Chapter 2 — NCSX Physics Design

The National Compact Stellarator Experiment will be the primary research tool in the U.S. for understanding the physics of compact stellarators. Theoretical calculations have demonstrated the exciting promise of quasi-axisymmetric stellarators (QAS) for fusion science. They show that a high-beta, low-aspect-ratio three-dimensional plasma configuration can have the magnetic symmetry of a tokamak, with similar charged-particle drift orbits and plasma flows. Externally generated helical fields provide most of the rotational transform and shape the plasma so as to stabilize the limiting instabilities such as external kink, vertical, ballooning, and Mercier modes without requiring a conducting wall or feedback control. Its bootstrap current will provide a significant fraction of the rotational transform and prevent the unstable growth of magnetic islands. The potential benefits of such a configuration are large: a compact, easily sustained configuration with the good plasma performance of a tokamak.

2.1 Physics Design Overview

NCSX has been designed to accomplish the mission and goals presented in Chapter 1. It is designed around a computed QAS fixed boundary plasma configuration, shown in Figure 2-1. It has three periods, an aspect ratio $R/\langle a\rangle = 4.4$, and strong axisymmetric shaping (average

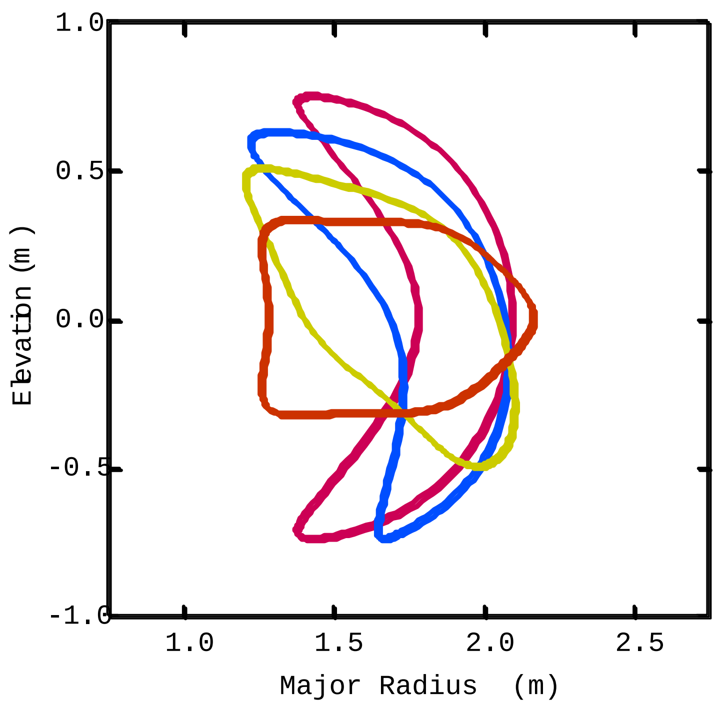

Figure 2-1 Plasma cross-sections at the symmetry planes (the bean- and bullet-shapes) and at equally spaced toroidal angles in between

elongation ~1.8), as well as the three-dimensional shaping that is clearly evident in the figure.

The shape was optimized for stability and minimum ripple (good quasi-axisymmetry) at $\beta = 4\%$. The effective (for neoclassical transport) ripple is 0.6% at $S = 0.5$ (where S is the normalized toroidal flux) and about 3.4% at the edge. An axisymmetric configuration would be unstable without a conducting wall at this β value, so this design provides a clear test of the stabilizing effect of three-dimensional shaping. A moderately broad pressure profile, typical of stellarators, and the bootstrap current profile shown in Figure 2-2 were used in the optimization. The bootstrap current provides about one-fourth of the rotational transform at the edge (Figure 2-3), while the remainder is provided by coils. The rotational transform increases monotonically, except very near the edge, from about 0.4 ($q \approx 2.5$) to 0.65 ($q \approx 1.5$). Except for a small removable island chain at the $\iota = 0.6$ surface, the plasma has good magnetic surfaces all the way to the edge.

The NCSX uses modular coils (Figure 2-4) to provide the helical magnetic field. Modular coils resemble toroidal field coils with out-of-plane deformations. The three coils on the v=0 symmetry planes were extended radially to allow tangential access for neutral beams and diagnostics. The design also includes toroidal field coils, poloidal field coils, and trim coils for

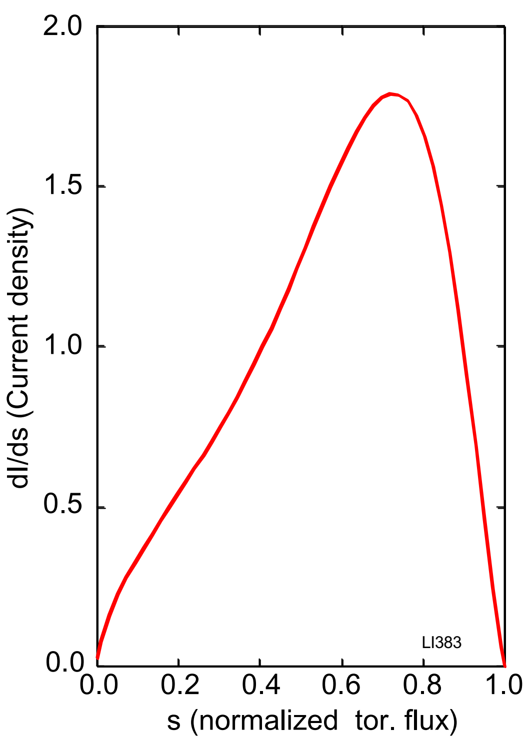

Figure 2-2 Current profile

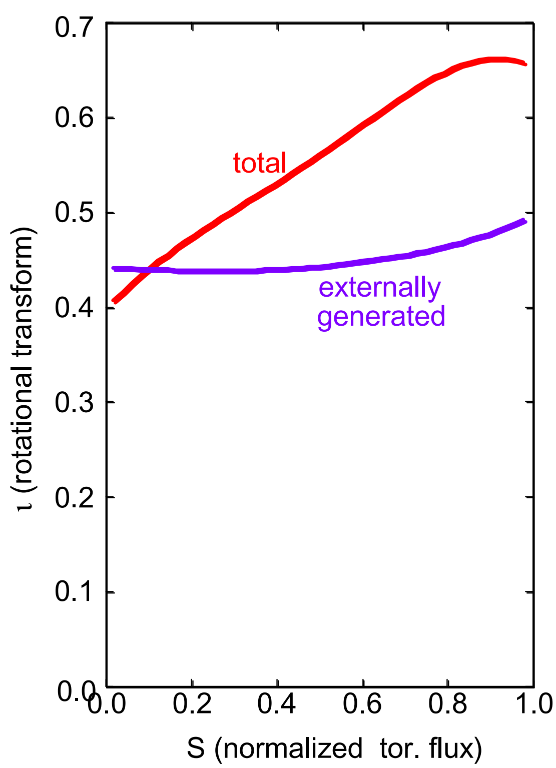

Figure 2-3 Rotational transform profile

Figure 2-4 1017 modular coil design

equilibrium flexibility. The coils are designed to produce good magnetic surfaces and to approximately reconstruct the physics properties of the reference plasma. The additional coils provide substantial experimental flexibility to test the physics while maintaining good quasi-axisymmetry and magnetic surfaces.

The NCSX design provides the physics properties and flexibility necessary to develop the physics of compact QA stellarators. The configuration physics design goals and achievements are:

- Reference beta high enough to demonstrate the success of three-dimensional shaping in stabilizing a sustainable toroidal configuration - achieved $\langle\beta\rangle = 4\%$.
- Aspect ratio ($A = R/\langle a\rangle$) substantially less than in existing drift-optimized stellarator designs - achieved A 4.4 (HSX: $A = 8$; W7-X: $A = 11$).
- Stellarator shear in the rotational transform profile - achieved $\iota(0) = 0.39$, $\iota(a) = 0.65$, monotonically increasing except very near the edge.

- At least half the rotational transform provided by coils, to be conservatively disruption-resistant - achieved ~3/4 from coils.
- Good magnetic surfaces – achieved by two independent means: via the application of island removal methods in the configuration design; and via "reversed shear" giving neoclassical island reduction and healing.
- Good quasi-symmetry – achieved an effective ripple of ~3.4% at the edge, helical ripple neoclassical transport low compared to toroidal, counter-injected beam-ion losses (30%) tolerable for beta experiments at B = 1.2 T.
- Stability to ballooning, external kink, vertical, and Mercier modes - achieved at $\langle\beta\rangle = 4\%$..

The key physics design goal for NCSX has been developing a flexible and robust coil design. Substantial advances in physics tools (e.g., equilibrium codes) and design methodologies have resulted in dramatic progress in this area beyond the interim NCSX design that was presented to the Fusion Energy Sciences Advisory Committee (FESAC) in mid-1999. New plasma design strategies have overcome the problem of edge stochasticity seen in the reference plasma in the 1999 design. Physics-based strategies for targeting surface quality as a coil design objective have been developed and applied. A different coil topology has been adopted. New tools for evaluating the flexibility of candidate coil designs, non-existent in mid-1999, have now achieved a high level of productivity.

The NCSX coil design now exhibits 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. They 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. The coils can provide a wide range of plasma current (up to 420 kA) and ramp rates (up to 3 MA/s), background toroidal field (±0.3 T), axisymmetric poloidal fields from the PF coils, and helical fields from the modular coils. Coil designs have been generated with these physics capabilities:

- Good plasma physics performance (stability and low ripple) is achieved over a wide range of beta and plasma current values for fixed profile shapes.
- Stable paths from vacuum to high-beta conditions are available.
- Good plasma physics performance (stability at $\langle\beta\rangle$ 3% and low ripple) is achieved over a wide range of profile shapes for fixed plasma current values.

- Shape flexibility to vary the theoretical stability boundaries for fixed pressure and current profiles is available. The stability beta limit can be reduced to below 1% or, by trading quasi-symmetry for stability, it can be raised to at least 4%.
- Flexibility to vary ι up and down at fixed shear, to vary the shear at fixed ι(0) or at fixed ι(a) is available.
- Good magnetic surfaces are achieved, by design, in vacuum and in the reference high-beta state. Resonant trim coils are provided in case they are needed to maintain good surface quality in other configurations.

Machine size, heating power, and pulse length are set by a combination of physics and practical considerations, including cost. Here we describe the main parameters of the facility and their motivation.

- <u>Major radius R=1.4 m</u>. Since the machine cost increases strongly with size, we have sought to set the major radius at the minimum value compatible with NCSX physics goals. The limiting factor is access. At the chosen size, available neutral beam injectors can be accommodated with acceptable injection losses. Access for diagnostic viewing and for personnel entry to the vacuum vessel interior is critical for the repair and modification of in-vessel hardware. While such access is available in the design, it is marginal, so improving access at fixed overall machine size is a high priority goal for design optimization during conceptual design.
- <u>Heating power: 3 MW initially, upgradable to 12 MW</u>. The facility will be initially equipped with 3 MW of neutral beam heating from two tangential injectors, one co- and one counter-, using two of the four existing 50 keV neutral beams from PBX-M to minimize cost. The device will be designed with provision for up to 12 MW of auxiliary heating: all four PBX-M beams (6 MW total) as well as wave launchers for up to 6 MW of radio-frequency heating.

With the full complement of hydrogen neutral beams (6 MW), the reference beta value of 4% and a collisionality $\nu^*=0.25$ are predicted to be attained at B = 1.2 T, assuming an enhancement factor of 2.9 times the ISS95 stellarator confinement time scaling, or 0.9 times the ITER-97L tokamak scaling (i.e., close to L-mode).

With the initial 3 MW beam system and the same collisionality and enhancement assumptions, the predicted beta value is 2.5 – 2.8%. The 3 MW system provides balanced injection necessary for discharge control (Chapter 10) and enough power to investigate enhanced confinement regimes, assess confinement scaling, and acquire a database for deciding on the amount and type of any plasma heating upgrades that might be needed.

Further details on projected plasma parameters and profiles are provided in Chapter 8.

- <u>Minimum magnetic field for good confinement of neutral beams: 1.2 T</u>. A minimum product of magnetic field and major radius B×R=1.7 T-m is required to keep the losses of tangentially counter-injected 40-keV hydrogen ions below 30%. For the R = 1.4 m design, the minimum magnetic field is 1.2 T.

- <u>Maximum magnetic field at full external transform: 1.7 T; minimum heating pulse length: 0.3 s</u>. The available heating systems can provide full heating power for 0.3 s pulse lengths initially. A cost-effective magnet design provides a 0.46 s flattop at 1.7 T, which allows 0.1 s for plasma initiation, 0.06 s for a rapid (3 MA/s) inductive current ramp (to produce bootstrap-like current profiles), 0.1 s to heat to full beta, and 0.2 s for the plasma to relax at full current and full beta.

  The currently available range of magnetic fields (1.2-1.7 T) will support physics studies of low-collisionality transport and confinement scaling, although it is marginal for these purposes. A factor of 2 would be more desirable. Using the supplementary toroidal-field coils, the magnetic field can be increased to 2 T, but with reduced external rotational transform. Increasing the maximum magnetic field towards 2.4 T through modular coil design improvements is an objective for design optimization efforts in the conceptual design phase.

- <u>Upgrade heating pulse length: 1.1 s</u>. At B = 1.2 T the coils and power supplies are capable of 1.1 s heating pulses. Realistic upgrade paths are available to extend the heating pulse length to match this value, which would allow time for true equilibration of the configuration (flat voltage profiles). The priority of such upgrades, and the optimum path to take, will depend on experimental results from the initial phases of NCSX operations.

In the remainder of this chapter we describe the physics design of the NCSX facility. We describe the methods used to develop the reference plasma configuration and associated coils, which we have introduced in this section, and then evaluate them in terms of their physics properties and flexibility. Then we summarize power and particle handling capabilities, plasma heating, diagnostics, and the planned research program. Later chapters describe engineering solutions and physics analyses in greater detail.

2.2 Reference Plasma

The NCSX reference plasma configuration was designed through the use of an optimization code. The optimizer adjusts the values of about 40 parameters specifying the shape of the plasma boundary to attain stability and other targeted configuration properties. For this purpose, the value of $\langle\beta\rangle$ was fixed at 4.25%, the pressure profile was specified, and the current profile was taken as the calculated bootstrap current (see Chapter 4). Quantities targeted in the optimizer for this calculation included: a measure of quasi-axisymmetry (the sum of the squares of the non-axisymmetric Fourier components of B in Boozer coordinates, with an adjustable weighting of contributions from different flux surfaces); the eigenvalue of the most unstable external kink mode calculated by the TERPSICHORE [1] code; ballooning eigenvalues calculated by the VVBAL or the COBRA [2] codes; the rotational transform on one or two flux surfaces; the complexity and current density of an external surface current constructed by the NESCOIL [3] code representing a first approximation to a set of coils. (The surface-current coil solving method is described in the next subsection.) The properties of these MHD equilibrium configurations are completely determined by the current and pressure profiles, as well as the plasma boundary, which is represented as a finite sum of Fourier harmonics for R and Z. Varying the Fourier harmonics as independent variables in the optimizer allows us to seek stellarator configurations in which the various criteria are satisfied as well as possible. The VMEC code [4] is used to calculate the MHD equilibria needed to evaluate the physics targets for arbitrary values of the independent variables.

In the design process, candidate configurations generated through use of the optimization code were evaluated in more detail with an array of standalone codes that consume too much computer time to be practical to run as part of the optimizer. Ion thermal confinement times were estimated by Monte Carlo simulation using the GTC code to simulate the full ion distribution function [5]. The energy losses of neutral beam ions were also calculated using Monte Carlo simulations [6,7]. The CAS3D MHD stability code [8] was used to confirm the Terpsichore calculations, and to evaluate free-boundary kink and vertical stability without relying on currents in the wall to provide stabilization. (Terpsichore calculations had a conducting wall at about three times the minor radius, where it was calculated to have little impact on the stability.) The integrity of the flux surfaces was monitored with the PIES code [9], which solves the three-dimensional equilibrium equations using a general representation of the magnetic field, so that it can handle islands and stochastic regions. The results of the calculations by this array of codes were used to fine-tune the optimizer and to help steer the design process.

Table 2-1 Properties of the reference plasma configuration

Number of periods	3
$R/\langle a\rangle$	4.4
$\langle\beta\rangle$	4.1%
$\iota(0)$	0.40
$\iota(a)$	0.66
$\iota(0)$ vacuum	0.45
$\iota(a)$ vacuum	0.49
R (meters)	1.42
$\langle a\rangle$ (meters)	0.33
w_{min} (meters)	0.13
I_P (kA) at B=1.7 T	175
ε_h effective	0.6%

Table 2-1 tabulates the key properties of the NCSX reference plasma configuration, with a major radius of 1.4 m. The properties of the reference plasma configuration have been evaluated at a $\langle\beta\rangle$=4.1%, where it is designed to be marginally stable. Very recent high-resolution stability analyses have found that the plasma has a relatively high order (n = 11, m = 17) mode, localized near the edge (see Chapter 5). The values of ι at the magnetic axis and the plasma boundary are given for the full current, full $\langle\beta\rangle$ equilibrium, and for the vacuum stellarator field having the same boundary. The quantities w_{min} and I_P are, respectively, the minimum half-width of the cross-section and the total plasma current (evaluated for B= 1.7 T). The quantity ε_h is an effective ripple strength evaluated at a normalized toroidal flux S=0.5 (r/a≈0.7). It is calculated numerically to match the $1/\nu$ transport regime [10].

As initially generated by the optimizer, the plasma configuration was calculated to have a total island width of about 15% of the minor radius at $\langle\beta\rangle$ = 4%, dominated by a single island chain at ι = 0.6 with an island width of about 10%. The islands have been removed, using a method based on the PIES code, either by a slight further adjustment of the boundary shape or by a slight perturbation of coils that have been designed based on the unmodified plasma configuration. The adjustment process, as well as the general subject of flux surface calculations, will be discussed in Chapter 4.

2.3 Helical-Field Coil Design Methodology

The design of coils for low aspect ratio stellarators presents new challenges due to the strong harmonic coupling between toroidal (1/R) and helical components of the field. In addition to preserving the good physics characteristics of the reference plasma, the coil set must be capable of satisfying engineering constraints. Feasibility constraints placed upon the coils include limitations on the allowable current density in the coils; lower bounds on the radius of curvature and wind-back characteristics (complexity) of the coils; separation of the coil winding surface from the plasma; and access through the coils for heating, diagnostics, and maintenance personnel.

The design methodology for obtaining coils for NCSX is based on the "reverse engineering" technique pioneered in the late 1980's by the W7-X group under Nuehrenberg, *et al*. Earlier coil design efforts worked with direct parameterizations of helical coils to obtain large aspect ratio (A) configurations with certain basic physics properties. A dramatic departure from this approach was to realize that a much broader spectrum of configurations could be investigated by separating the plasma optimization process from the coil design. Not only does this allow better optimization of physics properties, it also makes available a much larger design space (*e.g.,* multiple topologies and geometrical properties) for finding suitable coils. At large A, the plasma and coil optimization could be done independently of each other. However, at the lower A of NCSX, it has been found beneficial to weakly couple these two optimization processes in such a way that certain coil properties, approximated via a surface-current representation, are included directly in the physics optimization procedure, as mentioned in Section 2.2. In this way, physics solutions can be guided toward regions of parameter space where realizable coils exist.

In finding NCSX coil solutions by the reverse engineering method, the boundary shape of the reference plasma is targeted by minimizing the root-mean-squared normal component of the total magnetic field on the plasma surface. The plasma boundary is prescribed by a fixed-boundary solution from the VMEC equilibrium code. Coil solutions typically reproduce the desired physics properties when the total residual B-normal is less than 0.6% (in units normalized to the local value of |B|) on average and less than 2.5-3% peak. The value of B-normal includes contributions from the plasma's internal currents (which are calculated using the BNORM code) and, optionally, from specified background coils as well as from the helical-field coils.

Filamentary helical-field coils are arranged on a coil winding surface (CWS) that approximately conforms to the reference plasma boundary, with enough separation (typically about 18 cm on the inboard and 28 cm on the outboard side) to allow space for plasma-facing components, the vacuum vessel, and finite coil cross section. The CWS shape can be fixed or it can be varied as part of the optimization process, which has the benefit of further improving the physics and engineering objectives. Two methods are used for finding an optimum filamentary coil set on the CWS. The first is to solve for a surface-current distribution on the CWS and then convert it to a discrete-filament representation. The second is to solve for the filament geometries directly using a parametric representation of their trajectory on the winding surface.

The first method (without the discretization of this continuous distribution into coil filaments) is incorporated in the NCSX physics optimization process. The Neumann Equation Solver code NESCOIL [3] is used to find the current distribution that minimizes B-normal. To improve the targeting of both physics objectives (represented by B-normal) and engineering objectives (*i.e.,* low current density, gentle bend radii of the coils) for NCSX, a singular value decomposition (SVD) technique was developed. The new code, NESVD, runs as fast as the original NESCOIL code, which enables it to be placed within an optimizer loop so that the shape of the coil winding surface can be adjusted to yield further improvements in the target criteria. A newly developed method based on a genetic algorithm (GA) is available, if desired, to convert surface current potentials computed in NESCOIL or NESVD to an optimized set of discrete coils.

The second method for determining coils, the direct-filament method, has led to the modular coil designs that are of most interest for NCSX. This method uses the COILOPT code, which optimizes the coil geometry on a toroidal winding surface that is well separated from the plasma boundary. A parametric representation of the filamentary coils is used, with constraints to ensure preservation of stellarator symmetry. In addition, parameters representing the winding surface geometry can be varied to further improve the optimization targets. Target functions in the optimization problem include the B-normal error, the lengths of individual coils, the minimum coil radius of curvature, the minimum separation between adjacent coils, and the minimum separation between the coils and the plasma. Geometric constraints are introduced in the form of weighted penalty functions.

Methods for targeting magnetic surface quality as a coil design objective have been investigated. To date, the most success has been achieved using the PIES code to compute small modifications to the modular coil geometry determined by COILOPT. Coefficients in the modular coil representation are adjusted to cancel the normal magnetic field components at the dominant resonant island chains in the plasma interior. The resulting modified plasma has significantly improved magnetic surface quality (residual islands widths are greatly reduced). This method is described in greater detail in Chapter 4. Other, less computationally intensive methods have also been explored. One approach is to use an estimated island width calculated from the resonant magnetic field component at rational surfaces, thereby avoiding the time-consuming PIES calculations. This approach has not yet been fully tested but appears promising. Another approach, implemented within the NESVD scheme, adds the ability to minimize the resonant field errors on the plasma surface. The square of the field line displacement from a nominal flux surface can be written as a linear function of the Fourier components (in a straight magnetic field line system) of the normal field error on that surface. This may provide an ability to further improve magnetic surface quality at the edge.

2.4 Helical-Field Coil Designs

Modular coils were selected as the reference coil concept for NCSX because they were found to provide the best physics properties of the various topologies that were studied. Coils of this type have operated successfully in the W7-AS and HSX stellarators and a superconducting version is being constructed for the W7-X stellarator. In this section we introduce the basic NCSX modular coil design which reproduces reference plasma physics properties and provides neutral beam access, and which has been adopted as the reference for engineering studies. In Section 2.7 we discuss the physics validation of the reference design as well as design variations that have been explored in efforts to understand potential improvement paths.

Modular coils were designed using the reference plasma configuration as a target equilibrium. An initial modular coil solution, designated 0907, contained 7 coils per field period (4 unique coil types) including a coil on the v = 0 symmetry plane (the bean-shaped cross section). The number of coils per period (7) was chosen because it resulted in very low error in B-norm on the plasma boundary. This solution was found with COILOPT, targeting the minimum coil-coil separation ($\Delta_{cc,min}$), minimum radius of curvature (ρ_{min}), and coil length; and allowing the winding surface to vary. As a final step in the optimization procedure, the length constraints were removed and a minimum plasma-coil separation ($\Delta_{cp,min}$) target was imposed. Results of the optimization are given in Table 2-2. As a preliminary check, a free-boundary VMEC equilibrium was calculated using this modular coil solution and the reference plasma profiles, and it was found to closely match the original (fixed boundary) plasma configuration.

Table 2-2 Modular coil optimization results

Design	ID	N_c	δB_{avg}(%)	δB_{max}(%)	$\Delta_{cc,min}$(cm)	ρ_{min}(cm)	$\Delta_{cp,min}$(cm)
M2	0907	7	0.57	2.55	13.4	11.0	23.3
M3	1017	7	0.61	2.61	14.8	12.3	23.3

In order to achieve adequate access for the neutral beams, it was necessary to modify the 0907 coil set by extending the modular coils near the v = 0 symmetry planes in radius on the outboard side while preserving the low B-normal errors of solution 0907. Starting with the 0907 coil set, the coils were re-optimized using a modified winding law that imposes the radial extension on the coils, while maintaining stellarator symmetry. The resulting solution, the 1017 coil set, is compatible with the engineering design of the neutral beam injection system and has field errors comparable to 0907. The 1017 coil configuration is displayed in Figure 2-4, and a free-boundary reconstruction of the plasma with these modular coils is shown in Figure 2-5.

Coil solutions are not unique, and thus there is freedom available to optimize the coils for given physics goals. The 1017 design was selected for engineering assessment and an attractive machine concept has been developed based on it. However, at this stage of the design it is

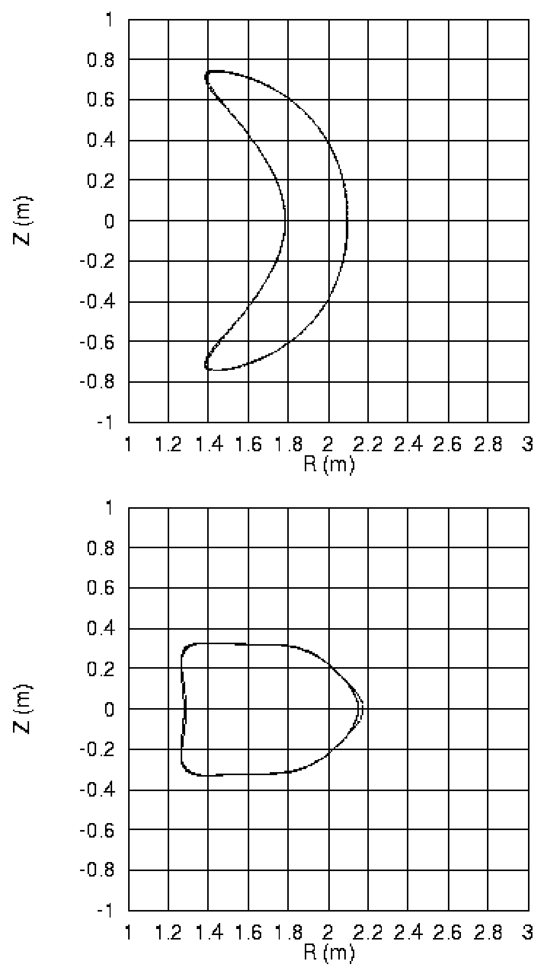

Figure 2-5 Free-boundary reconstruction (solid) with modular coil solution 1017, compared with the target (fixed boundary) plasma boundary (dashed), shown for the v=0 (top) and v=1/2 (bottom) toroidal planes

important to continue trying to optimize the design for manufacturability, access, and physics performance. Therefore, alternatives to the 1017 design have been studied, with the aim of reducing current densities, reducing bend radii, increasing coil-to-coil spacing, and reducing the number of coils consistent with required physics properties. The final coil configuration for the conceptual design will be selected later in the design process.

2.5 Robustness and Flexibility

In order to achieve its scientific goals, the NCSX device must be capable of supporting a range of variations in plasma configuration. While the helical-field coils provide the physics properties required for a particular high-beta equilibrium as a minimum condition, the complete coil design must provide the flexibility and robustness necessary to meet the research needs of the program. In particular, the coils must be capable of:

- Providing access paths to high-beta states, starting from an initial vacuum state.
- Providing good physics properties over a wide range of plasma pressure and current profiles in order to be robust against physics uncertainties.
- Being able to vary the physics properties of the plasma configuration, including rotational transform and shear, so that the understanding of how the physics depends on configuration parameters can be tested experimentally.

To meet flexibility and robustness requirements, the basic NCSX modular coil design is augmented in several ways:

- The 21 modular coils are grouped into four circuits, one circuit for each distinct coil shape. The like-shaped coils are connected in series and circuit currents are controlled independently.
- A supplementary toroidal-field coil set provides a weak (±0.3 T) 1/R field that can be in either direction relative to the modular coils. It provides the capability to vary the external rotational transform at fixed shear.
- A poloidal field coil system provides inductive current drive (1.4 Wb of flux swing with low stray fields in the plasma region, and current ramp rates up to 3 MA/s) in either direction. It also provides low-order axisymmetric multiple fields (dipole, quadrupole, hexapole, octupole) for plasma shape control. Several poloidal field coil design variants are analyzed in later chapters, including representation in terms of a toroidal multipole expansion. The final design of the poloidal field coils will be established during conceptual design.

- A trim coil system provides control of helical field components with poloidal mode numbers m = 5 and 6 and toroidal mode number n = 3 at specific resonant surfaces in the plasma. These provide capability for magnetic island width control over a wide range of plasma configurations, in case such control proves necessary to meet flexibility and robustness requirements. The chosen design approach allows the addition of trim coils for other poloidal modes if necessary. Trim coil design is described in more detail in Sect. 2.6 and the status of physics validation is given in Chapter 4.

Physics evaluations of NCSX flexibility and robustness are described in detail in Chapters 9 and 10.

2.6 Trim Coils for Maintaining Good Surfaces

The modular coils, in combination with a rich poloidal coil set plus a background 1/R field from a toroidal coil set provides the flexibility and robustness to reach a wide range of plasma states. This includes the time evolution of the discharge from vacuum to the high-beta state. It includes operation with different current, pressure or iota profiles. Using the island healing method previously mentioned, the basic coil set is designed to produce good surfaces in the reference high-beta state but may not have sufficient control over island producing resonances for other plasma states. For that we introduce a trim coil set.

The trim coils were designed to target the specific resonances based on the range of iota encountered. Hence, the resonances were limited to the lower poloidal modes (m=5,6) for the n=1 toroidal mode, the dominant modes for the LI383 family. Trim coils for other modes will be added if found necessary.

The design goal for trim coils was to provide coil sets which strongly couple to the targeted resonances on interior plasma surfaces with minimum current demands and minimal impact on the radial field at the plasma boundary. The design was guided by an initial evaluation of where trim coils would be most effective, on a specified surface conforming to the plasma. It was found that the regions of strongest coupling occur around v=0 inboard and outboard. This is the banana or crescent section of the plasma. Not surprisingly, this is where the resonant interior surfaces are closest to the trim coil winding surface.

Trim coils were located in these regions of strong coupling, leading to a simple racetrack shape. Four coils inboard and four outboard (see Figure 2-6) at each period provide approximately 33% of the effectiveness of an ideal helical winding for the m=5 resonance, while covering only a much smaller portion of the surface. The m=6 winding is more than 40% effective. The coverage may need to be reduced further to avoid interference with neutral beams.

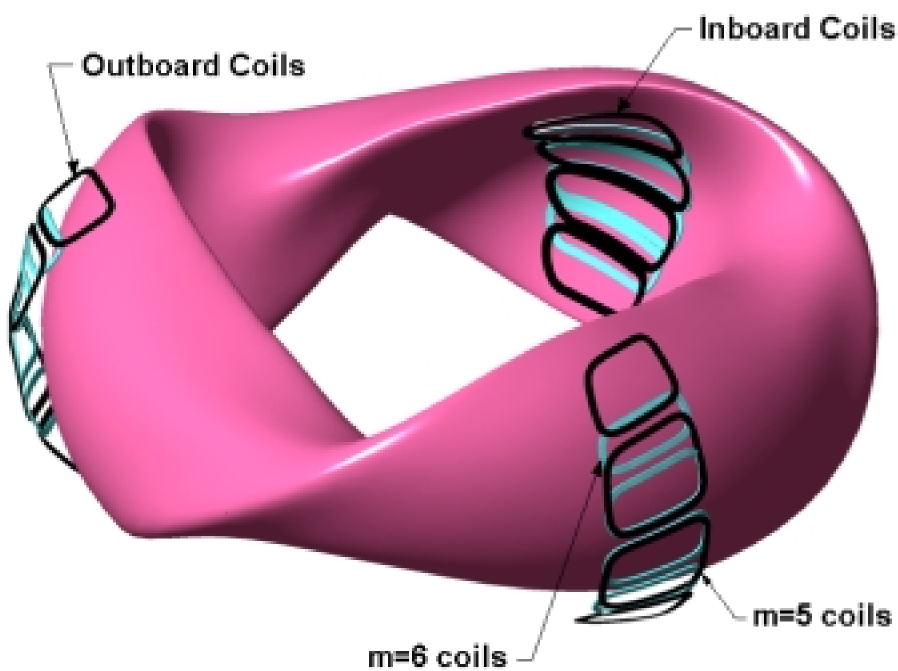

Figure 2-6 Resultant m=5 and m=6 trim coil sets

The offset of this surface was limited by engineering constraints. The current demands were seen to increase rapidly with distance from the resonant surface. This, plus accessibility to coils at the inboard side of the plasma, led to a selection of a surface inside the vacuum vessel behind the first wall. This also allowed locating trim coils between the co- and counter- neutral beams.

2.7 Physics Evaluation of Coil Designs

The main stellarator coils of NCSX are designed to reproduce the reference plasma configuration (LI383) at the state of full current and full beta, using the reverse engineering method discussed in Section 2.3. In this section, we discuss the physics validation of coil designs by evaluating the physics properties of free-boundary equilibria that are reconstructed from candidate coils. We make use of capabilities that are provided for flexibility: supplemental toroidal and poloidal field coils and the freedom to energize modular coils independently. These enable us to further improve and optimize plasma performance. Most of the physics and engineering analyses reported in this document are based on the 0907 or 1017 coil designs described in Section 2.4, but we also report on some more recent designs that indicate directions for improvement.

2.7.1 Modular coils for the reference plasma configuration

Summarized in Table 2-3 are selected sets of coils that are discussed in this section. All have seven modular coils per period and four unique types of coils. All designs preserve stellarator symmetry and in each period have a coil on one of the symmetry planes, either at the bean-shaped cross-section (v=0) or the oblate cross section (v=1/2). The M2 (0907) is an early design which did not accommodate tangential access needed for the neutral beams, but is used in some of the physics analyses. The M3 (1017) coil set has the outer legs of two coil types extended to accommodate tangential access. The M3 is used as the reference for engineering studies and most of the physics analyses. The M8 and M7 are later designs exploring methods for improvement. In both of these design the coil currents were allowed to vary independently in the design process and, in the case of M8, the constraint on the minimum radius of curvature was relaxed to achieve a smaller fitting error. The M7 design explores placing the symmetry coil on v=1/2 cross section. While the M7 has not specifically been designed to meet tangential access needs, the coil modifications that would be required are much less than in designs with a v=0 symmetry coil.

Table 2-3 Definition and characteristics of modular coils

Physics Evaluation ID	M2	M3	M8	M7
Coil Design ID	0907	1017		
Use in design process				
Engineering reference		X		
Physics analyses	X	X		
Exploration			X	X
Extended coil for tangential access?	No	Yes	Yes	No
Symmetry coil position	v=0	v=0	v=0	v=1/2
Treatment of coil currents in design optimization.	Constrained to be equal	Constrained to be equal	Allowed to vary	Allowed to vary

2.7.2 Methodology of Evaluation and Optimization

To evaluate the physics properties of plasmas reconstructed from coil designs, we represent each coil as concatenated line segments. Magnetic intensities inside the coil envelope for each design were calculated at 16 toroidal planes (per field period), with each plane further divided into 101 by 101 meshes in R and Z. This field map was used by the free-boundary VMEC [4] to construct equilibria consistent with the prescribed pressure and current profiles and the enclosed toroidal flux. In our evaluation, we used earlier machine parameters (R= 1.7 m, B = 2 T, and I_p = 250 kA) which differ from the current machine parameters (R = 1.4 m, B = 1.7 T, and I_p = 175 kA). The pressure and current profiles were the same as those in the reference configuration (Section 2.2).

As in the design of the reference plasma (Section 2.2), the VMEC code is the main tool used to calculate free boundary equilibria. These equilibria were evaluated by TERPSICHORE [1] and COBRA [2] for the kink and ballooning stability. The conducting wall, which was assumed to be conformal to the plasma boundary, was placed at a distance 2.5 times the minor radius from the plasma-vacuum interface. The NEO [10] was used for the effective helical ripple, $\varepsilon_{h,eff}$, in the asymptotic collisionless $1/\nu$ regime (the thermal helical transport scales as $\varepsilon_{h,eff}^{3/2}$). The ORBIT3D [11] code was used to evaluate the fraction of neutral beam energy loss. The n 0 components of the magnetic spectrum in Boozer coordinates were summed together as a simple figure-of-merit (χ^2_{Bmn}) for the quasi-symmetry of a configuration. In VMEC it is assumed that flux surfaces are nested without islands and stochastic regions. Equilibrium and flux surface analyses are discussed in Chapter 4, while the stability and transport codes are discussed in Chapters 5 and 8, respectively.

For the neutral beam loss calculation with ORBIT3D, hydrogen beam particles were launched at a toroidal plane corresponding to v=0 and at a tangency radius which was aligned with the magnetic axis. The beam particles had an initial energy of 40 keV and a deposition profile taken from a PBX high beta discharge. The background plasma was assumed to be H and had parabolic density and temperature profiles. The impurity species was assumed to be A=18 and Z=9, the density of which was 1/10 of the main ion species.

As in Section 2.2, a Levenberg-Marquardt technique was used to optimize coil currents to achieved desired physics properties, such as the quasi-symmetry, kink or ballooning eigenvalues. The pressure profile was not optimized, however, so further increases in beta are likely to be possible. Because the technique is a local gradient method and the topology of the optimization space is complex with abundant hills and valleys, further constraints, such as regularization of currents, or trials with different weighting may be needed in the search of the global optimum.

The results to be discussed below are mostly based on calculations with reasonable, but nevertheless limited resolutions. Detailed convergence studies, such as those shown in Chapter 5, are extremely time consuming and can only be done for the reconstructed cases at a later time. While results of some individual cases may change when the fine resolution calculation is eventually carried out, the general conclusions of the study presented below will remain valid.

2.7.3 Properties of Plasmas Reconstructed From Coils

In Table 2-4, we show physics properties of plasma configurations reconstructed from these four coil designs. Each case represents a free-boundary equilibrium in which the coil currents were adjusted using the optimizer to target MHD stability and quasi-axisymmetry. In each case, the four modular coil group currents (one for each unique coil type) were allowed to vary independently. An additional PF coil set was used (except for M8, which used axisymmetric multipole fields instead) and a 1/R TF field was used in the plasma optimization. The properties of the reference plasma configuration are shown for comparison.

Table 2-4 Comparison of properties of optimized free-boundary reconstructed plasmas

Coil ID Case No.	Reference plasma (LI383)	M2 M2.3.Z07	M3 M3.3.Z03	M8 M8.3.K594	M7 M7.3.K03
A	4.36	4.25	4.16	4.17	4.37
β	4.19	4.24	4.10	4.09	4.18
R (m)	1.734	1.738	1.738	1.726	1.736
<a> (m)	0.397	0.408	0.418	0.414	0.397
R-min (m)	1.209	1.206	1.196	1.186	1.212
R-max (m)	2.173	2.186	2.208	2.176	2.163
Z-max (m)	0.764	0.772	0.775	0.774	0.756
Min plasma-limiter distance (cm)	1.14	0.36	-0.80	-1.26	0.98
$\iota(0)$	0.394	0.409	0.413	0.429	0.402
$\iota(a)$	0.655	0.655	0.653	0.648	0.653
ι, Max	0.662	0.663	0.659	0.656	0.662
λ, Kink ($\times10^4$) [1]	Stable (3)	Stable (3)	Stable (3)	Stable (3)	Stable (3)
λ, Ballooning, ζ=60 [2]	0.91-0.96 (0.11)	0.92-0.96 (0.07)	Stable	Stable	0.91-0.96 (0.14)
χ^2_{Bmn} ($\times10^4$)					
S=0.3	0.5	0.7	0.8	0.8	0.6
S=0.5	1.7	2.2	2.5	2.6	1.8
S=0.8	6.9	8.9	10.3	10.7	7.0
$\varepsilon_{h,eff}^{3/2}$ ($\times10^4$)					
S=0.3	1.0	2.2	3.0	1.7	2.1
S=0.5	5.6	8.4	12	7.1	6.9
S=0.8	32	43	63	39	31
f_{NB} (%), 40KeV NBI, 2T, H	14.4	17.7	17.2	15.4	18.1

1. For kink unstable configurations, we give the eigenvalue followed by the most unstable mode in n/m, where n is toroidal mode number and m is the poloidal mode number.
2. For ballooning unstable configurations, we give the range of instability in s, the normalized toroidal flux, followed by the maximum eigenvalue in the unstable region.
3. Recent calculations with improved resolution indicate that these configurations may be unstable to localized kink-ballooning modes, such as m/n=11/17. See Chapter 5.

The coil designs reproduce the aspect ratio, rotational transform and beta of the reference plasma very well. Stability of the N=0 (periodicity-preserving) and N=1 families of external kink modes is preserved in the M3 design, but at the expense of quasi-symmetry.

The M8 and M7 designs explore avenues for improving the quasi-axisymmetry. The M8 design, in which the coil group currents were allowed to vary in the design process, has an improved effective ripple parameter and reduced neutral beam losses relative to the M3, while the stability properties are preserved. The plasma properties of the M7 design come close to those of the reference plasma configuration. Shape control at the outboard tip of the oblate cross section is important for maintaining a magnetic well, which in turn is important for stabilizing the pressure-driven kink mode, as explained in Chapter 5. Having a coil at the v=1/2 cross section is evidently beneficial for this; the benefit manifests itself in the form of better quasi-axisymmetry. In general, we find that the M7 gives the best reconstruction results. Although it would need to be modified to accommodate tangential access, the study shows the benefit of having the symmetry coil on the v=1/2 plane.

2.7.4 Summary

The performance of several modular coil designs has been analyzed in terms of the physics properties of the reconstructed plasmas. The coil design utilized for the initial engineering study (M3) has moderately good ripple characteristics, and the kink and ballooning modes can be stabilized. The freedom to independently vary the modular coil currents and the use of additional PF and TF coils are valuable design features in this regard. They allow us to find more MHD stable plasmas without losing beta or changing the aspect ratio at the full current state. Such modifications make configurations less quasi-symmetrical, with increased ripple along magnetic field lines, particularly inside 50% of the plasma radius. More optimized coil designs (M8) are able to reconstruct stable equilibria with favorable ripple and orbit confinement. There appears to be an advantage to placing a coil at the symmetry plane v=1/2, which results in a more MHD stable plasma. This will be further explored during conceptual design.

Further improvement may be possible by incorporating TF biasing in the design optimization process. Coupling physics optimization directly to the coil optimization could streamline the present design process and open a wider design space. Developing such a tool in the next phase of the NCSX project, thus, may lead us to even better coils.

2.8 Reference Scenarios

To implement coil physics requirements in the engineering design process, a set of reference scenarios is defined. These are idealized representations of real machine operation that establish a basic performance envelope for the coils and power systems. The machine capabilities so defined are then compared with the requirements derived from detailed physics evaluations of startup (Chapter 10) and flexibility and robustness (Chapter 9). Coil designs are modified as needed to meet these physics requirements. This iterative process leads to a machine design that is optimized to provide the required physics capabilities at minimum cost.

The reference scenarios consist of sequences of three equilibrium states with boundary shapes matching that of the reference high-beta plasma:

- S1 state: zero current, zero beta (i.e., vacuum). Provides large volume of closed vacuum magnetic surfaces for plasma initiation.
- S2 state: full current, zero beta (nominally end of current ramp, start of heating). Provides a hollow current profile setup by rapid current ramping before introducing plasma heating.
- S3 state: full current, full beta (reference high beta equilibrium). The target state.

The reference scenarios are defined by a sequence of phases with specified time intervals, as shown in Table 2-5. Machine design implications will be addressed in the next chapter, Chapter 3.

Table 2-5 Reference scenarios for initial coil design

	Units	Low Field Scenario	High Field Scenario
Maximum magnetic field at R=1.4 m	T	1.2	1.7
Maximum plasma current	kA	125	175
Initiate plasma	s	0.1	0.1
Ramp current to full current (S1 to S2 state)	s	0.041	0.058
Current ramp rate	MA/s	3	3
Heat to full beta (S2 to S3 state)	s	0.1	0.1
Maintain constant current, beta, and magnetic field	s	1.0	0.2

2.9 Electromagnetic Requirements

An important physics goal of the NCSX will be the demonstration that the NCSX target equilibrium is intrinsically stable to the external kink mode in the absence of a close fitting conducting wall. The NCSX is designed to minimize the influence of the vacuum vessel and other close fitting structure on the growth rate of the external kink. This can be achieved by keeping conducting structures beyond the critical plasma-wall separation, where the wall begins to have a significant effect on the mode, or by keeping the effective time constant for these structures short compared to plasma evolution time scales.

Design criteria are developed by performing stability calculations with the TERPSICORE 3-D ideal stability code. The reference plasma configuration, which is marginally kink stable at its design beta (~4%) without a wall, is used. To destabilize the kink, the beta is artificially increased. A conformal conducting wall is placed around and at some distance from the plasma. When the effective beta is increased by ~5%, the critical wall-plasma separation is reduced to about 5 cm, or $b/a \approx 1.13$. The NCSX vacuum vessel lies outside this layer over most of the surface, but is close to it on the inboard side, so it might be expected to modify the mode growth rate. However, the wall time constant for the vacuum vessel is much shorter than the pulse length. Thus, although the vacuum vessel may be close enough to affect the mode growth rate, it should not affect the ability to determine experimentally the intrinsic external kink stability of the NCSX plasma.

2.10 Power and Particle Handling

The plasma conditions needed to achieve the NCSX research goals require a level of plasma performance that may be achievable only with good control of neutrals and impurity influx. A power and particle handling program is planned to develop a heat removal and particle control capability consistent with the needed enhanced plasma performance. The goals for plasma-facing component design are to minimize the impact on plasma performance:

- Heat removal must be accomplished in a way that avoids excess temperatures on the material surfaces.
- Neutrals from recycling must be controlled so as to minimize their effect on plasma performance.
- Plasma-surface interactions must be designed for minimum impurity generation.

The design of stellarator power and particle handling hardware is strongly affected by the three-dimensional geometry and by the potentially complex topology of the magnetic field-line structure outside the last closed magnetic surface. Islands, ergodic regions with short and long field-line connection lengths, and ordered-layer structures are all possible. Hardware components, including limiters and divertors, can be localized both toroidally and poloidally, can follow the helical twist of the plasma for a distance, and can be contoured to meet specific performance objectives. For these reasons, the design of the optimum power and particle handling system will be a long-term program, based on the ongoing LHD and W7-AS experiments and on modeling studies now under way. The design of the plasma facing hardware will be iterated during operation to incorporate new experimental results and ideas.

2.10.1 Physics Modeling

Modeling studies that are under way or planned to support the physics design of NCSX power and particle handling systems, described in detail in Chapter 11, are summarized as follows:

- The magnetic field-line topology outside the last closed magnetic surface (LCMS) of NCSX plasmas are being studied with codes developed for Wendelstein 7-X and made available by stellarator researchers at Germany's Max Planck Institute for Plasma Physics. The codes have been modified to handle the large internal currents associated with quasi-axisymmetric stellarator plasmas. Field-line analyses for the NCSX modular coils exhibit an ordered-layer structure similar to that of an axisymmetric tokamak outside the LCMS.
- The spatial loss pattern of energetic ions from neutral beam injection is analyzed using the same Monte Carlo slowing-down model used to estimate beam heating efficiencies. The losses are found to be concentrated in helical stripes either above or below the midplane, depending on the ∇B drift direction.
- The heat flux profiles on plasma-facing component surfaces that intersect open field lines will be estimated using a technique that couples field line tracing with a model for transverse "diffusion" of the field lines. The field line diffusion simulates the effects of cross-field energy transport and has the effect of broadening the heat deposition profiles on target surfaces. These calculations will utilize codes provided by German colleagues, adapted for the NCSX application.
- Guidance on the placement of plasma-facing components for minimum neutral influx is being developed using the DEGAS 2 Monte Carlo neutral transport code. The volume distribution of neutrals inside the plasma is calculated for potential recycling sources. Initially, two-dimensional geometries based on axisymmetrized NCSX cross sections are being analyzed to obtain preliminary guidance.
- On a longer time scale, the project will move to more sophisticated edge-plasma transport models that self-consistently treat plasma-neutral interactions and couple to plasma edge turbulence models. These models could contribute to hardware upgrade decisions including the most effective location of baffles to control fueling, the type and position of edge-plasma diagnostics, and the level of impurities that can enter the core.

2.10.2 Physics Design of Power and Particle Handling System

The NCSX machine requires an initial system of power and particle handling hardware that supports plasma shape flexibility and experimental investigations of boundary physics issues. It will be upgradable to support the planned phased implementation of power and particle handling systems. The design must be robust against the physics uncertainties that exist in the distribution of heat and particle fluxes on the wall, although cost considerations limit the amount of area that can be covered with expensive material.

- Materials: the vacuum vessel material is Inconel and the plasma-facing components are made of graphite or carbon-fiber composite material. Because of the short heating pulse length, the allowable peak heat flux is quite high, up to 30 MW/m^2, and wall pumping can be used for particle removal.
- Configuration: The baseline design utilizes a contoured liner constructed of molded carbon fiber composite (CFC) panels mounted on a frame of poloidal rings. The molded panels form a continuous shell around the plasma with penetrations for diagnostics, heating, and personnel access. The continuous shell can meet the requirements of the plasma-facing surface, providing a high heat flux capability in the divertor region; providing an inboard belt limiter; providing a high heat flux surface for energetic ions from neutral beam injection; and providing armor for neutral beam shinethrough. During conceptual design, adding baffling in the divertor region will be investigated. The plan is to stage the installation of the liner, with limited wall coverage during initial operation, perhaps limited to covering the frame of poloidal rings with low-Z tiles.
- Vacuum pumping and wall conditioning: four 1500 l/s pumps will be provided using existing pumps. Base pressures of about $2\text{-}3\times10^{-8}$ torr are expected, based on previous experience with this same equipment. The graphite PFCs will be bakable to 350ºC to expedite the transport of hydrogen and impurity gasses out of the material. The vacuum vessel will be kept at 150ºC to facilitate diagnostic interfaces. Glow discharge cleaning and boronization will be provided for wall conditioning initially. Lithium deposition is of interest and the port space will be provided to accommodate various implementation options as upgrades.
- Fueling: A programmable multi-species gas injection system based on the existing PBX-M equipment will be provided. It will be capable of real-time feedback control based on a density measurement. High-field-side pellet injection may be added as an upgrade, to provide a tool for accessing enhanced confinement modes. The PBX-M pellet injector is available and any necessary guide tubes will be pre-installed in the device to ensure that high-field-side launch capability can be accommodated.

Further details about the design of power and particle handling systems and capabilities are provided in Chapter 11.

2.11 Plasma Heating

Auxiliary heating is required to achieve the plasma conditions (high $\langle\beta\rangle$ at collisionalities ν^* 0.25) needed to accomplish the NCSX research goals. A total of 12 MW can be accommodated by the device design, 6 MW of tangential neutral beam injection (NBI) and

6 MW of radio-frequency heating. Initially, the facility will be equipped with 3 MW of tangential NBI using two beamlines arranged in a balanced (1 co-, 1 counter-) configuration. The initial system provides balanced injection for discharge control (Chapter 10) and enough power to investigate enhanced confinement regimes, assess confinement scaling, and acquire a database for deciding on the amount and type of any plasma heating upgrades that might be needed. Detailed analyses of NCSX plasma heating capabilities and options are explained in Chapter 7.

Neutral beams are attractive because they have been successful over many decades as a tool for confinement physics experiments, particularly those addressing high-beta issues. They are flexible because they provide effective coupling and heat deposition over a wide range of plasma configurations and parameters. The availability of the existing 6 MW NBI system from PBX-M, developed in the 1970s by Oak Ridge National Laboratory and consisting of four beamlines and the associated controls and power systems, presents a cost-effective design solution. They have a maximum beam energy of 50 keV and a pulse length of 0.3 s, and can inject hydrogen or deuterium. The NBI system supports plasma startup by heating the plasma. Tangential injection and hydrogen neutrals are used to reduce beam ion losses. A balanced configuration is chosen to provide the control capability required to null the beam-driven currents, as explained in Chapter 10. The neutral beams provide a range of useful physics capabilities, such as ion heating, current drive (for flexibility), pressure profile control, and ExB shear control. Increasing the pulse length beyond 0.3 s is of interest, as explained in Sect. 2.1, to study fully equilibrated current profiles. It is likely that the pulse lengths can be extended to 0.5 s with no modification and little power degradation. The MAST program in the United Kingdom uses the same type of ORNL beamlines and is developing technology to extend their pulse lengths to >1 s. The NCSX program can take advantage of those developments to extend its neutral beam pulse lengths in the future.

The RF heating system for NCSX will be designed to deliver 6 MW of radio frequency (RF) power to heat the plasma. If current drive were desired, the available heating techniques could provide it with minor modifications to the system. Two options are available, high frequency fast wave (HFFW) and mode conversion heating.

High frequency fast wave heating utilizes fast magnetosonic waves at high harmonics of the ion cyclotron frequency, which minimizes ion damping while producing strong damping on the electron population. The operating frequency chosen for NCSX is 350 MHz. This choice of frequency dictates the use of klystrons for power sources, and folded-waveguide launchers for coupling RF to the plasma. The attractions of HFFW heating include insensitivity to the magnetic field, strong damping, and no significant damping on the neutral beam ions.

In ion Bernstein wave mode conversion heating, a fast magnetosonic wave, excited at the boundary of a multiple-ion species plasma, propagates to the ion-ion hybrid layer where it undergoes conversion to the slow wave. Damping can be on electrons or ions, depending on the ion temperature and species mix. Modeling of NCSX plasmas has indicated that a high field side fast wave launch is necessary to efficiently access the mode conversion surface. The modular coil design permits installation of a high field side “combline” antenna. This type of antenna can be constructed with a very small radial build, which lends itself to installation in a shallow "pocket" in the vacuum vessel, on the high field side.

Electron cyclotron heating has not been analyzed in detail, but because of its modest port requirements, it is expected that up to 2 MW can be accommodated as a future upgrade, if desired.

2.12 Diagnostics

A capable array of diagnostics is planned to make the plasma physics measurements necessary to accomplish program goals. The facility will be equipped at first with the minimum set of diagnostics needed to support shakedown of major machine systems and the first few phases of physics operation, including first-plasma, electron-beam mapping of flux surfaces, Ohmic plasma experiments, and initial heating experiments. It is expected that an expanded diagnostic system will be necessary to achieve the full range of NCSX physics goals, and that these will be added as upgrades during the operating life of the facility. Experimental results from the initial operating phases will help optimize the selection of upgrade systems and their design characteristics. Nonetheless, an implementation plan for upgrade diagnostics has been developed for use as a reference for design purposes. It is used in the ongoing design process to set port access requirements and ensure that a feasible solution exists for all required measurements. The diagnostic program is described in greater detail in Chapter 12.

2.12.1 Baseline Diagnostics

The initial complement of diagnostics includes the following:

- Magnetic sensors: a diamagnetic loop and arrays of flux loops, Rogowski loops, and B-field probes will provide signals to determine the plasma magnetic configuration with the aid of an equilibrium reconstruction code. Because of the strong shaping in NCSX plasmas, such a magnetic reconstruction can provide important information on profiles of plasma pressure and toroidal current.
- Visible cameras: tangential views of the plasma will provide a convenient monitor of the plasma discharge evolution and plasma-wall interactions.

- Interferometer: a single line of sight will provide a monitor of the line density through the core of the plasma during the initial plasma run.
- Vacuum ultraviolet survey spectrometer: This instrument will survey impurity emission from the VUV part of the spectrum, along a single sightline. This diagnostic, as well as the visible spectroscopic diagnostics listed below, are needed to look for signs of impurity contamination problems. This is a common concern in the shakedown phase of any device, when plasma control may be poor and internal hardware problems can be expected. Either can lead to unplanned plasma-material contact, resulting in impurity influx and an increase in radiated power. Detecting such radiation, most of which is in the vacuum ultraviolet, can aid in diagnosing and correcting such problems.
- Visible survey spectrometer: This instrument will monitor the plasma along several lines of sight. At one location, a special detector system will be provided, capable of monitoring selected impurity and hydrogen/deuterium recycling lines with high time resolution.
- Soft x-ray imaging array: This system will measure x-ray emissivity along a fan array of sightlines to detect internal magnetohydrodynamic (MHD) activity. Since such activity can greatly influence the behavior of the plasma particularly as plasma startup control is first being established, these measurements will be valuable in establishing good plasma control in the initial Ohmic experimental phase. The same array can provide other data useful for early NCSX operation, besides MHD. Significant bounds can be placed on the electron temperature; this is especially valuable since, at the low magnetic fields and high densities expected for early NCSX operation, electron-cyclotron emission measurements of electron temperature may be precluded. Finally, the soft x-ray array may be able to support rough measurements of the total radiated power profiles.
- Electron-beam field mapping apparatus: An electron gun and a fluorescent screen will be provided to investigate the magnetic flux surface quality of the device as constructed and to test the effects of coil-current variations on the magnetic configuration geometry.

2.12.2 Upgrade Diagnostics

After initial shakedown of the NCSX equipment is completed and control of the plasma becomes more routine, it is expected that attention will shift toward the research goals and progressively more diagnostic capability will be required. Diagnostic emphasis will shift from basic monitoring of global quantities and impurities to local measurements of plasma parameters in the core and edge and, later, to more detailed profile measurements. Diagnostics for measuring MHD activity, fast ion behavior, edge and divertor characteristics, and turbulence will see a steady improvement in capability.

Among the upgrade diagnostics, beam-based spectroscopy diagnostics (known as CHERS and MSE) will be used to obtain profile information on ion temperature, rotation velocity, magnetic field pitch angle, radial electric field, and current density. A multi-pulse Thomson scattering system will provide multiple snap shots, during a pulse, of the electron temperature and electron density profiles along a laser beam. Some of the diagnostics upgrades will be augmentations of the initial systems. For example, magnetic sensors will be added or improved through the life of the project, as experience is gained and as new physics or control needs arise. Multiple sets of soft x-ray arrays for monitoring MHD activity, and bolometer arrays to measure radiated power are similarly staged in the plan.

2.12.3 Diagnostic Access

During conceptual design, the project emphasis will be on the design of the stellarator core device, including the coils, vacuum vessel, and plasma-facing components. A systematic analysis of the access constraints and tradeoffs for both baseline and upgrade diagnostics will be an important element of the conceptual design activity. Diagnostic considerations have played a prominent role in the pre-conceptual design development of the NCSX. As a result, a large number of ports (about 87) is provided and many specific diagnostic needs have been taken into account, however the task of providing adequate diagnostic access has only begun. At this stage, the available access is marginal, and providing good diagnostic access will be an important design goal for the conceptual design phase.

2.13 Research Plan, Operation Requirements

The research plan for NCSX will be carried out in a sequence of experimental campaigns:

- First plasma operation: Short Ohmic pulses will be used to achieve the first-plasma milestone and carry out a brief campaign intended to test the ability to initiate and control the plasma and the operation of the initial diagnostics.
- Field-line mapping: This campaign will test the accuracy of the stellarator magnetic field generation by measuring properties of the magnetic surface configuration in vacuum.
- Initial Ohmic experiments: This campaign will establish good control of the magnetic configuration as well as good vacuum and wall conditions. Physics results on global confinement scaling, density limits, vertical stability, effects of low-order rational surfaces on stability and disruptions, and plasma-wall interactions, all at low beta, will be produced.

- Initial plasma heating and transport experiments. This campaign will explore the flexibility, plasma confinement, and stability of the stellarator experiment, starting at the initial heating power (3 MW from two neutral beams), magnetic field (at least 1.2 T) and pulse length (at least 0.3 s). The ability to control the discharge evolution to produce current profiles approximating the bootstrap profile will be tested. Physics results on the adequacy of neoclassical transport optimization, density limits, confinement scaling, and enhanced confinement regimes will be produced. Conditions for avoiding density-limit disruptions will be investigated. Boundary plasma conditions and plasma-wall interactions will be studied. This campaign will develop a database for deciding on the amount and type of any plasma heating upgrades that might be needed and for next steps in the implementation of plasma-facing components. In addition, this campaign will likely commission new diagnostics systems.
- Confinement optimization and increasing beta: This campaign will attempt to extend enhanced confinement regimes and investigate high-beta stability issues with a full neutral-beam complement (6 MW from four beams) and/or megawatt-level radio--frequency heating. Enhanced confinement will be pursued using the techniques developed on tokamak experiments, including sheared rotation from NBI, reduced recycling by wall coating (B, Li) and conditioning, by edge radiation (RI-mode), and by pellet fueling. The dimensional and non-dimensional scaling of confinement will be determined and compared to other configurations. These plasmas will then be used to test directly the predicted beta-limit and the predicted beta-limiting mechanisms. The configuration requirements to avoid disruptions and the disruption-free operating area at high beta will be documented.
- Long-pulse upgrade: This campaign will be preceded by an upgrade to the heating systems (to allow pulse lengths of ~1 sec, and power of as much as 12 MW) and a possible upgrade of the plasma-facing components for improved power and particle exhaust handling for long pulse. These upgrades will allow equilibration of the current profile to the bootstrap current, and will be used to document the high-beta disruption-free operating area in long-pulse operation (compared to the current-profile relaxation time).

These campaigns could be carried out in as little as four years with adequate operating time, diagnostics, and needed facility upgrades. All NCSX research goals could be accomplished in that time. The facility is designed for a 10-year operating life, which is typical for experimental facilities of this class. The design provides for a maximum number of pulses, based on operation in the reference scenario, of 100 per day, 13,000 per year, and 130,000 over the life of the experiment. The device and beams can be operated either in hydrogen or deuterium, with deuterium operation administratively limited by biological shielding capability as on PBX-M.

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