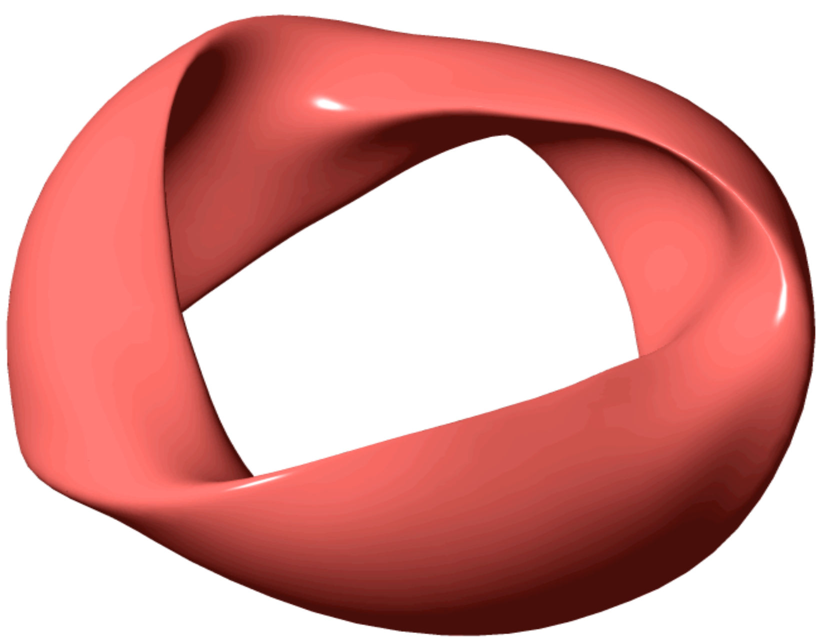
U. S. COMPACT STELLARATOR PROGRAM

Advancing the Understanding of Toroidal Confinement for Concept Improvement



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The cover shows the last closed flux surface for a three-field-period quasi-symmetric stellarator.

1. The U.S. Compact Stellarator Program

The U.S. compact stellarator program aims to improve our understanding of toroidal confinement through the use of three-dimensional (3-D) magnetic confinement concepts to resolve important plasma physics issues and to explore approaches that may lead to a more attractive fusion reactor concept. It complements the larger world stellarator program by extending research to lower aspect ratio, much as the spherical torus (ST) has done for the tokamak, and by incorporating magnetic quasi-symmetry into stellarator optimization.

1.1. Long-term program goal.

The U.S. fusion program has both science and energy objectives to which the compact stellarator program contributes. The U.S. compact stellarator program uniquely integrates four features in experiments: compactness (low aspect ratio), quasi-symmetry (low ripple and flow damping), good flux surfaces (finite, but low plasma current), and 3-D plasma shaping flexibility. The goal is steady-state disruption-immune toroidal plasmas with performance comparable to, or better than, that of tokamaks. This is possible because advances in 3-D theory and computation allow design of optimized configurations. The program is motivated by the excellent results from larger-aspect-ratio stellarators without benefits of quasi-symmetry optimization.

Compact stellarators offer solutions to steady-state burning-plasma challenges. Steady-state-compatible, quiescent, high-beta plasmas without disruptions have already been demonstrated in stellarators. Compact stellarators provide an alternate solution to the high-bootstrap-fraction Advanced Tokamak approach and would allow ITER to lead to the next step (DEMO), even if disruption-mitigated, steady-state, high-bootstrap-current operation is not fully attained. Stellarators have soft operating limits; they are not disruptive. This allows higher density operation that: allows a low-temperature plasma edge, which should ease divertor design; decreases the drive for fast-ion instabilities; and provides alternative solutions for ITER challenges. On the other hand, the orbit physics and turbulent transport physics of quasi-symmetric stellarators is directly connected to tokamak understanding. Thus, compact stellarators contribute to, and benefit from, ITER understanding.

The ultimate goal is a steady-state fusion reactor with no disruptions, no near-plasma conducting structures or active feedback control of instabilities, no current or rotation drive (for minimal recirculating power), and high power density (for economic competitiveness). Stellarators have distinct advantages: inherent steady-state capability with no disruptions, fully ignited operation with no power input to the plasma, and no need for rotation drive or feedback control of instabilities. On-going reactor studies show that compact stellarator reactors can also be comparable to tokamaks in compactness.

1.2. General features.

Toroidal confinement devices obtain their good confinement properties through a helical magnetic field on nested magnetic surfaces. In axisymmetric configurations (tokamaks, STs and reverse field pinches) the helical field is created by a combination of externally-driven and plasma-generated currents. Conventional stellarators create their confining magnetic field through currents in external coils. They have higher plasma aspect ratio, which leads to larger devices, and a 3-D plasma shape that provides the required magnetic field line geometry. Since their confining magnetic field is not produced by plasma current, stellarators do not the face the challenges of plasma current generation and control and disruption avoidance inherent to axisymmetric toroidal devices.

Compact stellarators rely on the 3-D geometry of stellarators and magnetic quasi-symmetry to obtain improved confinement properties. Magnetic quasi-symmetry (near uniformity of magnetic field strength in the symmetry direction) leads to low flow damping in that direction, which allows for large flows to shear apart turbulent eddies, and to electric fields that improve confinement. Self-generated bootstrap currents can flow and provide some of the confining magnetic field in quasi-symmetric stellarators. These devices have been optimized to have very small effective magnetic field ripple, which leads to low transport losses. Correlation between low effective ripple and improved energy confinement time has been observed in existing stellarator experiments.

Stellarators have pioneered the use of computer simulation to target specific physics properties (such as stability to MHD modes at high beta, low ripple through quasi-symmetry, low aspect ratio, and good magnetic surfaces) in the design of magnetic confinement configurations. This capability has been made possible through advances in physics understanding and computational capability. Research in compact stellarators will provide an important test of the ability of advanced scientific computation to accelerate progress in fusion research through the design of well-targeted experimental devices.

1.3. Program elements.

Section 2 discusses how the U.S. compact stellarator program contributes to advancing the physics understanding objectives and the long-term energy goals of the fusion program within the context of the U.S. program priorities. The other sections describe the particular features of each program component, its status/results, and how it specifically contributes to the physics issues discussed in Section 2 in the 5-10 year timeframe.

The National Compact Stellarator Experiment (NCSX), now under construction at Princeton Plasma Physics Laboratory, is the largest element in the US compact stellarator program. It employs quasi-axisymmetry (Q-A) to combine stellarator and tokamak features and obtain good physics properties, as discussed in Section 3. As a proof-of-principle experiment, NCSX will provide an integrated test of quasi-axisymmetry at higher plasma parameters through its high neutral beam heating power and high magnetic field strength. Figure 1 illustrates the 3-D shaping of the plasma surface and external field coils employed in compact stellarators. The Quasi-Poloidal Stellarator (QPS), now in the R&D and prototype development phase at Oak Ridge National Laboratory, complements NCSX by employing quasi-poloidal (Q-P) symmetry, combining stellarator and magnetic mirror features, to obtain good physics properties, as discussed in Section 4. QPS will extend understanding to the lowest aspect ratio and employ radio frequency heating to obtain high plasma parameters. Both experiments are scheduled to start operation before the end of the decade.

Two existing stellarator experiments, the Helically Symmetric Experiment (HSX) at the University of Wisconsin-Madison and the Compact Toroidal Hybrid (CTH) at Auburn University, are verifying the physics basis for the compact stellarator program in addition to exploring basic stellarator physics issues. HSX, discussed in Section 5, has demonstrated confinement improvement due to quasi-helical (Q-H) symmetry and reduction of flow damping.

CTH, discussed in Section 6, will explore the effects on MHD stability of added plasma current and test the 3-D equilibrium reconstruction techniques need to analyze compact stellarators.

The U.S. compact stellarator program is part of a larger world stellarator program that includes large (billion-dollar-class) high-power experiments that do not have the unique features of the U.S. program: quasi-symmetry, low aspect ratio, and incorporation of the bootstrap current. The two largest stellarator experiments, the Large Helical Device (LHD) in Japan and Wendelstein 7-X (W 7-X) under construction in Germany, are large devices with superconducting coils that aim for high plasma parameters, steady-state conditions, and divertor operation. The W 7-X will test a high aspect-ratio currentless optimization approach. As a component of its compact stellarator program, the U.S. collaborates on these and other international stellarator experiments to take advantage of capabilities not available on its domestic experiments, as discussed in Section 7. Figure 2 illustrates how compact stellarators and their supporting facilities bridge between the high-aspect-ratio currentless stellarators and the high-current toroidal devices with lower aspect ratio.

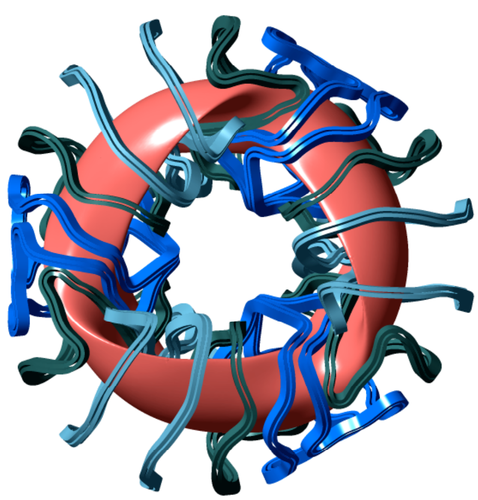


Fig. 1. NCSX plasma surface and coils.

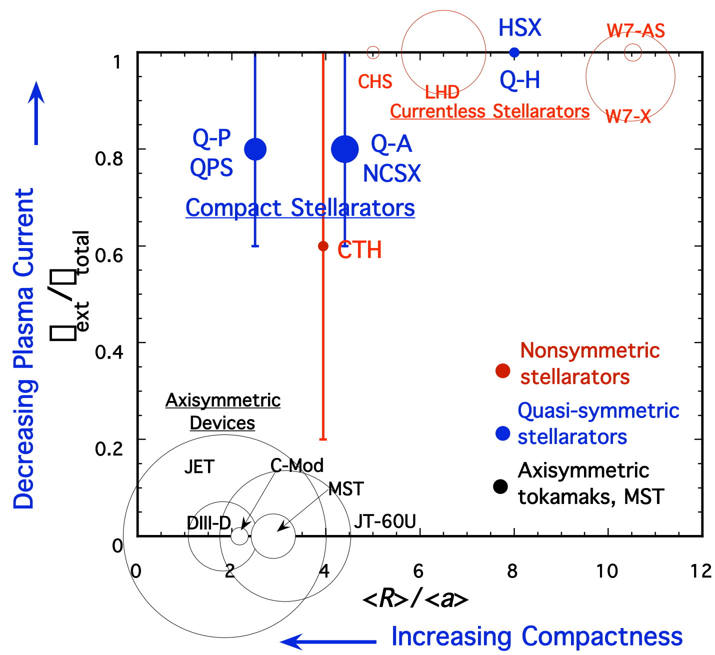


Fig. 2. Relationship of compact stellarators to other magnetic confinement concepts.

Stellarator theory research, carried out as part of the U.S. theory program, is also a key component of the U.S. stellarator program, as discussed in Section 8. Its roles are: advancing the fundamental understanding of 3-D plasma physics phenomena; developing numerical analysis tools, such as equilibrium, MHD stability, turbulence simulation, and boundary modeling codes, for experiment design and data analysis; and providing integrated simulations that enable knowledge gained in stellarator research to be applied to other configurations such as tokamaks, and vice versa.

Stellarator power plant studies (Section 9), carried out as part of the U.S. advanced design (ARIES) program, are used to develop an evolving vision for compact stellarator power plant designs based on current knowledge and highlight high-leverage R&D issues for compact stellarator physics research.

2. Advancing Toroidal Confinement Understanding and Concept Improvement

The unique features of compact stellarators (the ability to vary the degree of externally-generated vs. internally-generated poloidal field, the type and degree of quasi-symmetry and flow damping, the amount of magnetic field ripple, the wide range of plasma aspect ratios, etc.) allow these devices to make significant contributions to the US fusion program.

2.1. Compact Stellarators address key objectives of the U.S. fusion program

The Compact Stellarator is central to OFES's program to advance Fundamental Understanding and Configuration Optimization, fusion thrust areas outlined in the *Report of the Integrated Program Planning Activity for the DOE's Fusion Energy Sciences Program*. One of that report's policy goals adopted by DOE was to "Develop fusion science, technology, and plasma confinement innovations as the central theme of the domestic program." In particular the report sets as one of the ten-year goals determining the attractiveness of a compact stellarator.

Similarly, the compact stellarator program addresses the topical scientific questions identified in the 2004 *Interim Report of the Panel on Program Priorities for Fusion Energy Sciences Advisory Committee* (FESAC), in particular:

1. How does magnetic field structure impact fusion plasma confinement? This requires understanding the role of plasma shaping on plasma confinement, the effect of magnetic structure within the plasma, and the effect of self-generated currents.

2. What limits the maximum pressure that can be achieved in laboratory plasmas? This requires research on equilibrium limits and the onset of magnetic stochasticity.

3. How much external control versus self-organization will a fusion plasma require? This requires understanding and controlling pressure-gradient-driven plasma currents and flow self-organization and understanding the use of dominant external control (*e.g.* externally generated confining magnetic fields or flows).

4. How does turbulence cause heat, particles, and momentum to escape from plasmas?

5. How are electromagnetic fields and mass flows generated in plasmas?

6. How to interface hot plasma with room temperature surroundings?

Each of these thrusts requires an integrated effort in experiment, modeling, and theory.

2.2. Compact Stellarator Program Physics Issues

The physics issues that the U.S. compact stellarator program addresses in advancing toroidal confinement understanding and concept improvement are equilibrium, transport, MHD stability, particle/power control, configuration optimization, benchmarking and improvement of 3-D theory, and reactor concept improvement. An important advantage for transport and stability studies is the unique ability that compact stellarators have to vary important magnetic configuration properties over a very wide range: the 3-D shaping and effective magnetic field ripple, the fraction of trapped particles, the amount and sign of magnetic shear, the type and degree of quasi-symmetry, the degree of viscous damping and flow shear, the size (and sign) of the ambipolar electric field and internal transport barriers, the size of magnetic islands and ergodic regions, and internal vs. external rotational transform.

Equilibrium. Equilibrium limitations at low aspect ratio and finite pressure, particularly the effect of magnetic islands and ergodic regions, are important in stellarators not only in determining beta limits but also in defining the confined volume and determining the overall confinement time. Equilibrium is limited by the onset of magnetic stochasticity. While compact stellarators are designed to maintain good surfaces at high beta, minimization and control of magnetic islands can be done with bootstrap current and Ohmic current tailoring of the transform profile and the self-stabilizing effect of the plasma current. The type of quasi-symmetry and the aspect ratio affect the magnitude of the bootstrap current. The degree of quasi-symmetry affects local bootstrap current generation, which in turn affects the rotational transform profile, and hence the local equilibrium and MHD stability. The compact stellarator program will explore the non-linear relationship of configuration to equilibrium.

MHD Stability. Beta limits and the characterization of MHD instabilities (ballooning, kink, vertical mode, etc.) are not well understood in stellarators. Current data indicates that the plasma beta in stellarators is not limited by instabilities. Quiescent plasmas are routinely observed well above linear stability thresholds. Theory shows that the character of MHD instabilities differs from that in tokamaks. A particular advantage of stellarators that needs to be demonstrated in compact stellarators is disruption avoidance.

Transport. Understanding and controlling flows is important for confinement improvement. Quasi-symmetry and reduced flow damping in the direction of symmetry allows large flows that can shear apart turbulent eddies and reduce anomalous transport. The corresponding electric fields and their effect on flows also affect neoclassical and anomalous transport. The large reductions in the effective helical ripple due to quasi-symmetry and higher effective rotational transform in the U.S. experiments are predicted to greatly reduce neoclassical transport from that in conventional stellarators. Futhermore, edge and internal transport barrier dynamics are a subject of broad interest for toroidal confinement systems. Related issues are edge fluctuations and particle transport, and their control with electrostatic potential bias. The scaling of confinement with configuration parameters (effective ripple, aspect ratio, type and degree of quasi-symmetry, etc.) has not been explored at low aspect ratio. This is needed both for providing insight into other magnetic configurations and for extrapolation to future devices. Energetic ion confinement and the efficiency of neutral beam and RF heating are also important for configuration optimization.

Particle/Power Control. Edge and scrapeoff layer control is affected by the presence of magnetic islands, ergodic regions, and diverted field lines at the edge of the plasma. Compact stellarator's controllable 3-D shaping flexibility allows using these features, or combinations, for optimizing divertors. The use of divertor and baffle plates to control recycling and to handle the power on these components is planned for compact stellarators and would have a large impact on the reactor feasibility of this concept.

Configuration Optimization: The unique configuration flexibility of compact stellarators allows study of the above issues through variation of the: effective ripple, trapped particle fraction, amount and sign of shear, degree of quasi-symmetry, degree of viscosity and flow shear, ambipolar electric field and internal transport barriers, bootstrap vs. Ohmic current, and magnetic island size. Reconstruction of 3-D equilibria from internal and external measurements is essential for understanding the plasma parameters and geometrical features. The configuration flexibility of compact stellarators will help advance the understanding and quantification of the minimum acceptable quasi-symmetry to maintain good particle and energy confinement and to design tradeoffs of quasi-symmetry against other physics and engineering properties.

Benchmarking and Improvement of 3-D Theory. The 3-D nature and low aspect ratio of compact stellarators drives 3-D theory in the areas of plasma equilibrium, transport and stability. The challenges are to improve modeling of plasmas in which the toroidal and poloidal variations are strongly coupled, to understand nonlinear mode saturation and the character of instabilities, and to include magnetic islands and 2-D variations within a flux surface.

Reactor Concept Improvement. Compact stellarators have the potential for an attractive fusion reactor concept because of their inherent steady-state capability, fully ignited operation requiring no power input to the plasma, and avoidance of plasma current related problems. The ARIES group is assessing this potential. A previous large-aspect-ratio stellarator reactor configuration studied by the ARIES group was found to be competitive with tokamak reactors. The lower aspect ratio (leading to smaller size) and edge geometry of compact stellarators may further improve this reactor concept. However, there are a number of physics issues that need to be resolved to which the compact stellarator program can contribute: beta limits, improvement of thermal energy confinement, alpha-particle confinement, configuration optimization including the bootstrap current, and the divertor geometry.

2.3. Impact on other areas of physics. The theoretical and calculational techniques developed for compact stellarator optimization have applications to other areas of physics. Simple coils and a low-aspect-ratio plasma was designed for the Columbia Non-neutral Torus for study of non-neutral and electron/positron plasmas. Space plasmas are inherently 3-D in nature. The theoretical tools that have been developed can be applied to studies of solar flares and the structure of galaxies.

3. National Compact Stellarator Experiment

1. Mission

The National Compact Stellarator Experiment is the key component of the compact stellarator program. Its role to test and advance the understanding of a range of compact stellarator physics issues, encompassing MHD equilibrium and stability, turbulence and transport, and divertors. Its mission is to acquire the physics knowledge needed to evaluate the compact stellarator as a fusion concept, and to advance the understanding of 3D plasma physics for fusion and basic science. The NCSX is a quasi-axisymmetric stellarator (QAS), in which the single particle trajectories and plasma flow damping are similar to those in tokamaks. Based on this fundamental similarity, quasi-axisymmetric stellarators are expected to share the tokamak's good confinement performance. Also, because of this physics link with tokamaks, the compact stellarator has advanced relatively rapidly and economically by building on advances in the more mature tokamak concept.

The NCSX will advance understanding of three-dimensional plasma effects important to toroidal magnetic configurations generally. It will make strong contributions to resolution of the scientific issues facing magnetic fusion energy (MFE), in particular the critical questions identified by the FESAC Priorities Panel, for example:

- How does magnetic structure impact confinement? The NCSX, with its flexibility to vary important magnetic configuration properties (discussed below), will test the effects of 3D shaping and magnetic shear on confinement. It will test the effectiveness of quasi-axisymmetry and permit comparisons with axisymmetry.
- What limits maximum pressure? The NCSX will test whether 3D shaping can increase the β limit and whether reversed shear is beneficial. It will be used to study β-limiting mechanisms, and their control, with 3D fields.
- External control and self organization In NCSX, the internally- and externally-generated fields can be varied. Experiments will examine how high a bootstrap fraction is controllable and under what conditions disruptions are eliminated.
- Turbulent transport The NCSX will test the effects of 3D shaping on turbulent transport and whether reversed shear helps stabilize turbulence in 3D.
- High-energy particles interacting with plasma The NCSX will test the effects of 3D shaping on energetic-ion instabilities, and whether such instabilities can be stabilized?
- Plasma-boundary interfaces The NCSX will test various edge strategies involving, e.g., diverted field lines, island divertors, and ergodic layers, that are available in 3D configurations, and their compatibility with good core performance.

The NCSX mission in support of compact stellarator development goals is to:

- Demonstrate conditions for high-beta disruption-free operation, compatible with bootstrap current and external transform in a compact stellarator configuration.

- Understand beta limits and limiting mechanisms in a low-aspect-ratio current-carrying stellarator.
- Understand reduction of neoclassical transport by quasi-axisymmetric (QA) design.
- Understand confinement scaling and reduction of anomalous transport by flow-shear control.
- Understand equilibrium islands and stabilization of neoclassical tearing-modes by choice of magnetic shear.
- Understand compatibility between power and particle exhaust methods and good core performance in a compact stellarator.

Together with the other components of the compact stellarator program (Section 1), results from NCSX will permit an assessment of the compact stellarator's attractiveness for a practical fusion system in the next decade. And, while ITER advances the physics and technology of burning plasmas for MFE using the tokamak, research on compact stellarators will aim at the utilization of a continuously burning fusion plasma for power generation. A successful test of compact stellarator advantages in NCSX can provide a basis for proceeding to a next-step compact stellarator experiment operating closer to reactor conditions, including the use of DT fuel. Such an experiment will benefit fully from the progress made in burning plasma R&D through the ITER program. The quasi-axisymmetric design, because of its similarity to the tokamak, is advantageous in facilitating physics knowledge transfer between ITER and compact stellarators.

2. NCSX Physics Design

The NCSX is designed to test a QAS plasma configuration with β = 4% and an aspect ratio $R/\langle a \rangle$ = 4.4. The shape was designed to provide stability and low ripple (good quasi-axisymmetry) at β = 4%. The effective ripple for non-symmetric neoclassical transport is 0.4% at S = 0.5 (where S is the normalized toroidal flux) and about 2% at the edge. An axisymmetric configuration would be unstable without a conducting wall at this _ value, so this design provides a clear test of the stabilizing effect of three-dimensional shaping. External coils provide about three-fourths of the rotational transform (magnetic twist), while the remainder is provided by the self-generated bootstrap current. The plasma has good magnetic surfaces all the way to the edge.

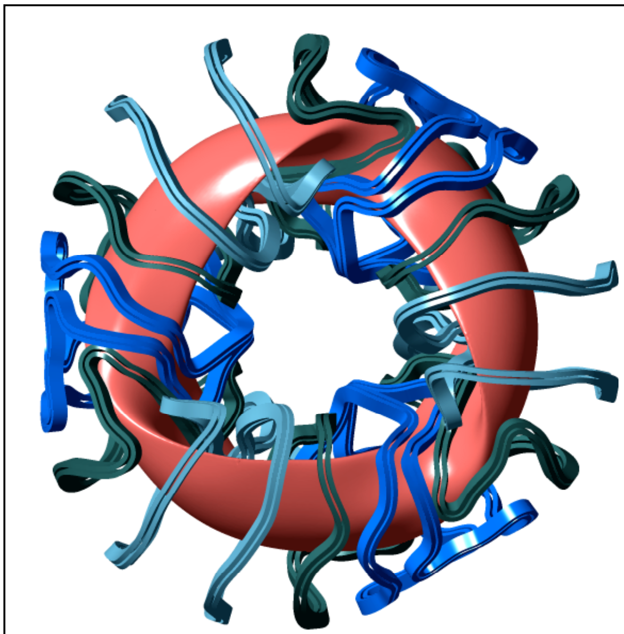


Fig. 1. NCSX Plasma and modular coils.

The NCSX uses modular coils (Fig. 1) to provide the helical magnetic field. The design also includes toroidal field coils, poloidal field coils, and trim coils for equilibrium. flexibility. The coils are designed to produce good magnetic surfaces and to construct the reference plasma. They also provide the flexibility needed to vary the equilibrium parameters, such as the external rotational transform profile, the shape, and the degree of quasi-axisymmetry, that determine plasma physics properties. The coils can support experiments to understand and control stability limits, transport, plasma-wall interactions, and conditions for disruption avoidance. Shear flexibility is illustrated in Fig. 2. It is also possible to vary the rotational transform at fixed shear, to vary the ripple, to lower the ideal instability thresholds to β = 1%, to increase β to 6%, and to support a wide range of pressure and current profile shapes. The coils are compatible with a stable startup pathway from vacuum magnetic surfaces to a target high-beta equilibrium. They are robust so as to accommodate the anticipated range of variation of profile shapes, betas, and currents that will be encountered in startup and physics experiments, compatibly with good quasi-axisymmetry and magnetic surfaces.

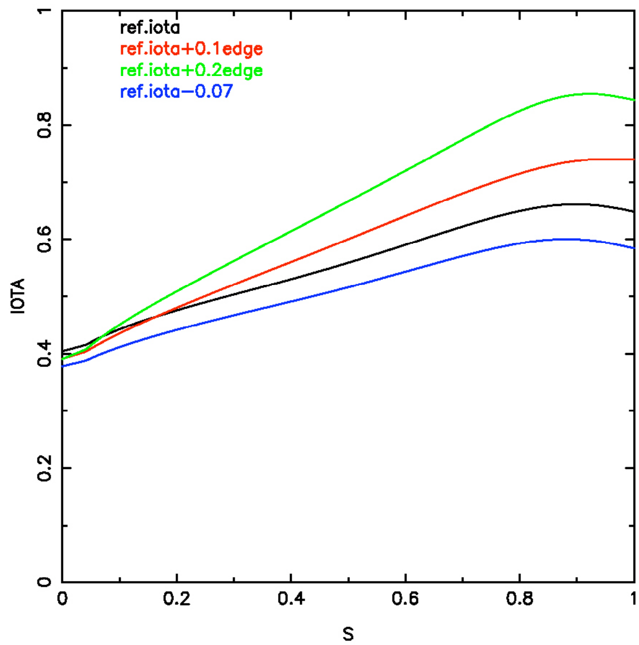


Fig. 2. Range of rotational transform profiles demonstrating NCSX shear flexibility.


Fig. 3. NCSX vacuum vessel with ports.

The NCSX device size (major radius R = 1.4 m), magnetic field range (B = 1.2-2.0 Tesla), pulse length (0.3-1.7 s) and plasma heating power (up to 12 MW) are set to produce the plasma conditions and profiles needed to test critical physics issues over a range of beta and collisionality values. The device will initially be equipped with two of four existing 1.5-MW neutral beam injectors used on earlier tokamak experiments. They will be configured so as to provide control over the rotation profile. Plasmas with β = 4% and low collisionality are predicted to be accessible with the full 6-MW beam system.

The vacuum vessel (Fig. 3) provides access for a large complement of diagnostics, which will be implemented during the research program. Radio frequency waves (up to 6 MW) can be launched from the high-field side to more directly heat electrons than with the neutral beams. Electron cyclotron heating can be readily accommodated through various combinations of NCSX's nearly one hundred ports. Both gas injection and inside-launch pellet injection will be implemented for fueling.

Control of impurities and neutral recycling is the main power and particle handling issue in the design of NCSX. For impurity control, low-Z materials (carbon) are planned for surfaces with intense plasma-wall interactions, though other materials including lithium can also be tested. For neutral control, recycling sources and baffles will be arranged so as to inhibit neutral flow to the main plasma. Motivated in part by recent W7-AS divertor results showing improved edge control and plasma performance, the NCSX is designed so that a pumped slot divertor can be installed.

3. Research Plan

The NCSX research program will proceed in a series of phases, as illustrated in Fig. 4. Acceptance testing of the device will included e-beam mapping of the magnetic surfaces to verify global construction accuracy and production of a 0.5-T stellarator plasma to test integrated system performance. The first phase of the research program, Magnetic Configuration Studies, will use e-beam mapping apparatus to map flux surfaces, measure configuration properties, verify coil flexibility, and test trim coils. The next two phases, 1.5-MW Initial Experiments and 3-MW Heating, will address numerous plasma physics research topics using NBI-heated plasmas. Research goals for the first three phases are summarized as follows:

1. Stellarator Acceptance Testing
   - Verify construction accuracy
   - First Plasma (B = 0.5T stellarator plasma)
2. Magnetic Configuration Studies phase
   - Document vacuum flux surface characteristics
   - Document control of magnetic configuration characteristics using coil current
3. 1.5MW Initial Experiments Phase
   - Explore and establish plasma operating space
   - Characterize low-power confinement, stability, and operating limits
4. 3MW Heating Phase

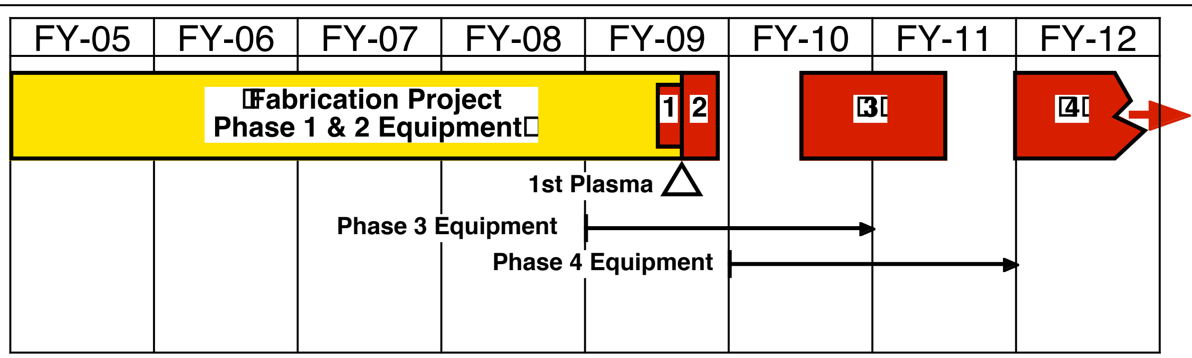


Fig. 4 NCSX Research Program timeline. Numbered research phases are described in text.

- Characterize confinement and stability at moderate power, and their dependence on plasma 3D shape
- Test plasma stability at moderate β, dependence on 3D shape
- Investigate local transport and effects of quasi-symmetry
- Characterize SOL properties for different 3D geometries, prepare for the first divertor design.
- Explore ability to generate transport barriers and enhanced confinement regimes.

Later phases will use increased heating and improved control of the magnetic configuration and boundary conditions to expand the range of accessible plasma conditions. Building on the understanding gained during the earlier phases, the program will explore beta limits, disruption-free operation, enhanced confinement, and long-pulse operation near NCSX design parameters.

4. National Collaboration in NCSX Research

The NCSX research will be carried out by a collaborative national team led by Princeton Plasma Physics Laboratory (PPPL) and Oak Ridge National Laboratory (ORNL) in partnership. A Program Advisory Committee advises on scientific priorities and research issues. Collaborators will lead experiments and hold leadership positions in the project organization. Research forums will be held starting about 2 years before First Plasma to facilitate national team formation. These national meetings will enable potential participants to learn about NCSX research issues, participate in the setting of priorities and research plans, and plan their participation in the program.

5. NCSX Construction

After a series of successful reviews and DOE decisions in 2001-2004, the NCSX project was approved for construction in September, 2004. Contracts for the two largest components, the modular coil winding forms (the structural supports) and the vacuum vessel, were placed in October, 2004, and the parts are being fabricated. A coil manufacturing facility has been established at PPPL, where the coils will be wound. The designs of the conventional coils, machine structures, and cryostat are in the process of being completed in preparation for fabrication. The completed device is illustrated in Fig. 5. Construction will be completed in 2009 with the current funding profile.

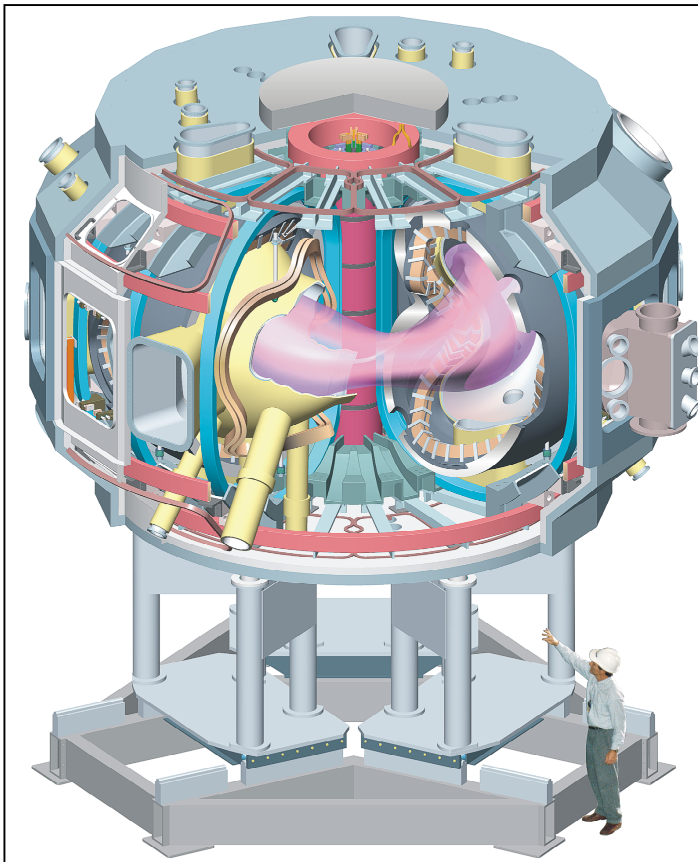


Fig. 5. NCSX stellarator device.

4. The Quasi-Poloidal Stellarator

1. Quasi-Poloidal Symmetry.

The quasi-poloidal stellarator (QPS) is a key part of the integrated national compact stellarator program designed to address important US program issues by taking advantage of QPS's unique features: quasi-poloidal symmetry and the ability to vary key configuration properties over a wide range. QPS is an optimized stellarator-mirror hybrid configuration with quasi-poloidal symmetry (a very small variation of $|B|$ in the poloidal direction, hence a larger "mirror" variation in the toroidal direction). This permits QPS to study reduction in neoclassical transport (low effective field ripple), reduction in anomalous transport (low flow damping for large poloidal flows and resulting ambipolar electric field), equilibrium robustness with strong toroidal/helical coupling (good magnetic surfaces) and healing of magnetic islands, the character of instabilities and $\langle\beta\rangle$ limits up to 5%, and edge topology for divertor design.

Figure 1(left) shows |B| contours| (blue-purple), magnetic field lines (red), and net flow directions (arrows) in flux coordinates on the $(\psi/\psi_{edge})^{1/2} = r/a = 0.2$ magnetic surface; θ is the poloidal angle variable, and ζ is the toroidal angle variable. Since the plasma generated E x B and diamagnetic flows are nearly poloidal, minimal parallel flows (and viscous stress) are required to achieve parallel pressure balance in comparison to configurations such as the tokamak, in which the plasma induced flows are nearly perpendicular to the direction of minimum viscosity and relatively larger parallel flows are required. Figure 1(right) shows the deviation from quasi-poloidal symmetry as a function of magnetic surface radius, indicating the relative insensitivity of the magnetic configuration to the pressure-driven current.

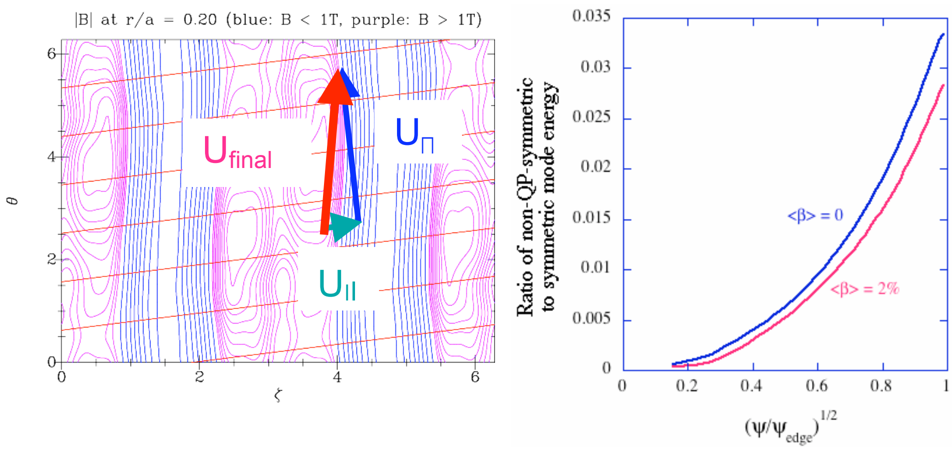


Fig.1 . Quasi-poloidal symmetry characteristics for QPS.

2. Plasma and Device Description.

Figure 2 shows a top view of the modular coils and the last closed magnetic surface for QPS that illustrates the 3-D nature of the low-aspect-ratio QPS plasma. The colors indicate the value of |B|, which varies a factor of 2 on the surface. The approximate poloidal symmetry in flux coordinates shown in Fig. 1 is achieved through a racetrack-shaped magnetic axis and vertically elongated cross-sections in the regions of high toroidal curvature. The main QPS parameters are average major radius $\langle R\rangle$ =

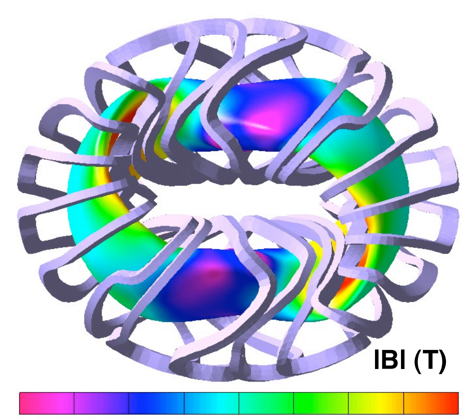


Fig. 2. Top view of QPS plasma and coils with |B| contours

0.95 m and an average minor radius $\langle a \rangle$ = 0.3-0.4 m, for an aspect ratio $\langle R \rangle / \langle a \rangle$ = 2.7;average on-axis magnetic field B = 1 T; an auxiliary toroidal field of 0.15 T; long-pulse heating power P = 3-5 MW, and a plasma current up to 50 kA. Figure 3 shows a cutaway view of the QPS device. The main device components are: (1) 20 non-planar modular coils that create the main magnetic field configuration; (2) an external vacuum vessel; (3) three sets of external circular poloidal field coils for additional plasma shaping and position control; (4) twelve auxiliary TF coils; and (5) central solenoidal coils for driving a plasma current. Parts of the structural shell that connects the individual modular coils and the divertor plates that control particle recycling are also visible in cutaway in Fig. 3. Nine sets of independent coil currents allow varying the quasi-poloidal symmetry by a factor of 9, the flow damping by a factor of 25, neoclassical transport (effective magnetic field ripple) by a factor of 20, amount of stellarator or tokamak shear, and the fraction of trapped particles. In addition, QPS can vary other configuration properties: the fraction of trapped particles, the amount and sign of the magnetic shear, the ambipolar electric field and internal transport barriers, the amount of bootstrap vs. Ohmic current, and the size of magnetic islands and ergodic regions. These features allow QPS to make significant contributions for: (1) advancing toroidal confinement understanding in the areas of 3-D equilibrium and MHD stability, reduced neoclassical and anomalous transport, and a natural divertor for particle and power handling; and (2) concept improvement due to a factor of 2-4 lower aspect ratio than conventional stellarators, which can lead to a smaller reactor embodiment.

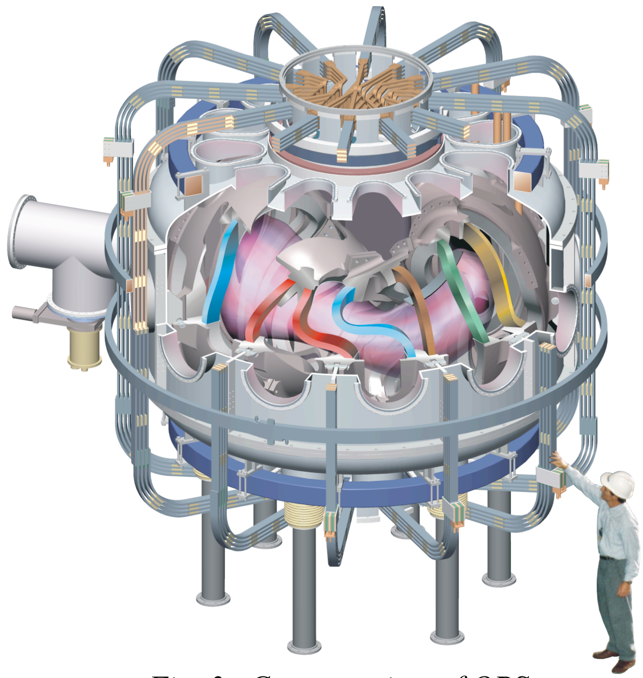


Fig. 3. Cutaway view of QPS

QPS is now in the R&D and prototyping phase at Oak Ridge National Laboratory and the University of Tennessee (UT) in partnership with Princeton Plasma Physics Laboratory. A stainless steel casting of the most complex of the modular coil winding forms (in red) shown in Fig. 3 is being fabricated. Internally cooled cable conductor will be wound on these forms after high-precision machining is completed on the forms. A vacuum-tight stainless steel can will cover the winding pack and the coil pack will be vacuum pressure impregnated with cynate ester resin to complete a modular coil. A coil winding, canning and potting facility is being prepared at UT and small R&D test coils are being wound to test winding, vacuum canning and potting techniques. The experience gained in winding and potting the NCSX coils will be factored into fabricating the QPS coils. A building to house QPS has been completed and a power supply enclosure is under construction. Start of operation is planned for 2010 assuming project funding starts in 2006.

3. QPS Program.

QPS contributes unique information on FESAC's high priority scientific questions.

3.1. How does magnetic field structure impact fusion plasma confinement?

Quasi-poloidal symmetry and the resulting low flow damping in the symmetry direction allow large flows that can shear apart turbulent eddies and reduce anomalous transport. Poloidal flows are the most effective method for this. Corresponding electric fields and their effect on flows can also affect both neoclassical and anomalous transport. The damping can be varied externally.

The large reductions in the effective helical ripple are expected to greatly reduce both neoclassical transport. The 2004 stellarator data base suggests that lower effective ripple may also reduce anomalous transport. Scaling of confinement with configuration parameters (effective ripple, aspect ratio, type and degree of quasi-symmetry, etc.) has not been explored at low aspect ratio or in a quasi-poloidal configuration.

These studies will provide insight for other magnetic configurations. The effective ripple can be varied over a very wide range and might be tied to flow damping physics.

3.2. What limits the maximum pressure that can be achieved in laboratory plasmas?

This requires research on equilibrium limits and the onset and control of magnetic stochasticity. Current data indicates that β in stellarators is not limited by instabilities because quiescent plasmas are routinely observed well above linear stability thresholds. The character of MHD instabilities is different in stellarators than in tokamaks. Ballooning instability occurs simultaneously on a surface in tokamaks, but occurs progressively line-by-line with different growth rates as β increases in stellarators. These differences should provide insight into the nonlinear character of MHD instabilities.

Experiments in the large-aspect-ratio W 7-AS stellarator indicate that achievable β values correlate with the loss of ~35% of the plasma radius due to stochastic regions or magnetic islands. However, magnetic islands can be controlled and minimized both through the self-stabilizing effect of a plasma current, the opposite effect to tearing modes in tokamaks, and by bootstrap- and Ohmic-current tailoring of the q profile. Figure 4 illustrates how a mixture of Ohmic and bootstrap current can be used to avoid low-order rational values of the rotational transform to preserve good magnetic surfaces in QPS.

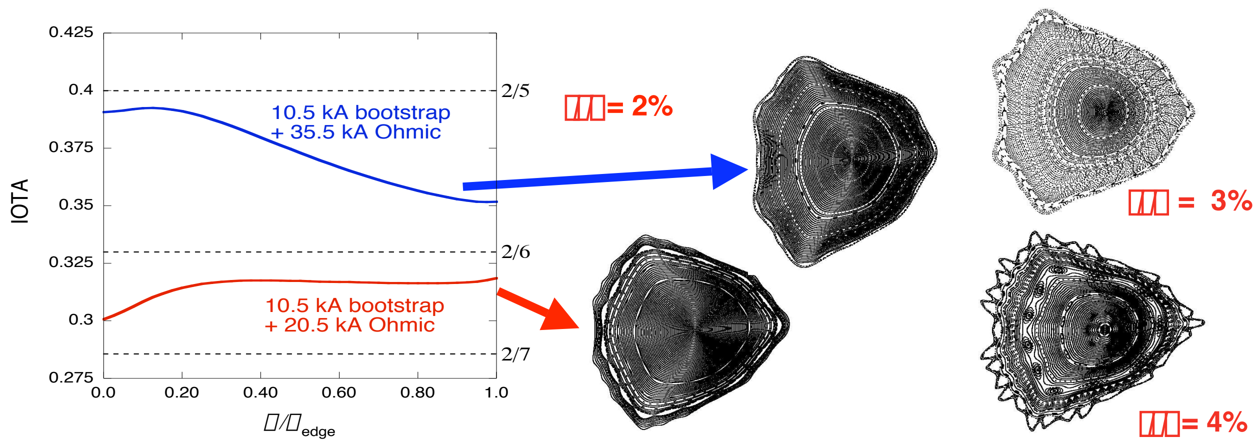


Fig. 4. Control of magnetic surfaces with admixture of Ohmic current in QPS.

3.3. How much external control versus self-organization will a fusion plasma require?

Understanding the use of dominant external control (externally generated confining magnetic fields or flows) involves understanding and controlling pressure-gradient-driven plasma currents (the net bootstrap current and the zero net Pfirsch-Schlüter current) and the flow self-organization described earlier. Currents in external coils can be used to control the magnitude and sign of the bootstrap current and the amount of the Pfirsch-Schlüter current. Figure 5 illustrates control of the poloidal (□) flows in QPS, which are much larger than the toroidal (□) flows.

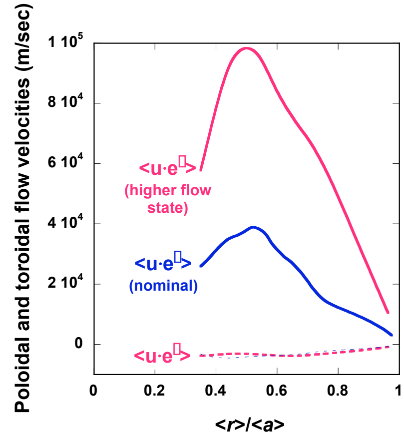


Fig. 5. Flow velocities in QPS.

3.4. How does turbulence cause heat, particles and momentum to escape from plasmas?

The functional form of the normalized probability distribution of edge fluctuations in different toroidal devices is very similar. Does it differ for a quasi-poloidal stellarator, or does the edge layer in all toroidal plasmas belong to a universal class of systems close to a critical point, implying correlations. In the plasma core, the quasi-poloidal structure could influence core turbulence and confinement. The low flow damping due to quasi-poloidal symmetry can allow zonal flow stabilization. The reversed magnetic shear can stabilize trapped particle instabilities, and increase damping of ITG modes. Internal islands can produce E x B shearing, generating transport barriers.

3.5. How are electromagnetic fields and mass flows generated in plasmas?

Quasi-poloidal symmetry provides a unique experimental vehicle to study both plasma generated mass flows and currents. This form of symmetry strongly damps parallel flow components while minimizing damping in the poloidal direction for the naturally driven E x B and diamagnetic flows. This also implies a suppression of the bootstrap current (proportional to the difference between the electron and ion parallel flow components) relative to a tokamak. In a device with perfect quasi-poloidal symmetry, the bootstrap current would be suppressed by a factor of ~ (□N_{fp}). This combination of reduced, but readily measurable, bootstrap currents and enhanced poloidal flows should allow both perpendicular and parallel flow damping effects to be measured and anomalies relative to neoclassical predictions assessed to a degree not possible in toroidally symmetric devices where only the parallel components are readily observable. The flexibility of QPS, both with respect to magnetic structure and plasma parameters, will also allow the effects of collisionality and electric fields on bootstrap currents and poloidal flows to be examined. Data on collisionality and electric field variations can lead to improved methods for separating out the electron and ion contributions to these flows and currents.

3.6. How to interface hot plasma to room temperature surroundings?

3-D shaping flexibility allows different edge strategies: diverted field lines, island divertors, ergodic edges, or combinations. W 7-AS and LHD divertors have successfully demonstrated density and impurity control, including high-□ plasmas. Divertor plates are planned to control recycling and handle the diverted power in QPS. The open geometry of QPS provides flexibility in changing these components. In addition, good or enhanced confinement has been obtained at very high density in stellarators with divertor operation.

The low aspect ratio and quasi-poloidal symmetry of QPS is driving theory development. In 3-D plasma equilibrium, the toroidal and poloidal variations are strongly coupled, which requires keeping more modes and makes convergence more demanding. In MHD stability, interpretation of high-n ballooning stability is different because calculations don't apply to the entire surface as in a tokamak. In transport, it is necessary to compute the full transport matrix and include magnetic islands and 2-D variations within a flux surface.

5. The Helically Symmetric Experiment (HSX)

The Helically Symmetric Experiment (HSX) is the only quasi-symmetric stellarator in the world and will continue to be the only operating advanced stellarator until NCSX is built by 2009 and Wendelstein 7-X is operational in about 2010.

The quasi-helical symmetry in HSX results in predicted neoclassical transport comparable or even lower than a tokamak and nearly two to three orders of magnitude down from conventional stellarators. HSX is a toroidal experiment with an aspect ratio of 8, but with a toroidal curvature (κ) of an aspect ratio 400 conventional device. The spectrum thus possesses a single dominant helical harmonic, with symmetry breaking terms well under 1%, even at the edge. Quasi-helical configurations have been found computationally at aspect ratio 6, and potentially even lower depending upon the degree of quasi-symmetry required.

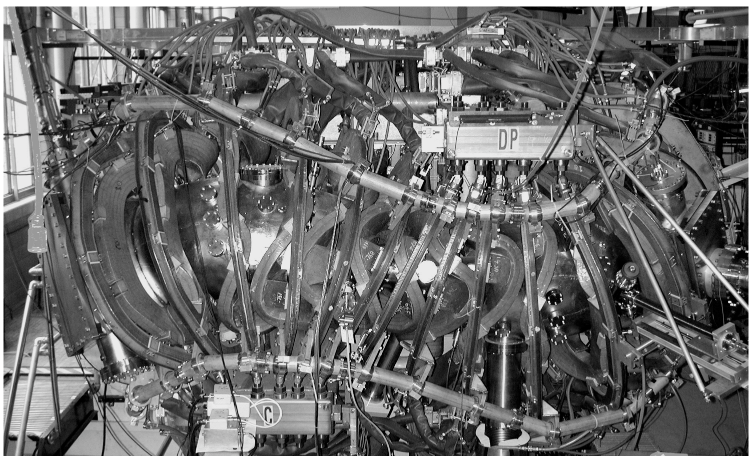
One unique feature of quasi-helical symmetry is the high effective rotational transform given by $|N - \iota|$ which for HSX with N = 4 and $\iota \sim 1$, gives $\iota_{eff} \sim 3$ or q = 1/3. This high effective transform has multiple benefits, which factor into the HSX Program and its relationship to other elements of the Compact Stellarator Proof-of-Principle (PoP) Program:

- The drift of passing particles from a magnetic surface and the banana orbits of trapped particles are very small. Also, the effective ripple, ϵ_{eff}, a measure of the asymmetric ripple-trapped neoclassical transport, is very low in HSX, and approaches that of NCSX, resulting in a minimal number of particles on direct loss orbits. In addition, the symmetric component of the neoclassical transport is greatly reduced because of the quasi-helical symmetry and the high effective transform.
- The Pfirsh-Schlüter and bootstrap currents are reduced in a quasi-helical stellarator compared to a tokamak or quasi-axisymmetric stellarator, resulting in smaller finite beta effects on the magnetic spectrum, surfaces, and equilibrium. Plasma currents are not required to attain the needed experimental configurations in HSX.
- High effective transport and low currents provide a clear separation of confinement and heating issues in HSX.
- According to the latest stellarator database studies, global confinement time in stellarators, which is typically dominated by anomalous transport, scales with transform to the 0.4 power. Also it has been shown that the confinement increases with lower ϵ_{eff} even at high collisionality where asymmetric neoclassical transport is unimportant. Both

results suggest that anomalous transport is improved when particle drifts are more closely aligned with a magnetic surface.
- The somewhat higher aspect ratio in HSX reduces concern of outer flux surface fragility present at low aspect ratio.

The primary goals of HSX are:

- Test reduction of direct loss orbits and neoclassical electron thermal conductivity.
- Demonstrate lower parallel viscous damping of plasma flows.
- Explore possible E_r control through plasma flow and/or ambipolarity constraint.
- Investigate turbulence and anomalous transport in quasi-symmetric configurations without plasma current
- Provide support for research preparations for other elements of the US CS PoP Program



HSX Parameters
R = 1.2 m
<a> = 0.15 m
B = 1.0 T
ι_0 = 1.05, ι_a = 1.12
Number field periods = 4

Electron Cyclotron Heating
28 GHz gyrotron
Power = 200 kW
Upgrade in FY2006 to 400 kW

Results to date have shown that the quasi-helical symmetry does indeed improve single-particle confinement over a non-optimized 3-D configuration, as predicted. This is evidenced by:

- A factor of two higher growth rate in plasma formation during ECH breakdown compared to a configuration in which the symmetry is broken.
- Significantly improved absorption of microwaves at low plasma density.
- Direct observation of deeply-trapped electron losses when the symmetry is broken.
- Increased flux and energy of hard x-rays; a factor of three longer decay time of the flux after the ECH turn-off compared to when the symmetry is broken.

In addition, we have made several other significant contributions to understanding the impact of quasi-symmetry on magnetic topology, confinement and stability:

- By analyzing the orbits of passing particles, we were able to map out the dominant magnetic field spectral components, demonstrate the absence of toroidal curvature and show that the high effective transform reduced the deviation of the orbit from a flux surface.
- For the first time in any stellarator, we have demonstrated that quasi-symmetry does indeed lead to reduced plasma flow damping. The externally driven flow in the QHS configuration rises and damps more slowly than in a configuration with the symmetry broken and attains approximately twice the flow velocity for the same drive.

- We measured two time scales for the flow evolution. This is a crosscutting contribution to the understanding of flows in toroidal plasmas, not just stellarators.
- We have made extensive H_{α} measurements and 3-D DEGAS modeling of the neutral density in HSX. With these results we were able to take into account the damping due to ion-neutral collisions as well as determine particle source rates.
- We were able to demonstrate that the reduced neoclassical transport in the quasi-helical configuration led to a decrease in the thermodiffusive part of the particle flux, resulting in peaked density profiles regardless of the temperature gradient. In contrast, the density profile with the symmetry broken is flatter or slightly hollow.
- Finally, we have evidence of an interesting and unforeseen MHD instability, possibly a Global Alfven Eigenmode, driven by fast electrons whose existence may be related to the good confinement of trapped particles with quasi-helical symmetry.

HSX provides a bridge between the 2-D world of the tokamak and fully 3-D systems. The HSX program meshes with many of the overriding themes under development by the Priorities Panel. By comparing data from the QHS and Mirror configurations we learn how the magnetic field structure affects plasma confinement. Our work on turbulence and momentum damping will help us understand how turbulence causes heat, particles and momentum to escape from the plasma. We will build on our present work on understanding how plasma flows are generated and damped in HSX. The work on plasma breakdown and future plans to increase the heating power will help us to understand how electromagnetic waves interact with plasma. The work presently being done on fast electron impact on MHD phenomena and GAE modes increases our knowledge of how high energy particles interact with plasma.

6. The Compact Toroidal Hybrid Experiment

The smallest of the experiments in the US Compact Stellarator program, the Compact Toroidal Hybrid (CTH) at Auburn University plays an essential role in the investigations of equilibrium and MHD stability outlined in Section 2. While integrated tests of optimized transport, stability, and equilibrium control will be performed on NCSX and QPS, it is very important over the next several years to develop a better understanding of a number of stellarator-specific issues that will figure strongly in the operation and performance of these innovative experiments, as well as finite-β stellarators in the international program. Studies of equilibrium and stability will be performed on CTH, which is now beginning operation after achieving its first plasma on Feb. 22, 2005. The research focuses on three related topics:

1. Maintenance of good flux surfaces in low-aspect ratio, stellarator plasmas with bootstrap and/or Ohmic current that significantly modify the vacuum magnetic equilibrium. The fragility of the outer vacuum flux surfaces and the reduction of magnetic islands that may be present at low-order rational surfaces will be addressed with a flexible set of auxiliary correction coils. In light of finite-β work to be carried out in the future on NCSX and QPS, it will be particularly important to develop methods to identify, and control, static magnetic islands in current-carrying plasmas.

2. Initial test and experimental validation of the new V3FIT 3-D equilibrium reconstruction code now under development by a multi-institutional collaboration (GA/ORNL/Auburn/PPPL). Within the US stellarator program, there is ~~thus~~ an urgent need for the capability to reconstruct the experimental magnetic equilibrium (rotational transform profile, location of boundary, Shafranov shift, local curvature, etc.) from both internal and external magnetic and kinetic measurements. The V3FIT code that is expected to provide this capability will be benchmarked on the CTH experiment in which Ohmic toroidal currents will modify the vacuum equilibrium. This comparison of experiment and computational theory is expected to be the main accomplishment of the near-term CTH program, and is intended to be one of the major deliverables of the project to the Compact Stellarator program.

3. Investigation of current-driven instabilities and disruption avoidance in compact stellarators. With the magnetic equilibrium parameters reasonably well-diagnosed, CTH will quantitatively examine the behavior of kink, vertical, and tearing instabilities in low-□ compact stellarator plasmas. Of particular interest is the behavior of current-driven disruptions in helical devices, and understanding of how the control of external helical field of the stellarator may be used to reduce or eliminate them as severe problems.

The CTH device, as shown in a recent photograph as it neared completion, is a continuous coil torsatron with five field periods with a poloidal periodicity of l = 2. It also incorporates auxiliary toroidal field coils and additional poloidal field coils for varying the vacuum rotational transform and its shear. The physical parameters are listed in the adjoining table. Target plasmas will be generated by electron-cyclotron heating, and plasma currents will be produced by an independent Ohmic coil set. ICRF heating will be implemented at a later date.

<u>CTH parameters</u>

M= 5, l = 2

R_o	= 0.75 m
a_{vessel}	= 0.29 m
a_{plasma}	≤ 0.18 m
B_o	≤ 0.7 T
I_p	≤ 50 kA
P_{input}	= 20 kW ECRH 100 kW OH 200 kW ICRF
n_e (est.)	= 0.2 - 1 x 10^{19} m^{-3}
T_e (est.)	≤ 400 eV
□	≤ 0.5%
Vacuum □	0.2 – 0.45
Discharge duration	0.4 s; 0.1 s w/ OH

7. Stellarator Theory

The goals of stellarator theory are to improve the understanding of plasma confinement in three-dimensional (3-D) systems by comparing theoretical models to experiment results and to factor this understanding into the design and optimization of the next generation of configurations. With the absence of a geometric symmetry direction, a number of issues such as nested equilibrium flux surfaces and collisionless particle confinement are not guaranteed for arbitrary 3-D configurations, but must be verified theoretically. The multiple coil sets of flexible stellarator experiments also provide opportunities for control over neoclassical transport and MHD stability that are not present in axisymmetric systems. These factors have motivated a close connection of theory both in the stellarator design process and in the interpretation of experimental results. The tools developed for the analysis of stellarators may also be applied to 3-D effects (field ripple, MHD perturbations, resonant field errors, etc.) in tokamaks.

Understanding of stellarator magnetic equilibria forms the basis for all subsequent calculations of transport and stability. Present experiments indicate that the maximum pressure in stellarators may be limited by changes in flux-surface topology. For these reasons, accurate and rapid 3-D equilibrium calculations are essential for future progress in stellarator modeling. Equilibrium reconstruction forms the primary interface between experiment and theory. As increasingly comprehensive plasma parameter and magnetic probe measurements are coupled to the reconstruction codes currently under development, it should become possible to more accurately infer the magnetic structure within the plasma. Equilibrium models that can treat islands and stochastic regions need to improve the modeling of the plasma kinetic interaction with the local magnetic field topology, including two-fluid and flow effects. Finally, improved equilibrium models are critical for understanding island suppression and in specifying field error tolerances.

Plasma transport reduction has been a primary focus of recent stellarator design and optimization efforts. Neoclassical losses have been significantly reduced from the levels of classical stellarators. However, in order to realize the maximum benefit of these optimizations, methods for reducing anomalous losses in the next generation of compact stellarators must be developed. This requires improved understanding from transport models that treat the couplings between plasma flows, cross-field turbulent transport and ambipolar electric field states (including bifurcations), and external momentum sources. Although the initial neoclassical components of such models are already in existence, they will need to be coupled together into comprehensive 1.5-D (1-D transport + 2-D averaged equilibria) transport codes, including better estimates for anomalous effects, and eventually into 2.5-D (2-D transport + 3-D equilibria with magnetic islands) codes. In addition to fluid transport modeling, more basic research will be needed in the area of microturbulence-driven transport in stellarators. Topics such as ion/electron temperature-gradient-driven modes, trapped particle modes, shear flow turbulence suppression mechanisms, and their configurational dependencies need to be addressed.

Plasma heating physics in stellarators is closely coupled with transport analysis. Neutral beam heating models have been based in the past on local solutions of the Fokker-Planck equation and more recently on Monte Carlo models. The Monte Carlo models will need to achieve improved computational performance for the rapid turn-around required to routinely model experiments. Also, they will need to include multiple background species and more comprehensive diagnostics such as heat and momentum deposition rates, and beam-driven currents. RF heating full-wave and geometric-optics models for 3-D systems have been developed, but require further improvements in computational performance in order to be coupled to transport codes. Also, such models do not currently maintain consistency between the plasma dielectric coefficient and the tail populations that are created by heating. This will require coupling the full-wave solver with a calculation of the absorption coefficient that takes into account non-Maxwellian and non-local transport effects.

In present day stellarator experiments, unlike tokamaks, MHD instabilities have not played an important role in constraining the operational space. In order to answer the question of whether that will remain true in the new generation of compact stellarators, the non-linear saturation of a range of MHD instabilities, including pressure-driven and current-driven instabilities, will need to be modeled in greater depth. Also, operating stellarators have observed a variety of energetic particle de-stabilized Alfvén, fishbone and EPM instabilities. These modes are of interest both as a result of their negative impact on confinement and heating efficiencies as well as their possible diagnostic use. In order to address these MHD issues, improved closure relations and hybrid models are needed.

A final area that needs ongoing development is optimization of stellarator configurations. The physics information developed in the new compact stellarator experiments will need to be factored into future optimization models to determine the next steps leading up to burning plasma and demo-level configurations.

8. INTERNATIONAL COLLABORATION

Collaboration with the larger international stellarator program on selected topics is an important element of the U.S. compact stellarator program because it provides information on stellarator physics that is not otherwise available in the U.S. program. The flagships of the international stellarator program focusing on the currentless large-aspect-ratio stellarator approach are billion-dollar-class facilities now operating in Japan (LHD) and under construction in Germany (W 7-X, 2010). These experiments are designed to demonstrate steady-state disruption-free stellarator operation and a level of performance at ~30 MW heating power that allows extrapolation to devices capable of burning plasma operation. These large facilities are supplemented by medium-size ($30-100 million scale) medium-power (few MW) experiments in Japan (CHS), Germany (W 7-AS, no longer operating), and Spain (TJ-II).

Experimental Collaborations. The wide range of stellarator configurations accessible on LHD, W 7-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 Q-A and Q-P 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. compact stellarator program, which focuses on lower aspect ratio (A = 2.7-4.4) and larger bootstrap current fraction.

<u>LHD (Japan)</u>. The order of magnitude increases in plasma volume, heating power, and pulse length of LHD over that of other stellarators allows studies of size scaling and stellarator physics

at more reactor-relevant parameters ($\langle\beta\rangle \geq 4\%$, $T_i \sim 14$ keV and $T_e \sim 10$ keV, τ_E hundreds of ms, pulse length > 1/2 hour with input energy = 1.3 GJ, etc.). U.S. collaborative studies have focused on: spatial and energy resolved energetic neutral measurements, neutral beam injection, ICRF heating, transport and equilibrium modeling, and magnetic diagnostics.

W7-AS (Germany). Confinement improvement and a magnetic-island-based divertor system have been studied in W 7-AS in a magnetic configuration that is complementary to that of LHD. U.S. collaborations have focused on ICRF heating, pellet injection, transport modeling, Alfven modes, and the equilibrium and stability properties of high-beta plasmas.

CHS (Japan). CHS allows study of transport and beta limits at plasma aspect ratios as low as 5. Two heavy ion beam probes are used to study the effect of electric fields and zonal flows. U.S. collaborations have focused on ICRF heating, local island divertor, fast ion losses, 3-D equilibrium calculations, and configuration optimization.

TJ-II (Spain). TJ-II allows study of beta limits and transport in a stellarator with a large helical axis excursion, an ingredient in U.S. stellarator configuration optimization. U.S. collaborations have focused on neutral beam injection, pellet injection, EBW heating, edge turbulence imaging, and code development.

Theory Collaborations. Collaboration with institutions in Germany, Switzerland, Japan, Spain, Austria, Australia, the Ukraine and Russia on stellarator theory and computational tools development benefits U.S. efforts in support of compact stellarator development. Areas of collaboration include: MHD equilibrium; Mercier, ballooning and kink stability; Alfven modes; microstability; bootstrap current; transport; turbulence; optimization techniques; coil design; effects of magnetic islands; divertor modeling; and reactor studies.

9. FUSION POWER PLANT STUDIES

Integrated physics and engineering systems studies are used to assess the potential of different confinement concepts as attractive reactor candidates and to highlight high-leverage issues that need to be studied in the U.S. fusion program. These assessment capabilities have been developed in the ARIES tokamak reactor studies and in an earlier (1994) U.S. Stellarator Power Plant Study (SPPS). Stellarators have important advantages as reactors -- inherent steady-state capability with no disruptions, fully ignited operation with no need for current drive or power input to the plasma, and no need for rotation drive or feedback control of instabilities. Their disadvantage had been their relatively large size. The SPPS reactor configuration allowed reducing the reactor size from $\langle R \rangle = 22$ m (for the W 7-X-based HSR) to $\langle R \rangle = 14$ m for the SPPS reactor. The possibility of further large reductions in reactor size is one motivation for the U.S. compact stellarator program.

The SPPS configuration extrapolated to a reactor power plant that was economically competitive with the second-stability ARIES-IV tokamak reactor because of its low recirculating power, assuming that stellarators have the same unit costs for components with complicated geometry as tokamaks and that tokamaks have the same high availability as stellarators. The SPPS reactor used a four-field-period $\langle R \rangle/\langle a \rangle = 8$ configuration with physics properties similar to the large-aspect-ratio W 7-X modular stellarator experiment now under construction. 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, vacuum vessel, coils, structure and other components that scale approximately with the plasma surface area).

The ongoing ARIES Compact Stellarator reactor study is examining coil configurations based on the three-field-period NCSX configuration, but modified to increase the spacing between the plasma and the coils, and a low-aspect-ratio two-field-period plasma configuration. Two types of blanket and shield configurations and both port-maintenance and field-period-maintenance approaches are being examined. An initial case has <R> = 7.2 m, a factor of 2 lower than the SPPS case. This example is comparable to tokamaks in compactness, but without the disadvantages associated with a large, driven plasma current. Newer plasma and coil configurations are being pursued that have low alpha-particle losses.

There are a number of important reactor-relevant questions that the compact stellarator program can answer. (1) What combination of bootstrap and externally generated rotational transform is optimal? (2) How strong does 3-D plasma shaping need to be to stabilize limiting instabilities? (3) How low must the field ripple be in a quasi-symmetric system to reduce helical ripple transport, alpha-particle losses, and flow damping to desired levels? (4) What topology should be used for power and particle exhaust via a divertor? (5) How low an aspect ratio is optimal? (6) How high does the <□> need to be?

Perhaps one of the most important outcomes of the compact stellarator program for reactor development is an improved energy confinement scaling, particularly clarification of the role of the effective helical field ripple. Understanding the effects of density/temperature profile shapes on equilibrium and MHD stability beta limits will help reactor configuration optimization, as will study of energetic particle loss mechanisms and methods of loss mitigation. Better understanding of 3-D edge physics and power/recycling control will be essential for divertor development, without which a reactor cannot function.