transport dropped to neoclassical levels in most of the plasma column.

Finite-Beta Behavior. W7-AS is a partially optimized stellarator in that the magnetic field was designed to reduce the Pfirsch-Schlüter current by a factor of two with respect to a conventional stellarator. The reduction in the resulting Shafranov shift as a function of beta was experimentally verified. Pressure surfaces measured by soft x-ray emission corresponded closely to the predictions of 3-D codes [1]. This important result is the first confirmation that stellarators can be designed explicitly about verifiable physics criteria. The next generation stellarator, W7-X, is designed to have small neoclassical losses as well as minimal bootstrap current. Experiments performed on ATF as well as on W7-AS showed that experimental measurements of the bootstrap current agreed with neoclassical calculations [31]. Furthermore, varying the spectral components of the magnetic field could alter the magnitude and direction of the bootstrap current. These results verify the validity of at least some of the W7-X optimization criteria.

The stability limit in stellarators is one of the key unanswered experimental questions. To date, three experiments have achieved average beta values on the order of 2 %: two have surpassed the theoretical stability limit without evidence of instability while one showed evidence of an internal disruption with moderately peaked profiles. All stability explorations in stellarators have been done at fairly low magnetic field. CHS achieved the highest average beta value to date, 2.1 % in a magnetic field of 0.57 T [3]. For this case, the Shafranov shift was 40% of the minor radius. Starting from a configuration in which there was a magnetic hill everywhere, the plasma was shifted to the outboard side and a magnetic well depth developed out to r/a ~ 0.7. In this configuration, the plasma is Mercier unstable at low beta, and becomes Mercier stable above 1.3%. Magnetic fluctuations were observed to increase with beta and then saturate at about 1%. Similar observations had been observed previously on ATF, however at lower beta [32]. The global energy confinement time on CHS did not degrade at the highest beta values.

Average betas of 1.8 % have been achieved in W7-AS at a magnetic field of 1.25 T [5]. Applying a vertical field to reduce the outward shift, the magnetic well depth was reduced in this configuration leading to a configuration that was ideal interchange stable over the whole plasma, but resistive interchange unstable over the outer 1/3 of the plasma. From fluctuation measurements and global plasma parameters, no indication of a stability limit was observed experimentally. Reducing the magnetic well depth even further, thereby extending the region that was resistive unstable, still gave no indication of instability onset.

In an earlier experiment, dating back to the early 1980's, the average beta in Heliotron E reached 2.0% in a magnetic field of 0.94 T [33]. Heliotron E has a broad magnetic hill and is unstable to resistive interchange modes. With high gas puffing and fairly broad pressure profiles, a fairly quiescent plasma could be produced at the highest beta values. With decreased puffing and more peaked profiles, sawtoothing is observed at lower beta leading to an internal disruption: clear evidence of a beta limit for these profiles. Internal disruptions can be driven by the m =1, n = 1 mode near the q =1 surface or by the m = 2, n = 1 modes near the q = 2 surface. More recent measurements showed the disruption dependence on the heating power and plasma density and demonstrated the dependence of the critical pressure gradient for the onset of instability on the resistivity [34]. The unstable mode was identified as a resistive interchange mode.

Plasma Heating. Electron cyclotron heating (ECH) and neutral beam injection (NBI) heating have long been used effectively on stellarators for plasma production and heating, but both heating methods have some disadvantages. High beta is obtained at low B and at high density n because of the confinement improvement with density in stellarators, but the maximum density achievable with ECH varies as B^2 ($\omega_p^2 \quad n \quad \omega_{ce}^2 \quad B^2$), so high beta [$T(n/B^2)$] requires high power. While this is not a problem in a reactor, it is difficult to obtain high beta in an experiment with ECH. NBI heating is effective at high n and low B for reaching high beta, but low ion collisionality studies require high ion temperature at low density and the particle source introduced by NBI makes this difficult. Ion cyclotron range of frequency (ICRF) heating in stellarators offers an additional method of heating without these potential disadvantages. Effective ICRF heating in stellarators is also important for steady-state operation of LHD and W7-X.

Although successfully used in tokamaks, bulk ICRF heating had not been observed in stellarators until recently. Sustained bulk heating with ICRF has now been demonstrated on CHS with a single-strap antenna on the low-field side [35] and on W7-AS with a 1-m toroidally extended antenna on the high-field side [36] in a very different magnetic configuration from that on CHS. The heating efficiency with second harmonic and hydrogen-minority-species plasma heating is similar to that obtained with electron cyclotron heating or neutral beam injection. Sustainment of the plasma density after ECH turnoff with ICRF alone for the duration of the ICRF pulse has been demonstrated on both CHS and W7-AS. In CHS the best discharges were dominated by electron heating. At lower density, ion heating was observed, but the confinement of high-energy trapped ions was poor. For this case, a large impurity influx was observed. For the W-AS results and for electron heating in CHS, there was little increase in the total radiated power.

Particle Control. In stellarators, the density is not limited by disruptions but by low-temperature slow radiative collapse. A comparison of density limits in ASDEX and W7-AS shows that stellarators can achieve higher densities than tokamaks [37], due to the lack of disruptions. However, particle and power handling is a key issue in both. The magnetic geometry of stellarators leads to two natural divertor configurations. Stellarators with continuous helical coils (torsatrons) have a continuous helical separatrix outside the last closed flux surface (LCFS). LHD is an example of this type of stellarator [39]. In modular-coil stellarators such as the Interchangeable Module Stellarator (IMS) [40] and W7-AS [41], the helical strip breaks up into a helical chain of magnetic islands. Hybrid configurations are possible where one or more islands are induced outside the LCFS by currents in specially designed coils, such as those tested on CHS [42] and on the TEXT [43] and JIPPT-IIU [44] tokamaks.

Recent progress has been made on island-based divertors for stellarators. A local island divertor concept which uses an externally produced $m = 1$, $n = 1$ island to avoid the leading edge problem of a pumped limiter by channeling the particle flux into a pump duct has been successfully tested on CHS [42]. In addition to shielding the plasma from incoming gas and impurities and depositing the diverted power on the back of the divertor head away from the leading edge, these experiments also show confinement improvement and the attainment of much higher plasma densities and betas at larger values of the plasma major radius than had been obtainable before in CHS. Understanding of the island structure at the edge of the plasma in W7-AS has progressed to the point where it has been used to design a full divertor system for both W7-AS and W7-X [45]. The W7-X divertor takes the form of a helical stripe that is only on the outside-major-radius side of the plasma where the access is easiest.

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