Chapter 1 -- Motivation and Goals

1.1 Introduction

Magnetic plasma confinement in toroidal geometries is a proven approach for advancing the science of fusion energy and is promising as a basis for future fusion reactors. Most of the research to date has concentrated on two types of configuration, tokamaks and (to a lesser extent) stellarators. Tokamaks, and other axisymmetric configurations, produce the confining poloidal magnetic field by toroidal currents in the plasma, which are typically generated inductively. Tokamaks have demonstrated excellent short-pulse plasma performance in "compact" geometries, with aspect ratios (ratio of the plasma major radius to the average minor radius) usually less than 4. Stellarators use three-dimensional magnetic fields generated by coils to produce some or all of the confining poloidal magnetic field. Stellarators have demonstrated levels of performance approaching those of tokamaks, but generally at aspect ratios in the range of 6-12.

Since stellarators can produce all components of the magnetic field directly from external coils, they are intrinsically well suited for steady state operation, and do not require external current drive systems. In addition, stellarators typically,do not experience disruptive terminations (disruptions) of the plasma. For example, both the W7-A [1] and Cleo [2] experiments were able to eliminate disruptions at the density limit and when passing through edge q=2 (rotational transform $\iota = 0.5$) by the addition of small amounts of externally (coil) generated rotational transform to plasmas with substantial parallel plasma currents. Recent experiments on W7-AS [3] have generated disruptions at edge $\iota \sim 0.5$ (q~2) when the plasma current profile is analyzed to be unstable to global tearing modes (extending from center to edge). This instability is well understood in both stellarators and tokamaks, and is easily avoided either by design or care in experiment operation. Extending the study of stellarator stability and disruption avoidance to high-β and low aspect ratio awaits new experiments.

As in tokamaks, the physics properties of a 3D stellarator are determined by the shape of the MHD equilibrium. Three-dimensional equilibria offer many more degrees of freedom than are available for axisymmetric configurations, and this additional shaping flexibility can be used to tailor the equilibrium to obtain desired physics properties. This was first systematically exploited in the development of the "Advanced Stellarator" (AS) concept [4], in which a stellarator configuration was numerically optimized to realize good equilibrium, stability, and transport properties, using theoretical/numerical models. The AS optimization approach produced the designs for the Wendelstein 7-AS (operating since 1988) and Wendelstein 7-X [5] (under construction) experiments at the Max-Planck-Institüt für Plasmaphysik in Germany.

Historically, the major challenge for stellarators has been to provide acceptable drift-orbit confinement, allowing adequate fast-ion confinement and low neoclassical transport losses. This is due to the fully three-dimensional shape, which has no ignorable coordinates and thus no conserved canonical momenta for drift orbits. Thus, the radial excursion of drift orbits is not necessarily bounded, as it is in axisymmetric systems. In addition, a general 3-D magnetic field will have a strong modulation in the magnitude of B, |B|, in every direction, producing strong plasma flow damping in all directions. With the availability of numerical optimization, two strategies have been developed to provide adequate drift-orbit and neoclassical confinement. The first is 'non-symmetric drift-orbit omnigeneity', whereby the magnetic field structure is optimized to approximately align drift-orbits with flux surfaces by counteracting the toroidal drift-terms with the helical drift-terms. This is the strategy developed for the design of the superconducting Wendelstein 7-X under construction in Germany, with aspect ratio 10.6, where the orbit confinement was specifically optimized to minimize the bootstrap current. This strategy, partially optimized, also underlies the design of Wendelstein 7-AS, the new 'shifted-in' configuration in LHD[19], and the new experiment Heliotron-J [6].

The second strategy for 3D drift-orbit confinement is called 'quasi-symmetry', which is based upon work by Boozer [7] showing that drift-orbit topology and neoclassical transport depends only on the variation of |B| within a flux surface, not on the dependence of the vector components of B. This was used by J. Nuehrenberg [8,9] and P. Garabedian [10] to develop stellarators that, while three-dimensional in euclidian space, have a direction (either helical or toroidal) of approximate symmetry of |B| in (Boozer) flux coordinates. Quasi-symmetric configurations have drift-orbits similar to equivalent symmetric configurations, and thus similar neoclassical transport. Rotation in the quasi-symmetric direction is also undamped, as in a symmetric configuration. The first experimental test of quasi-symmetry is the Helically Symmetric eXperiment (HSX) now beginning operation at the University of Wisconsin.

In parallel, there has been a tremendous advance in the understanding of tokamak experiments and the ability to manipulate tokamak plasmas. There has been a general confirmation of ideal MHD equilibrium and stability theory (excepting Mercier stability) and neoclassical transport theory. Methods for stabilizing and manipulating turbulent transport (particularly for ions) have been developed, allowing the elimination of anomalous ion-thermal and particle transport, and reduction of anomalous electron thermal transport. There is a general understanding of the importance of flow-shear stabilization as a mechanism for stabilizing ion turbulence. In addition, there are a number of theoretical predictions that undamped turbulence-generated flows (zonal flows) are significant in saturating turbulent transport at the levels observed.

1.2 The Compact Stellarator Opportunity

The Compact Stellarator Opportunity is to build upon the advances in understanding of both stellarators and tokamaks and to combine the best features of both. We wish to use the stellarator's externally generated helical field and 3D shaping to enhance the MHD stability without requiring external current drive or feedback systems, designing plasmas that have the potential to be free of disruptions. From tokamaks, we wish to use the excellent confinement, ability to stabilize and manipulate turbulent transport, and lower aspect ratio (compared to classical stellarators) to reduce development costs and system size. We also wish to make use of some bootstrap current (driven internally by the pressure gradient) to relieve the coils of having to produce all the the magnetic rotational transform.

In detail, the opportunity is to use the three-dimensional shaping flexibility of stellarators to passively stabilize the modes that limit the β and pulse-length of toroidal plasmas (particularly the external kink and neoclassical tearing modes), thereby expanding the safe operating β range. Quasi-symmetric design provides good orbit confinement and allows undamped flows to saturate and stabilize turbulence. In our studies, we have found that quasi-axisymmetry is best suited for higher β-limits and lower aspect ratio.

1.3 NCSX Mission, Goals, and Unique Contributions to Fusion Science

The mission of the NCSX research is to investigate the effects of three-dimensional plasma shaping, of internally- and externally-generated sources of rotational transform, and of quasi-axisymmetry on the stability and confinement of toroidal plasmas. In particular,

- What are the beta limits and limiting mechanisms in a low aspect-ratio stellarator? Can pulse-length-limiting instabilities, such as external kinks and neoclassical tearing modes, be stabilized by external transform and 3D shaping?
- How do externally-generated transform and 3D shaping affect disruptions and their occurrence?
- Can the collisionless orbit losses typically associated with 3D fields be reduced by designing the magnetic field to be quasi-axisymmetric? Is flow damping reduced? Is the resulting transport and confinement similar to actually axisymmetric systems? How does the transport scale in a compact stellarator?
- Do anomalous transport control and reduction mechanisms that work in tokamaks transfer to quasi-axisymmetric stellarators? Do zonal flows saturate turbulent transport in a quasi-axisymmetric stellarator at levels similar to tokamaks?
- How do the Alfvenic-eigenmode spectrum and stability of a quasi-axisymmetric stellarator differ from those of a tokamak or a non-symmetric stellarator?

- How do stellarator edge-field characteristics such as islands and stochasticity affect the boundary plasma and plasma-material interactions? Are 3D methods for controlling particle and power exhaust compatible with good core confinement.

In order to carry out this research program, the NCSX plasma and facility has been designed with the following goals

- Passive stability to the external kink, neoclassical-tearing, ballooning, and vertical instabilities to $\beta \geq 4\%$, without the need for nearby conducting walls or feedback systems.
- Good quasi-axisymmetry to provide closed drift-orbits and low flow-damping, sufficient to allow balanced neutral-beam injection (for control of parallel flows and beam-driven currents).
- $\beta \sim 4\%$ stable equilibria consistent with steady state without the need for external current drive.
- Aspect ratio ~ 4
- Good flux surface quality at high beta, with no more than 10% of the toroidal flux in islands or stochastic regions.
- Experimental flexibility to vary the plasma shape, external transform, pressure, and current to carry out the research mission.

$\beta \sim 4\%$ was chosen as a goal to provide an unambiguous test of 3D shaping stabilization of the external kink-instability. While the tokamak (A>2) and spherical-torus (A<2) have achieved very high $\beta_T \sim 12\%$ and ~ 40% (respectively) with large amounts of external current drive (including inductive), the β-limit compatible with steady state operation and low external current drive is much lower. Advanced-tokamak stability studies have found that the ideal β -limit is ~ 2-3% [11,12,13,14,15] with external current drive supplying < 30% of the total magnetic transform and without wall stabilization, even for strong shaping (elongation up to 2, triangularity up to 0.8). The advanced tokamak program is working to raise this limit by a combination of conducting walls and active feedback on the resistive-wall instability. For NCSX, achieving $\beta \geq 4\%$ without significant current drive or wall stabilization would demonstrate that 3D shaping is effective at raising the sustainable β-limit. Aspect ratio ~ 4 was chosen to be substantially below existing designs for stellarators with optimized confinement and stability, yet not so low that the other goals cannot be attained.

Since the issues addressed by the mission questions (high-beta stability consistent with steady state, disruptions, transport, and boundary conditions) are critical for magnetic confinement concept improvement, the answers will have far-reaching scientific benefits. Because of its hybrid character, 3D shaping, quasi-axisymmetric design, and flexibility, the NCSX is uniquely capable of providing the answers.

Thus, NCSX will provide a data base for accomplishing the 10-year milestone established by the Fusion Energy Sciences Advisory Committee in 1999: "Determine attractiveness of a Compact Stellarator by assessing resistance to disruption at high beta without instability feedback control or significant current drive, assessing confinement at high temperature, and investigating 3-D divertor operation." This database will provide the basis for designing follow-on experiments, and determining the value of the compact stellarator opportunity.

The breadth of the questions proposed for study by NCSX, and particularly the study of high-β plasmas, requires substantial diagnostics, plasma heating, and power handling capabilities. Significant heating power is required for high-β studies since the predicted β scales as $P^{0.5}$ for most empirical confinement scalings. 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 moderately collisionless plasmas for the studies of high-β stability and anomalous transport. 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. It is appropriate to address these questions and propose a Proof of Principle experiment, as NCSX builds upon the existing world stellarator program (which is already at the Performance Extension stage), the world tokamak program which has developed the physics of axisymmetric magnetic configurations, and upon HSX [16] which will experimentally test the principles of quasi-symmetry for the first time, at a Concept Exploration level. In addition,

The methods used to numerically design the NCSX plasma and coils have built upon established techniques developed to design W7-X [5] and used to design HSX [16] and previous design studies [9,10]. These techniques have been validated by the verification of fields and initial studies of HSX [17]. The methods have been extended to lower aspect ratio, to include the effects of the bootstrap current, and to address MHD stability in order to address the particular goals of NCSX. This has required significant enhancement of many of the numerical physics models, as discussed in Chapter 2. The codes used to evaluate the physics properties of NCSX and alternative configurations have been extensively benchmarked against both experiments and other codes, as discussed throughout this document.

1.4 Relationship to the World Stellarator Program

There is a strong world-wide program of stellarator research, due in part to their suitability for steady-state operation and stability against disruptions. Existing experiments range from the university scale up to the Performance Extension scale. This broad program has produced a physics and engineering knowledge base and experience that we have been fortunate to be able to build upon in our design studies for NCSX.

The Japanese LHD and German Wendelstein programs feature large experiments with superconducting magnets that will advance stellarator core transport and stability physics, boundary physics, and long-pulse sustainment. However, the LHD and W7-X devices have plasma aspect ratios of 6 and 10.6, respectively. W7-X was explicitly optimized to suppress the bootstrap current while LHD is expected to have relatively low bootstrap currents. Neither will test the effects of quasi-symmetry on the confinement physics, compatibility with the bootstrap current providing a significant fraction of the rotational transform, or reductions in the aspect ratio of stellarators down to values (~ 4) approaching those of tokamaks. These topics will be studied on NCSX as the core of its mission .

The world's largest existing stellarator, Japan's Large Helical Device (LHD), has already demonstrated significant confinement enhancement (up to 2.4 times the ISS95 [18] empirical stellarator confinement-time scaling) and β = 2.4% [19] and recently 3.5% [20] (heating-power-limited) in configurations with aspect ratio around 6. This β value is above the predicted ideal stability limit, yet no limiting instabilities are observed. Confinement times up to 0.3 sec and mid-keV peak temperatures have been obtained. With megawatt-levels of plasma heating, the LHD has operated for pulse lengths greater than 1 minute (recently > 2 minutes [20]).

The operating Wendelstein 7-AS experiment [21] is a partially optimized precursor to Wendelstein 7-X. It has demonstrated the effectiveness of the numerical optimizations in its design, and the benefit of using 3D shaping to control physics characteristics. It has observed enhanced confinement regimes and confinement up to ~2.5 times the ISS95 global scaling. In addition, there are a number of other international stellarator experiments exploring the physics of 3D confinement in a number of configurations at a variety of scales, including CHS[22], TJ-II[23], H-1[24], and Heliotron-J.

The NCSX is the centerpiece of a proposed U.S. compact stellarator proof-of-principle program. The U.S. program will complement the world stellarator program, producing results that are important for fusion science and that will not be obtained otherwise. The national compact stellarator program also couples to and builds upon the existing or already-approved concept-exploration experiments at the University of Wisconsin (HSX) and Auburn University (CTH). The HSX will investigate transport reduction by quasi-helical (QH) symmetry, while the CTH will investigate low-beta stability issues in current-carrying stellarators.

As a complementary part of the proposed U.S. compact stellarator program, a concept-exploration experiment, the QOS, is also being proposed by Oak Ridge National Laboratory. The QOS experiment will investigate the confinement physics of quasi-poloidal (QP) symmetry to reduce neoclassical losses. Since QP symmetry implies a large toroidal viscosity, the QOS experiment will investigate the role of poloidal flows in confinement enhancement, while NCSX with its low toroidal flow damping will focus on tokamak-like flow-shear stabilization. The QOS will push down to very low aspect ratio (<3), where equilibrium quality is a key issue, NCSX will push up in β to explore stability and disruptions. The scientific contributions and design characteristics of the NCSX and QOS will be complementary, in order to broaden the compact stellarator knowledge base.

A low-aspect-ratio quasi-axisymmetric experiment, CHS-qa, is being designed by a group at Japan's National Institute for Fusion Science (NIFS), and indicates the world-wide interest in compact stellarator concepts. The CHS-qa has not been formally proposed. The CHS-qa configuration design effort, so far, has evaluated plasma configurations assuming there is no net plasma parallel current [25], even though such currents are expected (via the bootstrap current) for quasi-axisymmetric configurations. In contrast, we have confronted the bootstrap current and its implications by designing the NCSX around a high-β quasi-axisymmetric configuration with self-consistent bootstrap current profiles. The NCSX design targets the issues of β limits and high-β disruption avoidance which are critical for the long-term attractiveness of the concept.

The scientific success of these experiments requires a continuing vibrant 3D theory and numerical modeling program. This is required in the near term to complete the design of NCSX and other new experiments, and in the longer run to enable interpretation and understanding of experimental results. General 3D topics of high importance include:

- Non-linear MHD stability analysis, including Alfvenic eigenmodes,
- Non-linear micro-stability and turbulence simulation, coupled with neoclassical transport effects,
- Edge modeling,
- Integrated discharge analysis and simulation,
- Faster 3D equilibrium calculations including islands, stochastic regions, and neoclassical effects,
- 3D equilibrium reconstruction and analysis,
- RF wave propagation and damping.

We will encourage and endorse efforts to address these issues, and we welcome collaborative approaches.

1.5 Relationship to Advanced Tokamak and Spherical Torus Programs

The physics challenge for the tokamak program is to integrate the recent advances in understanding to allow stable high-β_T (~5%) steady-state plasmas without disruptions, as indicated in the ARIES-RS study [26], for example. Sustainment of the plasmas should minimize the externally driven current, to minimize recirculating power. Meeting these challenges is the focus of the Advanced Tokamak and Spherical Torus research programs. As presently understood, this will require nearby conducting walls to stabilize the external kink instability, active feedback stabilization of the resistive wall instability, active feedback stabilization of neoclassical tearing modes, and current profile control using external current drive.

The proposed NCSX program complements this research by investigating an alternative innovative approach, where these instabilities are stabilized by the design of the three-dimensional shape of the plasma. Control of the equilibrium is maintained using the external coils to modify the plasma shape through the externally-generated poloidal magnetic field.

1.6 NCSX Contributions to Fusion Energy

If NCSX is successful, the compact stellarator will provide innovative solutions to some of the critical problems that lie on the path to an attractive fusion power plant. A key challenge for magnetic fusion energy research is finding a toroidal plasma configuration that has high power density, can be sustained with low power recirculation, and does not disrupt. The compact stellarator complements the present strategies based on advanced tokamaks, spherical torus, or currentless stellarators by combining helical fields and three-dimensional shaping from external coils at moderate aspect ratio to produce a configuration that could be passively stable and achieve tokamak-like power-densities without the need for external current drive. By optimization of the magnetic field structure, compact stellarators should have tokamak-like confinement, including the ability to manipulate the turbulent transport with flows. Thus, while compact stellarator research will surely broaden our understanding of magnetic fusion science, it may also provide an attractive[27] reactor solution, see Figure 1-1.

The NCSX will advance the compact stellarator as a fusion confinement concept by testing the understanding of the basic physics principles that govern its performance and ultimately will determine its attractiveness. The concept-improvement goals of NCSX are to:

- Determine conditions for high-beta disruption-free operation, compatible with bootstrap current and external transform in a compact stellarator configuration.
- Determine beta limits and limiting mechanisms.
- Demonstrate reduction of neoclassical transport by quasi- axisymmetric design.
- Demonstrate anomalous transport reduction by flow- shear control, using reduced flow damping of quasi-axisymmetry.
- Verify the stabilization of equilibrium islands and neoclassical tearing modes by design of magnetic shear.
- Test stability of Alfvenic modes and their coupling to fast ions in a 3D quasi-axisymmetric configurations
- Develop a power and particle exhaust solution compatible with a compact stellarator.

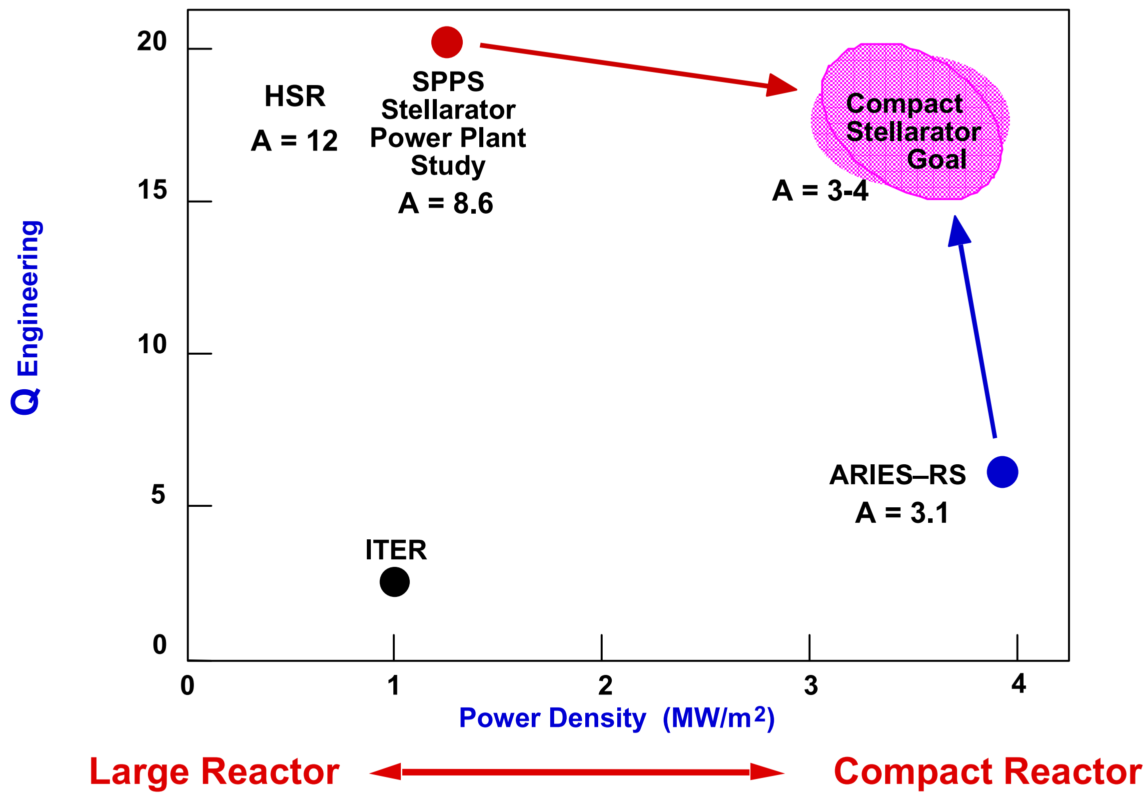

Figure 1-1. Compact stellarators, by combining the low recirculating power (high Q_{eng}) and disruption immunity of stellarators with the low aspect ratio and high power densities of tokamaks, could be attractive

This database, and the improved understanding of 3D plasma confinement that it will provide, will provide the basis for the design of follow on experiments and possible reactor design \]studies.

While the configurations investigated in the design of NCSX are not optimized as possible reactor designs, they do illustrate the advantages of compact stellarators in a reactor setting. Chapter 14 presents an initial analysis of the reactor implications of compact stellarator designs based upon NCSX-like configurations.

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