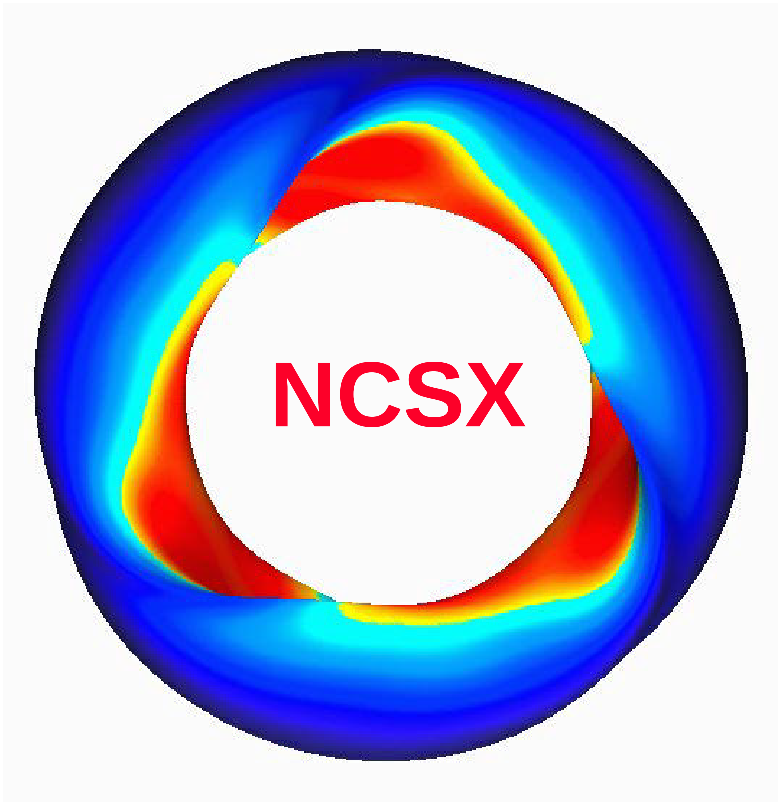
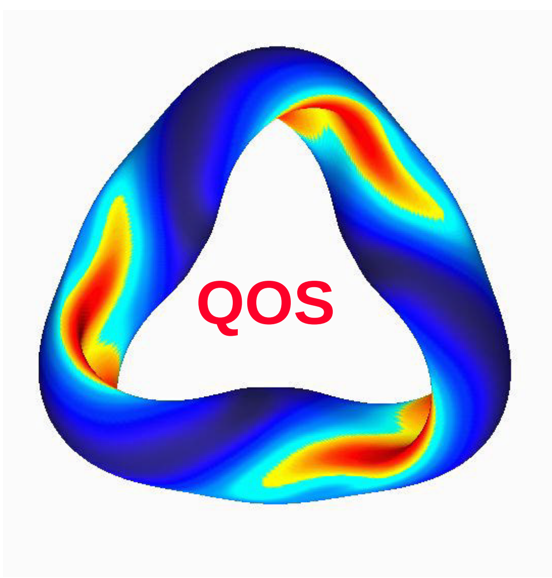
U.S. STELLARATOR PROGRAM

Opportunities for Concept Improvement



NCSX Proof-of-Principle Facility



QOS Concept Exploration Level Experiment

HSX Experiment

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The cover shows to scale the last closed flux surface and contours of constant magnetic field (in color) for the existing HSX and proposed NCSX and QOS Compact Stellarators.

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I. ADVANCED TOROIDAL CONFINEMENT APPROACHES

A key issue limiting progress in magnetic fusion energy (MFE) is the development of a core plasma configuration that can be sustained economically in a steady state. Tokamaks and stellarators are the most advanced of the toroidal magnetic confinement concepts. Together they constitute complementary approaches to MFE development that maximize our understanding of toroidal confinement and the probability of achieving an attractive toroidal fusion reactor featuring steady-state operation at high beta without disruptions. This is a major challenge for toroidal confinement. Because of its difficulty and importance, parallel approaches are necessary to maximize the likelihood of a satisfactory resolution. The advanced tokamak (AT) is one approach. Compact Stellarators with moderate plasma aspect ratio $A_p = R/a = 2–4$ [where R and a are the (average) major and minor radii of the plasma] provide an alternative solution. Research is needed to develop the scientific understanding of both approaches.

Tokamaks have obtained reactor-relevant temperatures, good confinement properties, and reactor-relevant beta values in short pulse operation. In the advanced-tokamak approach, the plasma-created bootstrap current, supplemented by external current drive, is used to create a continuously sustainable configuration. One advanced-tokamak reactor vision is the reverse shear ARIES-RS with $A_p = R/a = 3.1$, volume-average beta $\beta = 5\%$, and neutron wall loading $\Gamma_n = 4$ MW/m^2. Advanced-tokamak physics research is aimed at understanding the plasma control requirements (current and pressure profile control, conducting structures close to the plasma, active feedback of instabilities, rotation control) to sustain a high-performance configuration such as the ARIES-RS without discharge-terminating disruptions. Because nearly all (>90%) of the current is supplied by the bootstrap current, the equilibrium is expected to depend on the pressure and profiles in a highly nonlinear manner. The recycled power fraction and system complexity required to sustain the core will depend on what solutions are ultimately found to be successful in the course of AT research.

Currentless stellarators are inherently steady-state devices without these control requirements and exhibit no disruptions, even at their highest parameters. The most developed stellarator reactor embodiment is the large-aspect-ratio ($A_p = 12$) HSR based on the German Wendelstein 7-X (W7-X) stellarator, also with $\beta = 5\%$, but with a low wall power density ($\Gamma_n = 0.9$ MW/m^2) due to its large size ($R = 22$ m). Stellarators have not yet achieved the level of confinement improvement seen in tokamaks, and the highest value of β achieved thus far is 2%. Nevertheless the similarity in confinement scaling between tokamaks and stellarators and the evidence for improved confinement without disruptions give optimism for the stellarator approach.

II. COMPACT STELLARATORS

Compact Stellarators are hybrid configurations that combine the moderate aspect ratio and good performance of advanced tokamaks with the disruption immunity of stellarators, and hence could lead to a more attractive reactor. Like the AT, the Compact Stellarator approach uses the self-generated bootstrap current to sustain a configuration with plasma aspect ratios and power densities that are tokamak-like. In contrast to the AT, the CS uses the main magnetic field coils to shape the plasma in such a way as to make it stable against disruptions without close-fitting control structures or plasma controls that recycle plant output power back to the plasma. The Compact Stellarator combines the bootstrap current with three-dimensional plasma shaping to obtain the best features of both stellarators (low recirculating power) and advanced tokamaks (compact size and high power density). A recent accomplishment of Compact Stellarator research is the calculation of plasma configurations with reactor-like bootstrap current profiles that are stable without a close-fitting wall to external kink, ballooning, vertical, and Mercier modes at $\langle\beta\rangle$ and A_p in the range of interest (4% and ~3.5, respectively) for power plants. The calculations are based on recent theoretically-developed design strategies for achieving low neoclassical transport and energetic-particle orbit losses in three-dimensional magnetic fields. Coil solutions that reconstruct these plasmas with their key physical properties, while also satisfying engineering constraints, have been computed. The current research is focussed on the concept optimization needed for the experimental facilities that will be used to develop Compact Stellarators over the next decade.

A step toward a Compact Stellarator reactor design, documented in the Stellarator Power Plant Study (SPPS), assumed a stellarator configuration with HSR-like properties but with somewhat lower aspect ratio (A_p = 8). The projected cost of electricity for the R = 14 m SPPS reactor is similar to that for the R = 6 m second-stability tokamak reactor ARIES-IV and the R = 5.5 m reverse-shear tokamak reactor ARIES-RS. The low recycled power of the SPPS reactor off-set its larger size and lower wall power density. Compact Stellarators offer the possibility of a significant further reduction in stellarator size relative to SPPS, while maintaining the advantages of low recycled power and immunity to disruptions. Figure 1 illustrates the reactor vision for the Compact Stellarator approach: a more compact stellarator reactor with power density similar to toka-

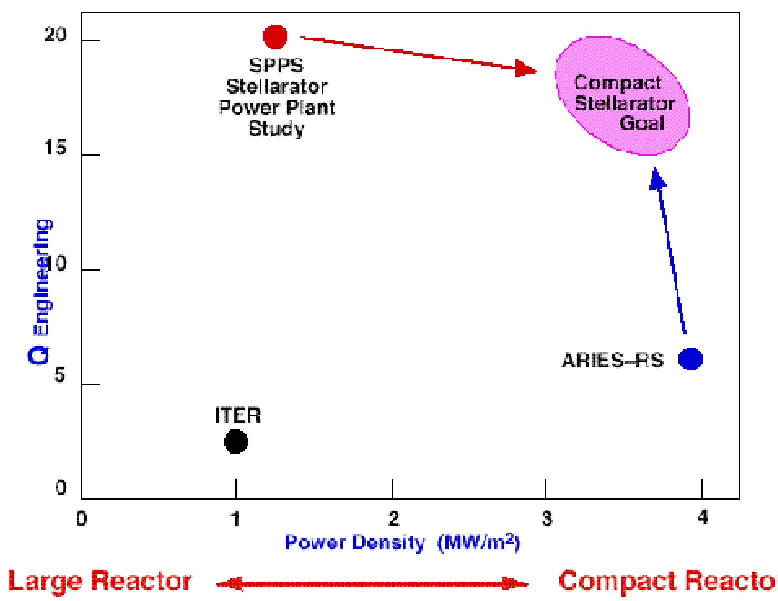

Fig. 1. Potential advantage of a Compact Stellarator reactor.

maks but without disruptions, feedback, or external current drive. The research goal is to determine an optimum set of solutions to the challenging problems of developing a steady-state disruption-free reactor with low recycled power. Experimental research is needed now to establish the scientific foundations for the compact-stellarator approach to this critical challenge.

The long-term goals for compact-stellarator research are:

* immunity to disruptions with a self-consistent bootstrap current in steady-state operation
* thermal plasma confinement a factor >2 better than stellarator L-mode-like scaling
* neoclassical transport << anomalous transport and losses of energetic particles <~10%
* reactor-relevant plasma parameters (T_i > 10 keV, $\langle\beta\rangle$ > 5%, $n\,\tau_E T > 10^{20}$ keV•s•m^{-3})
* compatibility of the bootstrap current (and its control) with operation at high β and low ν^*
* practical steady-state power and particle handling schemes that are extrapolatable to a reactor-relevant configuration
* reactor designs with good plasma-coil spacing and coil utilization.

The key near-term issues are: (1) demonstrating disruption-free operation at high beta (4-5%), (2) understanding what mechanisms limit the beta, (3) demonstrating improved neoclassical transport, (4) exploring improved confinement modes, and (5) developing practical particle and power handling approaches. A Compact Stellarator research program is planned to address these issues.

III. STELLARATOR RESEARCH OPPORTUNITIES

The large ongoing world stellarator effort focuses on currentless plasmas at high aspect ratio, which extrapolates to large reactors. The new Large Helical Device (LHD) in Japan and the Wendelstein 7-X (W7-X) under construction in Germany are ~$0.5-1 billion-class stellarator experiments designed for a level of performance that allows extrapolation to burning plasma devices. They will provide data on divertors, high-power heating, steady-state operation, and superconducting coils that will be relevant to all stellarator concepts.

The U.S. stellarator community, in its 1998 white paper, “U.S. Stellarator Program Plan,” identified an opportunity to explore physics attributes that could improve the vision of stellarator reactors: lower aspect ratio, use of plasma current, and incorporation of magnetic symmetry in the concept design strategy. Compact Stellarators obtain plasma aspect ratios ~1/3 that of conventional stellarators by supplementing an externally-produced rotational transform with that produced by the plasma-generated bootstrap current. This additional degree of freedom allows plasma configurations to be realized with specified physics properties such as stability to various instabilities, reduced neoclassical transport and energetic-particle orbit losses, and rotational transform profile characteristics (shear, well, fraction of transform produced externally), within specified space envelopes and engineering constraints.

Two complementary approaches have emerged: quasi-axisymmetry (QA), which uses the bootstrap current to produce about half of the poloidal field and has tokamak-like symmetry properties, and quasi-omnigeneity (QO), which approximately aligns bounce-averaged drift orbits with magnetic surfaces and aims at a small bootstrap current. The edge magnetic shear in both can be opposite to that of the advanced tokamak, stabilizing neoclassical magnetic islands across the entire profile and permitting higher external kink stability limits without a nearby conducting wall. The extensive tokamak and stellarator database coupled with 3-D design tools that now allow optimization of plasma configurations with specified physics properties and coil optimization codes that include desired engineering properties have led to two proposed experiments, the QA National Compact Stellarator Experiment (NCSX) proof-of-principle (PoP) facility and the Quasi-Omnigeneous Stellarator (QOS) concept-exploration-level experiment. These proposed new experiments would complement the larger world stellarator program that focuses on large-aspect-ratio experiments with negligible net plasma current and produce the physics data base needed to decide whether to proceed to a next step in the Compact Stellarator line.

An initial test of quasi-symmetry is being conducted in a modest-size stellarator (HSX, with R = 1.2 m, $\langle a \rangle$ = 0.15 m, $B \leq 1.3$ T, P = 0.2 MW) at the Univ. of Wisconsin. A small stellarator (CAT-U with R = 0.5 m, $\langle a \rangle$ = 0.1 m, $B \leq 0.5$ T, P = 0.2 MW) at Auburn Univ. will test effects due to plasma current of interest to Compact Stellarators. The NCSX (with R = 1.45 m, $\langle a \rangle$ = 0.45 m, B = 1-2 T, P = 6-12 MW) at PPPL would exploit tokamak-like quasi-axisymmetry and test beta limits at $\langle \beta \rangle \sim 4\%$. The QOS (with $R \leq 1$ m, $\langle a \rangle < 0.3$ m, B = 1 T, $P \leq 3$ MW) at ORNL would test features of the complementary quasi-omnigeneous approach to Compact Stellarator optimization. The device parameters for NCSX and QOS would be competitive with those for other world stellarators, as illustrated in Fig. 2 where the dot sizes are proportional to the plasma cross-sectional area, would allow plasma parameters beyond those achievable in the present U.S. stellarators HSX and CAT-U, and would extend stellarator research to much lower aspect ratios. In Fig. 2 black indicates existing experiments, blue those in construction or modification, and red the new proposed U.S. Compact Stellarators.

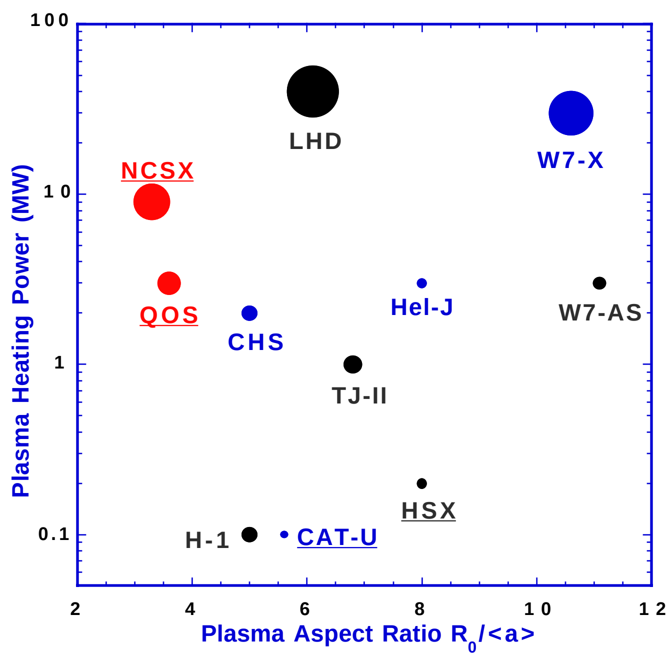

Fig. 2. Comparative sizes and heating powers for the world stellarators.

IV. THE NATIONAL COMPACT STELLARATOR EXPERIMENT

Mission and Role

The National Compact Stellarator Experiment (NCSX) is proposed to answer the key scientific question,

> Can a high-beta low aspect ratio stellarator configuration avoid disruptions in a confinement-optimized configuration consistent with bootstrap currents?

The NCSX is needed to understand the requirements for disruption avoidance in stellarators with aspect ratios (<4) and beta values (3-5%) similar to those of advanced tokamak steady-state reactor studies. Previous stellarator experiments with driven currents at high aspect ratio and low beta showed that disruptions were suppressed by the addition of small amounts of external rotational transform (15-20% of the total). NCSX will test how this extends to high beta. Achieved beta values in stellarator experiments have been limited to ~2% to date. This overall goal sets the basic requirements for the magnetic configuration, size, and performance of the NCSX device. The facility will also provide the experimental flexibility and the diagnostic, heating, and power and particle handling capabilities needed to examine a range of physics questions to assess the future role of Compact Stellarators. The specific scientific goals of the NCSX are to:

1) Demonstrate the ability of Compact Stellarators to operate at $\beta \sim 5\%$ without disruptions, with the rotational transform generated by coil currents and the bootstrap current. Determine the configuration requirements to avoid disruptions at high beta and high density.
2) Determine the beta limit and limiting mechanisms and their scaling with plasma parameters.
3) Determine the adequacy of the neoclassical-transport optimization to ensure good confinement at a reactor scale, and to ensure confinement of energetic particles (e.g. alphas).
4) Determine the ability to control turbulent transport and enhance confinement using flow-shear, the magnetic configuration, and control of particle fueling and radiation. Determine the dimensional and non-dimensional confinement scaling. Compare the observed transport with theoretical predictions and empirical scalings.
5) Test stabilization of neoclassical tearing modes and equilibrium islands at high beta by proper choice of magnetic shear for the bootstrap-current direction.
6) Explore the compatibility of Compact Stellarators with methods to control the power and particle exhaust.

NCSX is the largest element of the proposed U.S. Compact Stellarator program. Its characteristics are consistent with the FESAC model for a proof-of-principle experiment defined in its 1996 Alternative Concepts review and the draft 1999 Criteria, Goals, and Metrics document.

Magnetic Configuration

NCSX uses the quasi-axisymmetric (QA) approach to obtain well-confined drift trajectories in a Compact Stellarator configuration. Its magnetic field structure, while three-dimensional in physical space, possesses an underlying tokamak-like approximate symmetry as seen by energetic particles in the system. This produces tokamak-like drift trajectory confinement and tokamak-like neoclassical transport properties. When compared to other stellarators, the damping of parallel flows is reduced and should allow efficient manipulation of the electric field for controlling turbulence. In a QA stellarator the bootstrap current is comparable to that in a tokamak and adds to the rotational transform produced by external coils. This reduces the rotational transform that must be generated by the coils, simplifying their shape and allowing more space between plasma and coils for a reactor blanket. However, the bootstrap current also non-linearly modifies the equilibrium and complicates control, as in tokamaks. At the beta limit, NCSX has been designed to have ~50% of the rotational transform generated by the coils and ~50% by the bootstrap current, substantially reducing the degree of non-linearity expected relative to advanced tokamaks.

The three-dimensional shape can be tailored to avoid instabilities that could otherwise cause disruptions or limit the accessible beta. NCSX has "reverse shear" (i.e., rotational transform profile monotonically increasing toward the edge) over the entire cross section, which suppresses the unstable growth of magnetic islands (neoclassical tearing modes). The reversed shear also helps partially stabilize the ballooning and kink MHD instabilities. Full stabilization of the Mercier, ballooning, and external kink modes at beta ≥ 4%, without a close-fitting conducting wall or feedback systems, is designed by numerically optimizing the 3D plasma shape using 3D stability codes. Strong axisymmetric shaping (n=0 ellipticity and triangularity) stabilizes the ballooning modes and mild corrugation of the plasma boundary on the low-field side is found to give kink stabilization. Remarkably, the equilibria are calculated to be passively stable to the vertical instability, even for average elongation values well above the tokamak stability limit. These stability properties illustrate the flexibility theoretically available with 3D shaping. During the design of NCSX, many configurations have been explored with differing combinations of characteristics. These include ballooning and kink stable configurations with 7% beta but degraded quasi-symmetry or with non-monotonic rotational transform (i.e. having a shearless surface).

The NCSX experiment design is based on a reference plasma configuration, optimized using three-dimensional equilibrium and stability codes, that is quasi-axisymmetric and passively MHD stable at a stable β value of 4%. The core region (r/a < 0.55) is calculated to be second stable, in that the ballooning and kink modes become more stable as the pressure increases. Its outer boundary shape (Fig. 3) exhibits three-fold toroidal symmetry and has an aspect ratio (A_p = 3.4) compatible with reuse of

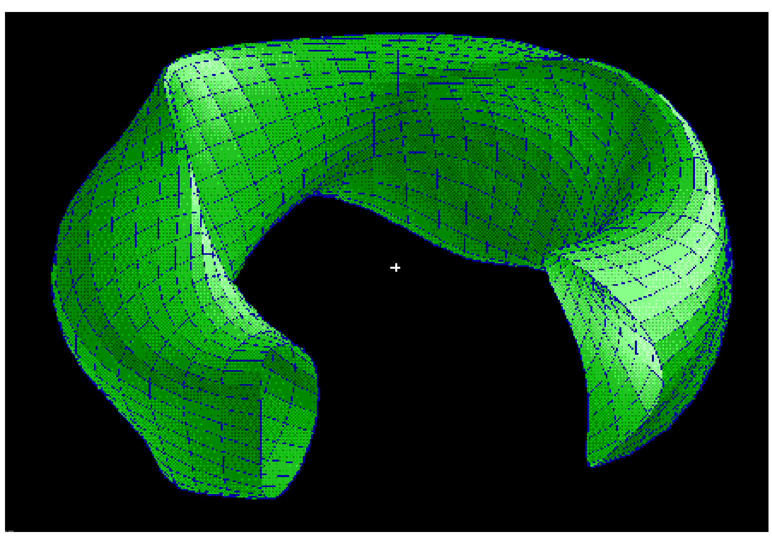

Fig. 3. Isometric of the NCSX QA plasma

components of the PBX-M tokamak at the Princeton Plasma Physics Laboratory. This choice reduces the construction cost by using the toroidal and poloidal field coils of PBX-M to produce the axisymmetric components of the magnetic field, including a toroidal field strength up to 2 T. The non-axisymmetric field components will be produced by a new array of "saddle coils" (seen in Fig. 4). These have been numerically designed to accurately reconstruct the reference plasma with its important physics properties. The use of separate coil sets for the axisymmetric and 3-D magnetic field components provides unusual flexibility for experimentally exploring the physics effects of 3-D shape changes.

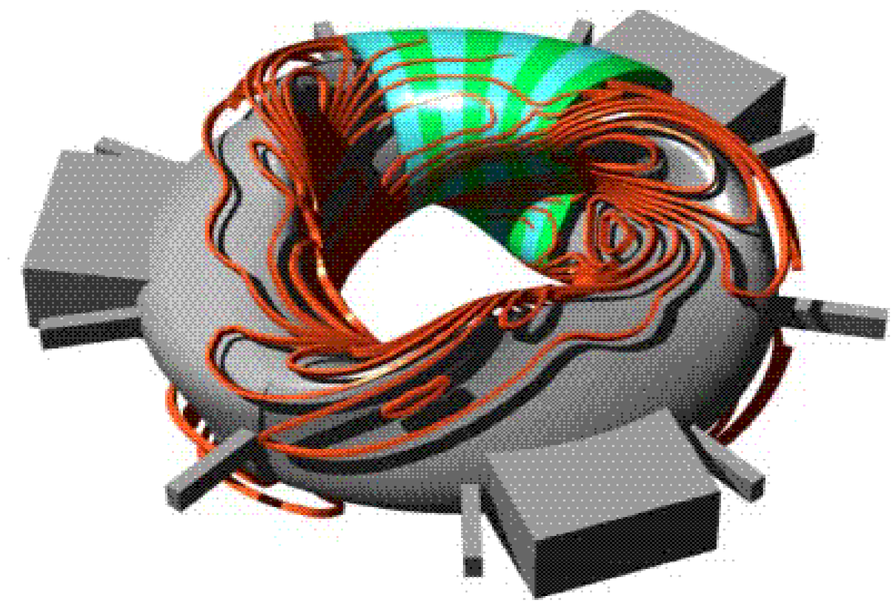

Fig. 4. CAD drawing of the NCSX core, with cutaways to show its construction.

Performance

The NCSX design has adequate plasma size (major radius 1.45 m, minor radius 0.42 m), performance, and heating capability needed to produce and study high-beta operation with significant bootstrap current effects. Taking advantage of existing resources, it will use the existing PBX-M 50-keV neutral beam injectors to provide 6 MW of injected plasma heating power for 0.3-s pulses, and will use available magnet power supplies to operate the stellarator at a magnetic field up to B = 1.2 T. For flexibility, the capability for operation at up to 2 T is a design goal for the device. The plasma heating system can be augmented in power and pulse length, if necessary, by the future addition of up to 6 MW of radio-frequency heating power from available sources.

Since the quasi-symmetry for the designed configurations is only approximate, the neoclassical transport is calculated using Monte-Carlo codes to ensure adequate beam-ion and thermal confinement for a given plasma size. Neoclassical thermal transport in the stellarator fields is simulated (full-f for ions, δf for electrons) using an approximate radial electric potential equal to the ion temperature, as observed on other experiments. The electron neoclassical thermal transport is calculated to be negligible, indicating the near quasi-symmetry obtained. The total neoclassical thermal confinement time is calculated to be at least 4.6 times the empirical scaling global confinement time, see below, ensuring that the neoclassical confinement is adequate.

These calculations of beam-ion and thermal energy losses in the stellarator magnetic fields are combined with empirical confinement scaling projections, such as the ISS-95 scaling derived from the world stellarator data base or the ITER-89P scaling from the tokamak database. The combined calculation predicts that the four beams will be able to heat the plasma to its 4% beta limit at B = 1.2 T if a confinement enhancement of 2.3 times ISS-95 or 1.6 times ITER-89P can be obtained. For this field, approximately 32% of the injected beam power is lost due to exiting orbits. The projected plasma parameters are $n = 10^{20}$ m^{-3}, T(0) = 1.4 keV. For comparison, PBX-M obtained 6.8% beta at B = 1.1 T using 5.5 MW of NB heating with a confinement enhancement of 1.7 times ITER-89P.

Facility Design and Engineering

The NCSX device will consist of a three-dimensional stellarator core assembly (Fig. 4) installed in the PBX-M TF and PF magnet set. A structural shell provides mechanical support, accurate positioning, and cooling of the 3D saddle coils. In operation, the shell and coils will be pre-cooled to liquid nitrogen temperature to reduce resistive losses, and during a pulse the coils will warm adiabatically through joule heating. The shell is assembled around a three-dimensional vacuum vessel. The vessel supports carbon plasma-facing heat-removal structures on its interior surface and is bakeable to a temperature of 350 C to provide the clean vacuum environment needed for high-performance plasma operation. Three large ports and a number of smaller auxiliary ports provide access for the heating, fueling, and diagnostic systems. To assemble the machine, the demountable PBX-M toroidal field coils will be disassembled and then reassembled around the installed core. A cutaway view of the installed NCSX stellarator core and TF magnets with the four PBX neutral beams is shown in Fig. 5.

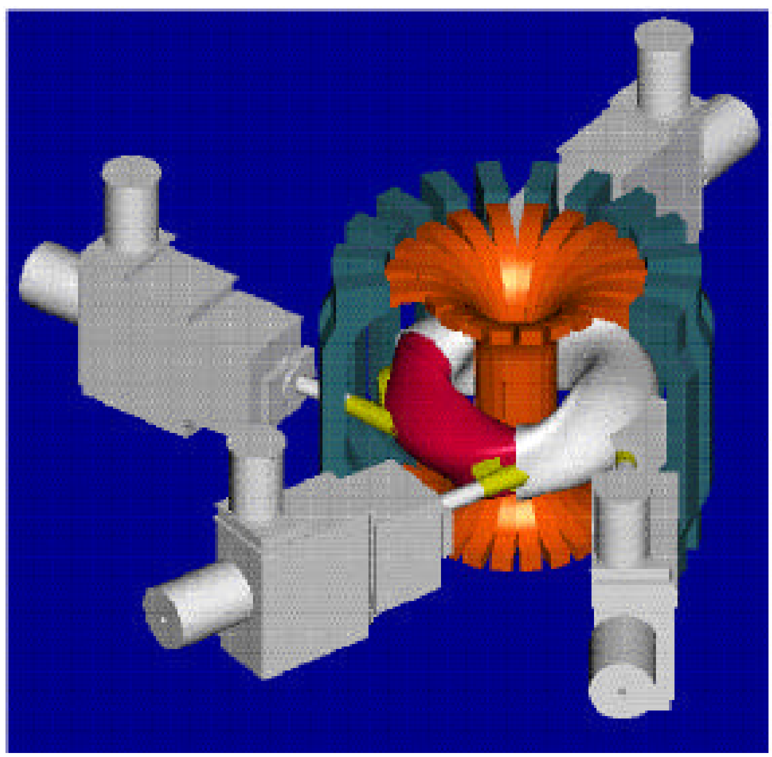

Fig. 5. CAD drawing of the NCSX core in the PBX-M TF coils with neutral beams installed.

At the present time, the NCSX effort is focused on design and R&D to establish the physics basis and a cost-effective engineering embodiment for the experimental facility. The design is being carried out by a national team led by Princeton Plasma Physics Laboratory and Oak Ridge National Laboratory, with many U.S. and foreign collaborators participating. When constructed, the NCSX will be operated as a national research facility, with similarly broad participation, at an anticipated cost of about $20M/year (in FY-1999 dollars). The precise construction cost and schedule will depend on the resolution of remaining design issues and the availability of funding. One possible scenario would have Title I engineering design beginning in October, 2001, first plasma in September, 2005, and a total project cost of $46M (in FY-1999 dollars). Annual funding requirements for NCSX construction will remain well below the $20M/year required for operation.

V. THE QUASI-OMNIGENEOUS STELLARATOR EXPERIMENT

QOS is a low-aspect-ratio stellarator designed as a concept-exploration-level experiment to test the quasi-omnigeneous (QO) optimization approach for Compact Stellarators as a complement to the larger QA NCSX. The QO approach uses a non-symmetric spectrum of magnetic field spatial harmonics to minimize the deviation of bounce-averaged drift orbit (approximate second adiabatic invariant J*) surfaces from magnetic surfaces. This transport optimization has resulted in good confinement of thermal ions (reduced neoclassical transport) as well as confinement of energetic trapped ions needed for some forms of ICRF heating. The QO approach also reduces the pressure-driven bootstrap current to ~1/10 that in an equivalent tokamak, which reduces its relative contribution to the rotational transform (compared to transform from external coils) and thus leads to relative insensitivity of the magnetic configuration as beta changes. These configurations should also be stable against current-driven modes (external kinks), vertical instabilities, and disruptions.

The reference QOS configuration shown in Fig. 6 has three toroidal field periods, a plasma aspect ratio $A_p = \langle R\rangle/\langle a\rangle = 3.6$, a factor of 4 reduction in toroidal curvature over that in an equivalent tokamak (which reduces grad-B drifts at this low A), a large helical axis excursion, high rotational transform (0.78 to 0.91), and a ballooning stability limit of $\beta \sim 4\%$. QO and helias (W7-X) configurations are both drift-optimized configurations with a strong helical deformation of the flux surfaces. However, QO configurations differ in their: low aspect ratio (<R>/<a> = 1/3 – 1/4 that of W7-X; larger plasma current; larger helical component (the dominant term in QOS); and a smaller mirror-like variation of the magnetic field on the flux surfaces (the dominant term in W7-X).

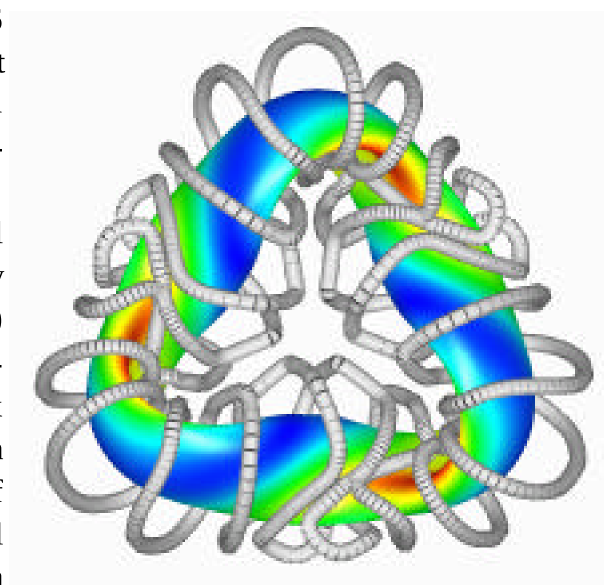

Fig. 6. The QOS plasma and modular coil set.

The modular coil set in Fig. 6 accurately recreates the optimized plasma surface. Here the colors indicate contours of constant magnetic field strength and indicate the helical nature of the QOS plasma. Coils for plasma positioning and additional shaping are not shown. The plasma shaping and the small self-consistent bootstrap current should lead to stability against external kinks and vertical displacements; this is now being evaluated. The device parameters assumed for the scoping study are $R \leq 1$ m, $\langle a \rangle < 0.28$ m, and a 1-s pulse length at $B = 1$ T. QOS could take advantage of the ATF stellarator infrastructure including power and controls, cooling systems, control room, diagnostics, and the plasma heating systems (0.6-MW ECH and 3-MW ICRF).

QO optimization reduces neoclassical transport because cross-field drifts scale with $\langle v_d \bullet \nabla\psi \rangle \propto \partial J^*/\partial\theta$ where v_d is the particle drift velocity, ψ is the flux, and θ is the poloidal angle. Figure 7 shows the result of a Monte Carlo calculation of the particle diffusivity D and heat diffusivity χ for an $N_{fp} = 4$ $A_p = 4.2$ QO configuration. It was obtained by following four groups of test particles with energies 0.5, 1, 2, and 3 keV in a background plasma with 1-keV temperature and 5×10^{19} m^{-3} density with different radial electric fields. The results were integrated over a Maxwellian distribution and the 0-D energy confinement time $\tau_E = \langle a \rangle^2/4\chi$ where χ was obtained from the energy moment in the integration. The values for D and χ decrease with decreasing density (and collisionality ν^*) and do not exhibit the prohibitively large $1/\nu^*$ transport scaling normally associated with ripple-induced losses. For comparison, twice the τ_E^{ISS95} is 11.4 ms at $P = 2$ MW, about the same as the neoclassical value with no ambipolar electric field. The compatibility of QO configurations with ion cyclotron range of frequency (ICRF) energetic-tail heating was simulated by launching 20-keV ions at their turning points at the field resonance and following them collisionlessly. The calculated loss rates were less than that for CHS in which ICRF heating was used successfully.

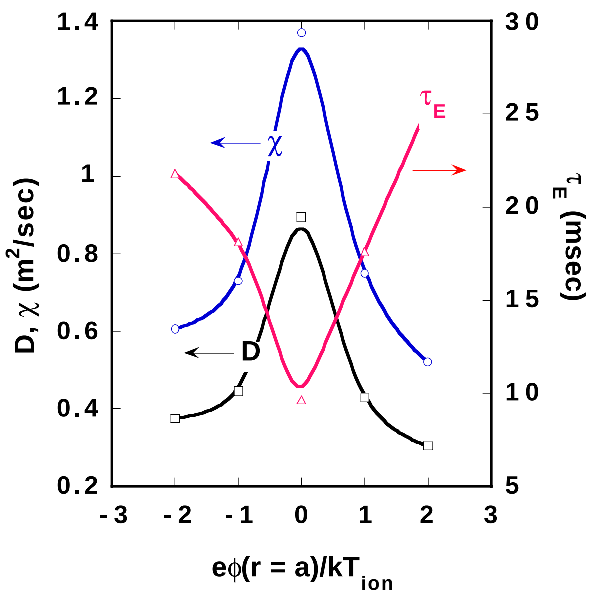

Fig. 7. Dependence of QOS transport on the radial electric field.

Table 1 gives projected QOS parameters based on ISS95 stellarator scaling with a confinement improvement factor H = 2 (H = 1.4 – 3 obtained in W7-AS) and τ_E (neoclassical) > several times τ_E (ISS95) where $\tau_E\text{(ISS95)} = 0.079\langle a \rangle^{2.21} R^{0.65} P^{-0.59} n^{0.51} B^{0.83}\, \bar{\iota}^{0.4}$. Although QOS would not be able to test its 4% beta limit, 1-MW heating and H = 2 would allow $\langle\beta\rangle$ = 1-3%, close to or above that obtained in CHS and W7-AS. This value of beta is sufficient to study the configuration dependence on beta and the reduction and control of the bootstrap current. The mission of QOS would be to test reduction of: (1) neoclassical transport via nonsymmetric QO, and the effect of electric fields on confinement; (2) energetic orbit losses in non-symmetric low-aspect-ratio stellarators; (3) the bootstrap current, its control, and the configuration dependence on β; and (4) anomalous transport by methods such as sheared E × B flow, and to understand flow damping in non-symmetric magnetic configurations.

The QOS project is performing scoping studies in preparation to start design work in FY-2000. The scope is defined by a concept-exploration-level budget. The target for the Total Project Cost is ≈8 M$ in FY-1999 dollars based on rough scaling from HSX and ATF, a three-year construction schedule (2001-2003), and experience with ATF. First plasma is projected for Sept. 2003 assuming a 0.65 M$ budget in FY-2000. The total QOS budget would rise to ≈4 M$ after first plasma.

Table 1. Consistent Sets of Plasma Parameters for QOS with $\langle R \rangle = 1$ m and $B = 1$ T.

Plasma Parameter	**0.4-MW ECH**		**1-MW ICRF**
Line-Average Density n_e (10^{19} m^{-3})$_e$	1.6[a]	3.2[b]	11
Central Electron Temperature, T_{e0} (keV)	3.4	2.4	1.0
Central Ion Temperature, T_{i0} (keV)			1.0
Energy Confinement Time, τ_E (ms)	17	24	26
Volume-average beta, β (%)	0.7	1.0	2.8

[a] 2nd harmonic X-mode; [b] O-X electron Bernstein wave mode conversion

VI. THE HELICALLY SYMMETRIC EXPERIMENT

The Helically Symmetric Experiment (HSX) is the principal element in the U.S. stellarator program at the present time. 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 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 (QHS) device, and the only device of this type in the world program. The physical parameters of HSX are shown in Table 2. 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%.

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	
Edge	1.12	(with 100 kW absorbed)	~ 1 keV
Magnetic well depth	0.6%	Energy confinement time (LHD)	2 ms
Magnetic field strength	1.37 T	Plasma electron β	0.3%
Magnet flat-top (full field)	0.2 s	ν^*_e	<0.1

QHS configurations have an effective transform given by the number of field periods minus the actual transform, | N-ι |; for HSX with four field periods and near unity transform, ι_{eff}~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 above and other elements of the experimental program:

- Reduction of Pfirsch-Schluter and bootstrap currents; small finite beta effects on the magnetic field spectra and equilibrium.
- Smaller poloidal gyroradius with accompanying improved confinement of high-energy particles; HSX can fit as many poloidal gyroradii 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.
- The high effective vacuum transform and very low plasma currents provide for a clear separation of confinement and heating issues.

HSX is an extremely flexible device. 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 can be varied by a factor of 3, with only small changes in the neoclassical transport. In the mirror mode, direct losses are dramatically increased and the neoclassical electron thermal conductivity jumps 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.

The experimental program will also provide insight as to the mechanisms by which the E × 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) 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 level of symmetry.

HSX will use 200-kW 28-GHz ECH to heat the electrons into the collisionless regime for the first part of the HSX program. This heating does restrict HSX to operation at 1.0T or 0.5T and densities less than 10^{13} cm^{-3}. The high effective transform and good trapped-particle confinement in HSX should permit effective ion or electron heating in the QHS mode in a later stage with ICRF heating.

VII. THE COMPACT AUBURN TORSATRON UPGRADE

Parameter	**CAT**	**CAT Upgrade**
Major Radius (m)	0.53	0.53
Minor Radius (m)	0.15	0.15
Avg. Plasma Radius (m)	0.11	0.11
Magnetic Field (T)	0.1	0.5
Density (m^{-3})	$7x10^{15}$	$0.5 – 1 x 10^{19}$
Avg. Elec. Temperature (eV)	10	250
Ion Temperature (eV)	0.5	≤ 50
Plasma Current (kA)	0	25
Input Power (kW)	2 (ECH)	150 (ICRF); 50 (OH)
Pulse Duration (s)	120	0.4 (magnets); 0.1 w/ OH
Edge Transform	0.15–0.7	0.15–0.7 (vac);+ 0-0.5 (OH)
Plasma β (%)	~ 0	0.5-1

Table 3. Parameters of the present CAT experiment and those expected for CAT upgrade.

The issue of disruptions in stellarators will be systematically investigated in the upgraded Compact Auburn Torsatron (CAT) at Auburn University. Table 3 summarizes the new parameters expected of the upgraded CAT plasmas compared with the present parameters. With an aspect ratio of A_p = 5, CAT, along with the Compact Helical System (CHS) in Japan, has the lowest aspect ratio of any existing stellarator. Moreover, CAT has a rotational transform profile similar to the optimized Compact Stellarators being proposed for the US stellarator program. The vacuum rotational transform of the CAT can be varied considerably because of its two, separately-controllable helical field coils. Since the onset of current-driven resistive and kink instabilities that can lead to disruptions is determined largely by the rotational transform profile, the flexible CAT device is ideally suited to carry out exploratory studies of stellarator disruptions that will be relevant to the design and operation of the innovative Compact Stellarators. The experimental program is summarized below:

- Investigate the MHD stability of ohmic currents in a Compact Stellarator plasma over a wide range of magnetic field configurations.
- Measure the stability of both peaked ohmic current profiles and transiently hollow, bootstrap-like current profiles.
- Measure the onset and growth of current-driven resistive tearing and kink instabilities as a function of plasma current, pressure, and vacuum rotational transform.
- Perform ICRF plasma generation and electron heating.
- Study pressure-driven external kink instabilities in finite β (~1%) plasmas.

VIII. INTERNATIONAL COLLABORATION

Collaboration with the larger international stellarator program on selected topics is an important element of the U.S. stellarator PoP program because it provides information on stellarator concept improvement that is not otherwise available in the U.S. program. The international stellarator program is already at the *proof-of-performance* stage for the currentless large-aspect-ratio stellarator approach. It features billion-dollar-class facilities now operating in Japan (LHD) and under construction in Germany (W7-X, 2006) that 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. These large facilities are supplemented by *proof-of-principle* ($30-100 million scale) experiments in Japan (CHS), Germany (W7-AS), Spain (TJ-II), etc.

Experimental Collaborations. The wide range of stellarator configurations accessible on LHD, W7-AS, CHS, and TJ-II allows study of the role of different aspect ratios, degree of helical axis excursion, magnetic-island-based divertors, and the consequences of a net plasma current, elements that are incorporated in the low-aspect-ratio QA and QO stellarator concepts. 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. Study of these issues at higher aspect ratio (A = 5-11) and low bootstrap currents in foreign experiments complements the U.S. stellarator PoP program, which focuses on lower aspect ratio (A = 3-4) and larger bootstrap current.

LHD (Japan). The order of magnitude increases in plasma volume, heating power, and pulse length of LHD over that of existing stellarators allows studies of size scaling and stellarator physics at more reactor-relevant parameters ($\beta \geq 5\%$, T_i ~ 10 keV, τ_E hundreds of ms, etc.). Studies on LHD include: (1) ion heating, neoclassical and anomalous transport, and orbit confinement for understanding the effectiveness of neutral beam and ICRF heating; (2) energetic-particle orbit losses and their amelioration by tailoring the magnetic field spectrum and by ambipolar electric fields; (3) electron heating and transport; (4) beta limits; and (5) the spatial distribution of impurity radiation and the overall power flow in LHD.

W7-AS (Germany). Confinement improvement and a magnetic-island-based divertor system are being studied in W7-AS in magnetic configurations complementary to that of LHD.

CHS (Japan). CHS allows study of transport and beta limits at plasma aspect ratios as low as 5.

TJ-II (Spain). TJ-II allows study of beta limits and transport in a stellarator with a large helical axis excursion, an important ingredient in U.S. stellarator configuration optimization.

Theory Collaborations. Collaboration on stellarator theory and computational tools development benefits U.S. efforts in support of Compact Stellarator concept development. Areas for international collaboration are MHD equilibrium; Mercier, ballooning and kink stability; microstability; bootstrap current; transport; optimization techniques; coil design; and effects of magnetic islands. Institutions in several countries contribute to the U.S. Compact Stellarator program: IPP-Greifswald (Germany); CRPP-Lausanne (Switzerland), NIFS-Toki (Japan), Kyoto University (Japan), CIEMAT-Madrid (Spain), Universidad Carlos III, Madrid (Spain), NPFRF-Canberra (Australia), and Kurchatov-Moscow (Russia).

IX. SYSTEM STUDIES

Integrated physics and engineering systems studies can assess the reactor potential of Compact Stellarators and set criteria that they should meet to be an attractive reactor. These capabilities have been developed in previous U.S. stellarator reactor studies and in the ARIES tokamak reactor studies. The most recent stellarator reactor example is the U.S. Stellarator Power Plant Study (SPPS), a "scoping study" at a smaller scale than the typical ARIES study. The four-field-period SPPS configuration has physics properties similar to the W7-X configuration, but allows reducing the reactor size from R_0 = 22 m (for the W7-X-based HSR) 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. Compact Stellarator PoP program.

The SPPS configuration extrapolated to a reactor power plant that was economically competitive with the second-stability ARIES-IV tokamak reactor assuming that stellarators have the same unit costs for components with complicated geometry as tokamaks and that tokamaks have the same availability as stellarators. Reducing the plasma aspect ratio should lead to significant cost reductions through reducing the mass of the most expensive parts of the fusion reactor core (the first wall, blanket, shielding, and other components that scale with the plasma surface area).

Studies are needed to assess the potential advantages and design issues for Compact Stellarator configurations as fusion power plants relative to conventional stellarators and tokamaks. Initial scoping studies are needed to examine the differences between QA and QO configurations as fusion power plants; in particular the design consequences of the higher degree of spatial non-axisymmetry for QO configurations and the startup and control consequences of the larger bootstrap current for QA configurations. An in-depth study (similar in scope to the ARIES studies) would clarify the trade-offs on more issues than were possible in the SPPS, and would clarify the directions for Compact Stellarator research beyond the proof-of-principle stage. The areas that need to be explored in detail 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 optimiza tion to guide the development of self-consistent attractive reactor configurations.