Scientific and Programmatic Review of NCSX

Stellarator Community Input on Charge Questions

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1. Critical Scientific Issues for the U.S. Compact Stellarator Program

The purpose of stellarator research is to gain understanding of non-axisymmetric plasma shaping of magnetically confined plasmas and its potential benefits for fusion energy. Stellarators are inherently steady state because they use non-axisymmetric magnetic fields from coils to create the basic conditions for toroidal confinement- nested magnetic surfaces and rotational transform- without requiring a large driven current. Experiments have demonstrated that non-axisymmetric shaping allows: (1) robust plasma stability, eliminating disruptions and resistive wall modes (RWMs), (2) control and maintenance of the q profile without current drive, and (3) a broad range of operating densities, much above the Greenwald density limit. The ITER team has recognized the importance of resolving these issues going from ITER to DEMO class machines [1].

A scalar pressure plasma equilibrium is defined by the shape of the outermost surface and the profiles of pressure and safety factor. The safety factor q is the number of toroidal transits a magnetic field line makes of the torus while making one poloidal transit; the inverse is the rotational transform ι=1/q. Self-consistent transport processes in a steady-state high-gain fusion plasma dominantly determine the pressure and current profiles, but plasma control can be accomplished through control of the plasma shape.

The importance of axisymmetric shaping to the performance of tokamaks is well established, as is the need for disruption mitigation techniques, active RWM feedback control, active NTM feedback control, ELM control, and current-profile control. Non-axisymmetric shaping provides steady state solutions to these issues as well as allowing higher density operation in a DEMO class machine than a conventional tokamak. The highest possible density for a tokamak, the Greenwald limit, when included in a self-consistent DEMO design by the ITER group yields an on-axis ion temperature of about 45 keV, in part to improve current drive efficiency. However, this (1) makes the power handling in the divertor more difficult than at higher density/lower temperature, and (2) increases the alpha particle pressure, which enhances the drive for Alvénic instabilities.

Non-axisymmetric shaping requires the explicit imposition of certain constraints: (1) good magnetic surfaces, (2) particle trajectories that remain close to the magnetic surfaces, (3) practical coils and structures and, for compact reactor size, (4) a low aspect ratio. The imposition of these constraints while optimizing performance through plasma shaping defines the U.S. compact stellarator program.

The constraint that particle trajectories remain close to the magnetic surfaces can be achieved if the magnitude of the magnetic field, $|B|$, has one of three symmetries: (1) toroidal, (2) poloidal, or (3) helical. The deviation of particle drift-orbit trajectories is determined by the magnitude of the magnetic field; the geometric shape of magnetic surfaces has no direct relevance. A quasi-symmetry means $|B|$ accurately approximates a symmetry in flux coordinates, but the shape of the magnetic surfaces does not. The constraint on trajectory confinement is important not only for particle confinement but also because it leads to low flow damping in the symmetry direction. Self-driven plasma flows, termed neoclassical and zonal flows, can break up turbulent eddies and greatly reduce the level of transport due to microturbulence.

Quasi-axisymmetry (QA) means $|B|$ has toroidal symmetry. Quasi-axisymmetry can be imposed at any level of 3D shaping of the magnetic surfaces from zero, the conventional tokamak, to sufficiently large that the rotational transform is predominately produced by the shaping rather than by the plasma current. In a typical NCSX high-beta equilibrium, 75% of the transform is from shaping. Quasi-axisymmetric shaping can be viewed as a 3D extension of axisymmetric shaping to enhance the performance of tokamaks. NCSX is the only quasi-axisymmetric stellarator in the world program. Results from NCSX could lead to a reactor based on an NCSX-like configuration, as exemplified by ARIES-CS, or could emerge along the continuum between NCSX and the advanced tokamak.

Quasi-poloidal symmetry (QP) means $|B|$ approximates symmetry in the poloidal direction. In exact quasi-poloidal symmetry, the Pfirsch-Schlüter and the bootstrap currents would be zero, which implies the shape of the magnetic surfaces and the safety factor would be determined purely by the externally produced magnetic field and have no dependence on the plasma pressure. In addition, the particle drift trajectories would not leave the magnetic surfaces. For quasi-poloidal symmetry the mathematically required deviation from perfect quasi-symmetry is greater than the deviations for quasi-axisymmetry and for quasi-helical symmetry. Nevertheless, toroidal equilibria can be found that approximate attractive features of quasi-poloidal symmetry by minimizing the deviation of the action contours $J=\oint v_{||}dl$ from the magnetic surfaces to ensure good particle confinement. Two distinct approximations to quasi-poloidal symmetry are being explored in the world program: the large aspect ratio (R/a = 10) W7-X, which is under construction in Germany, and the compact ($R/a \approx 2.5$) QPS, which is in prototype development at Oak Ridge. Different physics constraints were imposed while optimizing the transport. The Oak Ridge group finds much larger flows and flow shear in QPS for suppression of turbulence than in W7-X

Quasi-helical symmetry (QH) means $|B|$ has symmetry in the helical direction. Within the magnetic surface $|B|$ is a function of $\theta_h = \theta + N\varphi$ alone, where N is the number of toroidal periods of the stellarator. Theoretically a stellarator with helically symmetric magnetic surfaces has stable equilibria at high beta with excellent particle trajectory confinement. Perfect helical symmetry cannot exist at finite aspect ratio, but quasi-helically symmetric stellarators do. The HSX stellarator at the University of Wisconsin is the only quasi-helical stellarator in the world program. Its research program is to study the importance of quasi-symmetry on plasma confinement.

A substantial world stellarator program exists, including large experiments on a scale comparable to the largest tokamaks. The largest stellarator in the world, the Japanese superconducting LHD, was designed before solutions for particle trajectory confinement were fully understood. Nonetheless, LHD shows much improved plasma performance when operated in a configuration that approaches quasi-poloidal symmetry. Both LHD and W7-X are focused on stellarator reactor development paths parallel to and distinct from tokamaks.

The unique contribution of the U.S. stellarator program is the understanding of quasi-symmetric shaping and how it can be used to control plasma performance and solve problems for magnetic fusion. The U.S. program emphasizes compact design, with aspect ratios comparable to those of tokamaks, leading to reactors comparable in size and cost to tokamak reactors. Stellarator research addresses the same critical science and technology issues as are being addressed by the U.S. program as a whole (e.g., FESAC Priorities Panel report, April, 2005). Moreover, stellarator research will permit a credible assessment to be made of the level or the type of non-axisymmetric shaping that should be used in fusion plasmas.

Transport: Stellarator experiments have found a confinement scaling that is similar to that in tokamaks. Interestingly, microturbulent transport in stellarators is known to be smaller when the magnetic configuration is consistent with particle drift trajectories that stay close to the magnetic surfaces; reduced damping of zonal flows is thought to be a major factor. Sufficiently rapid transport of impurities relative to the plasma energy is critical to any fusion concept. The radial electric fields in stellarators can be manipulated, the so-called ion and electron roots, but a predictive capability for impurity transport needs to be established.

Beta limits: Stellarators have been operated at average values of beta above 4%, and higher beta limits are predicted for optimized stellarator configurations. As the empirical beta limit is approached, a gradual degradation of confinement is found.

Disruptions: Experimentally, disruptions are not observed in stellarator experiments, except in very specific situations such as a global reconnection with very low shear. Even when the empirical beta limit is above the ideal MHD stability threshold, no disruptions are observed. Experimental and theoretical work remains to accurately determine the level and type of non-axisymmetric shaping that is required to eliminate disruptions and resistive wall modes. Studies with Ohmic current on CLEO indicated that disruptions could be stabilized if 15% of the rotational transform was produced by the 3D shaping.

Density limits: Stellarators have operated at densities well above the Greenwald limit. The radiative cooling of the plasma exceeding the input power generally sets the density limit. However, further research is required to have a predictive capability of the density limit.

Energetic particle behavior: As is well known from tokamak and stellarator experiments, energetic particle confinement is extremely sensitive to lost particle trajectories in the equilibrium magnetic fields. This is mitigated by the higher density and lower temperature operation that is typical of stellarator reactor designs. The electrical currents that are associated with the parallel viscous force, which gives neo-classical transport, tend to symmetrize the magnetic field and enhance the confinement of particle trajectories. These issues require further interaction of experiments and theory. The higher density limit available in stellarators strongly reduces fast-ion beta, reducing the drive for Alfvénic eigenmodes and energetic particle instabilities, and it increases the mode damping rates. Calculations for ARIES-CS parameters, for example, show that TAE modes may not be unstable, in contrast to their robust instability in ARIES-AT.

Plasma edge and divertors: Although stellarator divertors have been designed, the experimental tests are limited. Higher density eases divertor power-loading issues, and edge stochasticity, which arises naturally in stellarators, is being prominently considered in the tokamak program to spread exhaust power and control ELMs. In cases where ELMs are observed in stellarators, they are generally very small.

Response to magnetic perturbations: An issue in common with tokamaks is the effect of magnetic perturbations, generally called field errors, on plasma performance. External trim coils have been successful in mitigating the effects of field errors on low-order resonant surfaces. The PIES code has been used to assess the expected quality of magnetic surfaces in stellarators. Perturbed equilibrium codes have recently been used with considerable success on tokamaks to understand their behavior in the presence of error fields and similar techniques exist for stellarators. A new equilibrium code is under development at ORNL to study equilibria with islands.

Simplified coils and structures: Coils and structures for non-axisymmetric plasmas are more complex than for axisymmetric plasmas. Issues that have arisen in the construction of NCSX and W7-X are in large part addressed by the experience gained while building these devices. Nevertheless, research on simplified coil and construction solutions is required to have a realistic assessment of the implications of non-axisymmetric shaping on device complexity and cost.

- Required coil tolerances: It has been customary to construct stellarators with much tighter tolerances than tokamaks. This drives the cost at every stage of construction, from design to final assembly and test, more so than geometrical complexity. To mitigate the risk of disruptions due to mode locking, modern tokamaks such as ITER now require assembly tolerances similar to (or tighter than) stellarators. In stellarators, field errors degrade confinement, but do not cause mode-locking or disruptions. The use in stellarators of error field correction coils to ease tolerance requirements needs further experimental and theoretical investigation. In stellarators, one can directly measure the quality of the vacuum magnetic surfaces, to confirm the quality of the field provided by the coils.
- Simpler coils: Improved methods of determining what coil features are required to obtain certain physics objectives need to be developed. If objectives are reduced, e.g. ripple or stability margins, then the shaping requirements are relaxed and the design of the coils is simplified. Modular coils have been a satisfactory solution for many stellarator experiment and reactor designs, but alternative topologies and construction methods that could ease geometrical and assembly complexity need to be investigated.

The U.S. stellarator community addresses these issues through a coherent program consisting of several elements, of which NCSX is the largest (currently about three-fourths of the effort). All program elements are needed and should be strengthened to capitalize on the opportunities afforded by compact stellarators in a complete way.

- NCSX: A high-beta, very low ripple QA stellarator test at $R/a = 4.4$, currently under construction by PPPL and ORNL. Motivated by its direct linkage with the tokamak program and ITER, the NCSX is being conducted on a scale sufficient to study the integrated issues of confinement, performance and disruptivity at high beta, edge stability, and Alfvénic stability.
- HSX: A QH stellarator with low ripple and high effective transform (low q), and currently the only operating quasi-symmetric device in the world. It is providing fundamental tests of quasi-symmetry, and has already shown reduction in parallel viscous damping, thermo-diffusion of particles and electron thermal diffusivity with quasi-symmetry. (Operating at the University of Wisconsin)
- QPS: Will test a high-beta very low aspect ratio ($R/a \approx 2.5$) QP configuration with physics connections to W7-X. QPS differs in physics optimization from W7-X (with 4 times higher

R/a than QPS) by having low average grad-B drifts due to alignment of **B** and grad-B rather than low average grad B as in W7-X. (QPS is in R&D and prototype fabrication at ORNL.)

- CTH: Though not itself quasi-symmetric, CTH addresses issues of passive disruption avoidance in compact current-carrying plasmas in which 3-D plasma shaping and vacuum transform are applied to passively control MHD instabilities. Also, CTH is validating new methods of magnetic equilibrium reconstruction in 3-D plasmas. (Operating at Auburn University)
- CNT: Studies confinement of non-neutral plasma in a non-symmetric stellarator constructed from four circular coils. (Operating at Columbia University)
- Collaboration on LHD and W-7X: Taking advantage of superconducting coils, high-power heating and high plasma parameters, comparative studies of confinement and its dependence on configuration parameters (rotational transform, helical ripple, aspect ratio, etc.), maintenance of and limiting phenomena in high-beta plasmas, long-pulse to steady-state operation, and divertors.
- Reactor studies. Compact stellarator reactor studies are conducted through the ARIES program. They provide assessments of compact stellarator reactor issues and clarify compact stellarator research priorities from a reactor perspective. The ARIES-CS compact stellarator reactor study, recently completed, projects that the NCSX configuration extrapolates to a power plant with costs comparable to tokamaks.
- Theory and simulation: Addresses fundamental 3D physics, analytical tools, development of predictive understanding of stellarators, connections between stellarators and 2D systems, configuration optimization and improvement. This element provides the scientific integration for the program, using the knowledge gained from a variety of experiments together with constraints of physics and mathematics to establish a predictive understanding of the effects of plasma shaping. The US stellarator theory program has developed and applied world-standard tools for 3-D MHD equilibrium and stability, transport, plasma edge and divertor development, concept improvement, and reactor optimization.

Quasi-symmetry in general combines the excellent orbit confinement of tokamaks with the robust stability, q profile control, and a broad range of operating densities of stellarators. The non-axisymmetric shaping of stellarators solves issues that must be addressed before going to DEMO. Quasi-axisymmetry provides continuity between tokamaks and stellarators and, therefore, gives the clearest path from ITER to DEMO that takes advantage of the benefits of non-axisymmetric shaping.

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[1] Mukhovatov et al., Nucl. Fusion 47, S404 (2007)

2. Role of NCSX in the International Context

NCSX is a 'Proof of Principal' experiment, designed to explore and understand 3D QA shaping to address the plasma physics challenges of enhanced performance steady-state operation without disruptions [1]. NCSX uses 3D shaping to enhance MHD stability without requiring external current drive or feedback systems, producing high-beta plasmas that have the potential to be free of disruptions. Due to the quasi-axisymmetry of the shaping, with low remnant effective ripple, the plasma transport properties in NCSX are predicted to be similar to equivalent tokamaks. Thus, NCSX seeks to combine the established features of stellarators (robust stability without feedback control, no disruptions, steady-state without external current-drive, higher density limit) with the excellent transport and compact size of tokamaks.

NCSX is the only experiment in the world exploring quasi-axisymmetric 3D shaping. NCSX has been designed with a wide range of flexibility to vary key configuration parameters (e.g. iota and shear), to understand the underlying physics and to determine how much 3D QA shaping is required. Due to the shared symmetry and transport physics, the combined understanding of NCSX and tokamaks including ITER, will enable the fusion program to evaluate the benefits and applicability of 3D QA shaping for DEMO. The importance of 3D QA shaping for fusion energy was recently validated by the ARIES-CS study [2], which projected that a reactor based on a NCSX-like plasma could be more economical than ARIES-RS, and comparable to ARIES-AT if it used the same blanket technology.

2.1 The NCSX mission

The NCSX mission is to provide the experimental basis needed to understand and assess the use of 3D QA shaping for controlling high-beta toroidal plasma confinement, and in particular for providing solutions to the physics issues that stand between ITER and DEMO. NCSX will investigate the full breadth of issues for 3D QA shaped plasmas, including beta limits and limiting mechanisms, disruptivity, turbulence and confinement, rotation and damping, magnetic surface structure, MHD and fast-ion instabilities, and divertor solutions. This research will be used to test, improve, and ultimately verify theoretical and computational models of plasma confinement with 3D QA shaping, and to develop a common understanding with tokamaks, incorporating the rho*-scaling and burning-plasma understanding from ITER. This understanding will enable accurate assessment of the use of 3D QA shaping for DEMO, including the shaping needed for high β disruption-free operation compatible with steady state sustainment of the magnetic configuration. The improved understanding obtained from NCSX may allow future 3D QA designs to be significantly simplified, by reducing the constraints imposed on the NCSX design (see below) [ref Ku].

The breadth of this mission and the study of high-β plasmas requires substantial diagnostics, plasma heating, and power handling capabilities. High heating-power is also required to study plasma wall interactions and edge solutions. Moderate plasma size (similar to PLT or D-III), in terms of poloidal flux or ion drift-orbit width divided by minor radius, is needed to obtain collisionless plasmas for the studies of high-β stability and anomalous transport, and to confine co- and counter-injected neutral-beam ions. The breadth of questions proposed and these required capabilities typify a 'Proof of Principle' experiment, as defined in the FESAC Report on

Alternate Concepts, DOE/ER-0690. FESAC recommended in 2001 that it was appropriate to address these questions and to include a Proof-of-Principle compact stellarator experiment in the U.S. fusion research portfolio. The choice of quasi-axisymmetry was based, in part, on the close connection with the large tokamak physics data base.

2.2 The NCSX design.

The NCSX design builds upon the experimental and theoretical understanding of plasma transport and stability from both tokamaks and stellarators, producing a unique set of capabilities. The design was extensively analyzed to ensure that the goals were robustly achieved, were accessible, and that the design had flexibility to explore parametric variations in the theoretically key parameters. The NCSX physics design characteristics include (simultaneously):

- Very good quasi-axisymmetry (low residual ripple), such that the ripple-driven neoclassical thermal transport is negligible [3] and fast ions are confined. NCSX has the lowest residual ripple of any stellarator proposed or constructed. This is predicted to give tokamak-like zonal flows for turbulence suppression [4] and low rotation damping.
- Passive stability at beta = 4.1% to all MHD instabilities that limit beta in tokamaks, including kink, ballooning, vertical, Mercier, and neoclassical-tearing instabilities, without the need for feedback stabilization, driven rotation, or nearby conducting walls [5]. The MHD stability threshold can be raised above beta=6.5% by adjusting coil currents. While stellarators can in some circumstances operate above their projected beta limits, in the case of high-n instabilities this may be an artifact of the high rho* of current experiments at high beta.
- Passive disruption stability, in that the magnetic equilibrium is maintained even with a total loss of beta or plasma current.
- 'Reversed shear', i.e. q decreasing or rotational transform increasing across the plasma. This enhances MHD stability, suppresses islands and tearing modes, stabilizes trapped particle instabilities, and reduces turbulent growth rates [6].The ITG/TEM linear growth rate decreases with increasing beta. In tokamaks, this produces self-stabilization of turbulence and neoclassical ion, particle, and momentum transport.
- 75% of the edge magnetic rotational transform is produced by the coils, 25% by the bootstrap current at 4% beta, with no externally driven plasma current (thus compatible with steady state). This is a substantially lower bootstrap current than advanced tokamaks, significantly reducing the non-linear influence of beta on the equilibrium.
- Very good flux surface quality for high beta [7],vacuum, and intermediate beta, with less than 3% of the flux lost to islands at beta=4%.
- Expanded boundary divertor-like edge geometry, independent of the edge rotational transform, due to the strong plasma shaping [8]. NCSX will install a baffled divertor for controlling power and particle exhaust and will study the divertor and SOL characteristics. The full vacuum system is bakeable to 350C, consistent with modern tokamak practice.
- Aspect ratio of 4.4, chosen to be similar to the ARIES-RS tokamak design, and significantly lower than W-7X and LHD. Comparisons across the transport optimized stellarators will provide information to understand the effect of aspect ratio on stellarator confinement.

Beta = 4.1% was chosen as the design goal because it is substantially above the predicted no-wall stability limit for advanced tokamaks with high bootstrap-current fractions (such as ARIES-RS and ARIES-AT). A stable access path from startup to high-beta has been simulated [9], and the current profile was calculated to be stable to tearing modes [10].

NCSX has been designed as a flexible experiment, allowing a wide variation of the 3D plasma shape and magnetic configuration characteristics by varying the nine external coil currents [11]. This will enable controlled experiments challenging and validating our understanding. For example,

- The degree of QAS or effective ripple can be continuously varied at least a factor of 30.
- The vacuum magnetic rotational transform (from the coils) can be varied by at least a factor of 4. The vacuum shear can be varied to both positive and negative (stellarator and tokamak-like). At high beta, the global shear can be doubled or eliminated.
- The MHD kink stability threshold can be reduced down to beta ~1% at either constant rotational transform or constant edge shear.

This flexibility is obtained using auxiliary coil sets that can be eliminated to simplify future energy systems.

The magnetic field strength can be varied up to 2 T, giving access to higher temperature, lower collisionality plasmas. With 6MW of heating, ion temperatures above 4 keV and ν_i*~ 0.04 are predicted. This enables experimental study of the confinement scaling with normalized gyroradius (rho*) separated from the collisionality (nu*) and beta scalings, for comparison with theoretical transport models and projections to future devices. This should also allow NCSX to overlap the dimensionless parameter regimes of middle-scale tokamaks. This capability is important for testing the connection between NCSX and tokamak transport understanding.

NCSX can accommodate up to 12 MW of external heating (NBI, ECH, ICRF) [12] and a comprehensive set of plasma diagnostics. The early experiments will emphasize neutral beam heating, to provide access to high beta. The heating systems will be refurbished and reused from previous experiments. The ECH sources are available via collaborations with IPP/Greifswald and GA. Several groups, including NIFS (Japan), have expressed interest in diagnostic collaborations on NCSX. The diagnostics and heating systems will be installed throughout the evolution of the research program.

2.3 The NCSX Timeline

The planned timeline has NCSX operating in FY2012, 2013, and 2015 with the following capabilities:
FY2012: first plasma and vacuum magnetic configuration characterization
FY2013: Initial plasma experiments, with 3MW of NBI, B>1.2T, initial diagnostics: magnetics, core profiles, scrape-off layer and edge impurities. Based on tokamak scalings, this power level should give access to H-mode.
FY2015: High beta experiments, with 6MW of heating power, B=2T, enhanced resolution diagnostics, rotational transform and fluctuation measurements.

NCSX and NSTX will operate in alternating years until NSTX shuts down, with NCSX conducting experiments in odd-numbered years and upgrading diagnostics and capabilities in even-numbered years.

In FY2013, NCSX will investigate
The effect of QAS and residual ripple on plasma confinement and flow damping.
Resilience to disruptions from MHD instabilities and density limits.
Initial comparisons between the measured and calculated linear MHD stability threshold.

In FY2015, NCSX will investigate
β-limits and limiting mechanisms, and the operating area safe against disruptions.
Local transport properties at fields up to 2T and powers up to 6MW, including impurity transport.
Fast ion transport due to effective ripple. Alfvén-mode stability and fast ion losses.
Divertor effectiveness, scrape off characteristics.

Enough information should be available after the FY2015 campaign to start evaluating the implication of 3D QA shaping for future devices, and to begin the design of follow-on experiments. It is expected that NCSX will continue developing and refining the scientific understanding of QA shaping for a number of years.

2.4 Summary
NCSX offers exciting opportunities for unique fusion science, investigating the effect of 3D QA shaping on MHD stability, confinement, and disruptions. It will test the theoretical predictions that 3D shaping can stabilize high-β plasmas without the need for nearby conducting structures, instability feedback systems, or external current drive. It will demonstrate whether tokamak-like transport can be obtained in a QA stellarator, and develop a common understanding of transport with ITER and tokamaks. It will study the physics of fast particle instabilities and high-power plasma-wall interactions in this configuration. The understanding developed by NCSX may allow a simpler design in future devices, using less 3D QA shaping along the continuum from the advanced tokamak to NCSX. It can make fusion energy more attractive by producing stable high-β, high density plasmas with good confinement that are compatible with steady state operation, without disruptions or the need for current drive. This would greatly simplify the first wall design in fusion power systems. As seen from ITER studies and extrapolations to DEMO, disruption immunity may be required for a viable breeding blanket and first wall. Current drive in advanced tokamaks can very costly in recirculating power. The very high density achievable in stellarators may also provide needed solutions for alpha stability and power handling. The mission of NCSX is to investigate these issues to assess the potential impact of QA 3D shaping and compact stellarators on both fusion science and fusion energy. There is no other experiment in the world able to investigate these issues or carry out this mission.

As discussed in the next chapter, the U.S. could address important and interesting compact stellarator issues at moderate plasma parameters in the absence of NCSX, but in that case:
- The U.S. would not have a compact stellarator experiment capable of addressing the issues of confinement, stability, energetic particle physics and divertors in an integrated manner at high beta, low rho* and low collisionality.
- The opportunity to build on the synergy between tokamaks and QA stellarators in all of these areas of fusion science would be lost.

The ability to influence the world program and the design of a U.S. DEMO based on results from the Compact Stellarator program would be greatly reduced or lost.

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3. Options to address the key issues of quasi-symmetric and compact stellarators in the absence of NCSX

The US compact stellarator program has been designed as an integrated program to assess the attractiveness of compact stellarators and to advance understanding of non-axisymmetric shaping and the resulting physics. Compact stellarators are not addressed in stellarator work abroad, so there is an opportunity to maintain the leading US role in this area in both experiment and theory.

Approximately 75% of the U.S. stellarator effort is presently focused on the construction of NCSX, so the loss of NCSX would fundamentally change the character of the U.S. stellarator program. The program would lose its integrated PoP facility and the clear connection to tokamaks through quasi-axisymmetry. Enhancement of the other U.S. stellarator program elements could address important compact stellarator issues, albeit in a less complete and less integrated way. The different US components add to the knowledge base from different perspectives.

Theory and computation are critical innovating and integrating elements of the US compact stellarator program. The US stellarator theory program has developed and applied world-class tools for 3-D MHD equilibrium and stability, transport, concept improvement, and reactor optimization. These capabilities will be applied to planning, interpreting, and extrapolating experiments in the US and overseas, and in developing improvements to the compact stellarator concept. U.S. leadership in stellarator theory and modeling should be strengthened, particularly in the high-priority areas of understanding the high-beta and high-density capabilities of stellarators, confinement, zonal flows and flow damping, and simplification of compact stellarator coils for future devices. Investment in theory has particular leverage because it enables US researchers to add considerable value to collaboration on diverse experiments around the world during the lead time required for construction of new experimental facilities in the US.

HSX with $R/a = 8$ is the only operating quasi-symmetric stellarator in the world program and provides a fundamental test of quasi-symmetry and very low ripple. Its high degree of QH symmetry, high effective transform and lack of toroidal curvature lead to unique properties for physics studies. A key goal of the HSX program is understanding the effects of QH symmetry, high effective transform and low effective ripple on both anomalous and neoclassical transport. HSX has shown that the deviation of circulating particles from QH-symmetric flux surfaces are very small [1] and is now investigating the helical Pfirsch-Schlüter current and reversed (compared to a tokamak) bootstrap current that are properties of the QH design. HSX has already demonstrated some key properties common to all quasi-symmetric approaches: reduced parallel viscous damping [2], the reduction of the neoclassical particle flux due to temperature gradients [3] and the decrease of the neoclassical electron thermal conductivity in the low collisionality regime [4].

QPS [5] is based on a very-low-aspect-ratio ($R/a \approx 2.5$) QP symmetric configuration that approximates linear linked mirrors without end losses in a toroidal geometry. It would allow large poloidal flows and flow shear for effective suppression of turbulence, low effective ripple for reduction of neoclassical losses, and connection to the high-aspect-ratio W7-X. Its nine independent coil sets would allow varying key physics features by >10x: quasi-poloidal symmetry, poloidal flow damping, and neoclassical transport; stellarator vs tokamak shear; and the fraction of trapped particles. QPS also extends the stellarator data base on confinement and stability to very low aspect ratios, similar to what the ST does for tokamaks.

CTH does not have quasi-symmetric geometry, but directly addresses the issues of passive disruption avoidance in compact ($R/a \geq 4$) current-carrying helical plasmas in which 3-D plasma shaping and vacuum transform are applied to passively control MHD instabilities. Also, CTH is validating new methods of magnetic equilibrium reconstruction (developed in collaboration with the theory program) that can be applied to all 3-D plasmas and is studying the detailed control of magnetic islands. In doing so, it is applying new procedures to determine the most accurate model of as-built coil geometry from field-mapping. [6] These topics are central to the compact stellarator mission of demonstrating robust stability and equilibrium control of plasmas.

International collaboration allows access to stellarator experiments (particularly LHD and W7-X) with capabilities beyond the scope of the US program. Although these devices are not compact stellarators (R/a = 6-7 in LHD and 10.4 in W7-X), they can be used to obtain important information on plasma behavior at high parameters (density, ion and electron temperatures, β), energetic-ion stability and transport, steady-state operation and β maintenance at high power, and 3-D power and particle exhaust methods (3-D divertors). In addition, they are additional sources of information on the effect of lower effective ripple on neoclassical and anomalous transport, and on density and beta limits and mechanisms.

The key physics issues for a reduced US compact stellarator program are similar to those that would have been addressed in an integrated fashion in NCSX, and they would be addressed at significantly lower plasma parameters, since the Kadomtsev similarity parameter Ba^(5/4) is significantly higher in NCSX than in other U.S. experiments.

1. *Effects of strongly reduced effective ripple on energy confinement*. Various experiments address this issue for different symmetries: HSX (QH), QPS (low-R/a QP), W7-X (high-R/a QP) and LHD. Deviations of particle trajectories from magnetic surfaces are greatly reduced in LHD by shifting the magnetic axis inward, which produces a large improvement in microturbulent transport. With a good understanding of the neoclassical level, the anomalous contribution can be examined in HSX by varying the effective ripple using an auxiliary coil set with matched temperature and density profiles. QPS can provide similar information at higher plasma parameters by varying the effective ripple over a wide range. These results can be used with the helical ripple scans from LHD and results from W 7-X to test the expectation that confinement improves with decreasing effective ripple. Ambipolar electric fields are important in determining both neoclassical and anomalous transport in 3-D systems, and will be studied using various diagnostics in these experiments.

2. *Pressure limits and limiting mechanisms at moderate* $\langle\beta\rangle$. LHD and W 7-X will address this issue at high aspect ratio and QPS at low aspect ratio. Beta limits in stellarators appear to be set by degradation in confinement rather than by disruptive plasma instabilities—a topic that requires further exploration. QPS is predicted to be stable to drift wave turbulence over a range of temperature and density gradients without additional momentum input, which should reduce anomalous transport even in absence of flow shearing. QPS also has a large (and variable) frac-

tion of trapped particles in regions of low/favorable field line curvature for studies of suppressing trapped-particle instabilities. It is the only device with a region of stability for the CTEM.

3. *Disruption stabilization and avoidance*. CTH is the only experiment in the existing world program that can study elimination of disruptions by large (factor ~ 10) variations in the ratio of the externally produced rotational transform to that from the plasma current. Stabilization and mitigation studies can span the range from near-tokamak geometry to helically-dominated configurations. Other experiments (HSX, W 7-X, QPS, LHD) avoid disruptions through no or low plasma (bootstrap) current and tailoring the rotational transform profile.

4. *Reduction of turbulent transport by flows*. HSX can explore the underlying mechanism of configuration-dependent changes in turbulence by examining zonal flow dynamics and equilibrium flows and radial electric fields, coupled with fluctuation measurements and microstability calculations. QPS would exploit its low R/a, low transform and QP symmetry to obtain large flows and flow shearing to reduce anomalous transport and increase stability. QPS supplements W 7-X (with 4 times QPS's R/a) by having low average grad-B drifts due to alignment of **B** and grad-B rather than low average grad B as in W 7-X. In the absence of external momentum input, QPS has larger poloidal flow shear than other toroidal devices (up to an order of magnitude in some regimes), and reduced growth rates for trapped particle and ITG modes. The maximum poloidal flow shearing rates within the flux surfaces in QPS also significantly exceed those of other configurations and can reach levels (0.1 to 0.5 of the inverse Alfvén time) that could influence MHD stability thresholds (ballooning, resistive tearing, etc.). Theoretical and computational studies of these effects in quasi-symmetric configurations is now stimulating experiments on LHD that are looking for evidence of flows associated with improved confinement regimes obtained in very high density plasmas obtained with pellet injection. [7]

5. *Stabilization of equilibrium islands and tearing modes*. An important issue for stellarators, as well as tokamaks, is their response to small magnetic perturbations. LHD experiments [8] have examined these effects in low-order islands near the plasma periphery, but these results need to be extended to higher-order internal islands in different configurations. CTH studies will address the consequences and need for control of static magnetic islands in helical plasmas with significant toroidal current. At higher β, QPS and W7-X could investigate finite pressure modification of islands. Induced currents in QPS could be tailored to study as well the passive stabilization of neoclassical tearing modes. The topic of special interest here is the shielding of islands when neoclassical-tearing-mode effects are stabilizing rather than destabilizing.

6. *Improved energetic-ion stability and confinement*. LHD and W 7-X are best suited to provide information on this topic with high-power neutral beam heating.

Opportunities exist to strengthen the compact stellarator program, to partially offset the impact of the loss of NCSX:

Theory. Significant near-term opportunities for advancing toroidal confinement physics exist in the following areas: (1) 3-D MHD equilibrium modeling and reconstruction (fast codes incorporating plasma current and islands that can be used for optimization studies), 3-D MHD stability (nonlinear ideal and resistive codes), turbulence (effects on momentum transport, gyrokinetic codes to study micro-turbulence), thermal transport (drift kinetic calculation at lower collisionality with more dimensions), energetic particle physics (Alfven and resonant wave-particle modes), and plasma edge and divertor physics.

ECH heating has put *HSX* plasmas into the electron root (calculated) where the large electric field minimizes differences between symmetric and non-symmetric operation; E_r will be measured with CHERS and a novel HIBP to measure potential fluctuations for anomalous transport measurements in the plasma core. Two HIBPs (or HIBP and coordinated BES diagnostic) would allow measurements of zonal flow formation as a function of quasisymmetry, as well as electron versus ion root formation. An additional gyrotron for EBW studies would help expand the density and temperature range in which HSX operates. Because of the short connection lengths in HSX it is relatively easy to heat ions to the low collisionality regime. An ICRF program would allow studies of low collisionality ion confinement, and testing of beta limits at modest values in a device where the only currents that flow are from equilibrium currents.

QPS would be the only compact quasi-symmetric stellarator in the worldwide fusion effort in the absence of NCSX, but it is still in the R&D and prototype fabrication stage; actual construction of QPS has not yet been authorized. The project is fabricating the largest and most difficult of the modular coils as a prototype and is working toward the firm cost and schedule baseline (CD-2) needed for project approval.

To broaden its fundamental role of understanding disruption avoidance in driven helical systems, *CTH* would expand its scope to extend its range of vacuum to total rotational transform to better connect with tokamak configurations. Furthermore, it would attempt to obtain higher plasma pressures for its MHD and equilibrium studies. CTH will continue to develop its diagnostics for measuring the internal rotational transform profile.

The U.S. would seek opportunities to study 3D physics issues of importance to compact stellarators through expanded *international collaborations*, particularly with LHD and W7-X.

Quasi-axisymmetry is sufficiently important that it would need to be pursued even in a reduced compact stellarator program. Studies would be undertaken to determine the best path forward in that case.

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