Physics Considerations in the Design of NCSX*

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Abstract. Compact stellarators have to potential to make a steady-state, disruption-free magnetic fusion system with β~5% and relatively low aspect ratio (<4.5) in stellarator terms. Magnetic quasi-symmetry is used to reduce losses. The National Compact Stellarator Experiment is designed to produce a quasi-axisymmetric target plasma with low ripple, good magnetic surfaces, and stability to the important ideal modes at β~4%. The device size, available heating power, and pulse lengths provide access to a high-beta target equilibrium condition. To support its physics mission the NCSX has magnetic flexibility to explore a wide range of equilibrium conditions and operational flexibility to achieve a wide range of collisionalities and betas. Space is provided to accommodate a sequence of first-wall configurations including baffles and pumps for divertor operation. Port access is provided for the extensive array of diagnostics that will support the planned experimental program.

1. Introduction

Fusion energy research is increasingly focused on the problem of finding the best magnetic configuration for a practical fusion reactor. Stellarators are of particular interest because they can solve two major problems for magnetic confinement– achieving steady state operation and avoiding disruptions. Consequently substantial investments have been made in new stellarator facilities, including the large superconducting LHD [1] and W7-X [2] experiments. The three-dimensional plasma geometry of stellarators provide degrees of freedom, not available in axisymmetric configurations, to target favorable physics properties (low magnetic field ripple, well-confined particle orbits, high-beta stability without the need for current drive or feedback, and good magnetic surfaces) in their design. Design features now being studied experimentally include the use of large helical magnetic-axis excursions (TJ-II [3], H1-NF [4], and Heliotron-J [5]) and approximate alignment of particle drift orbits with magnetic surfaces (W7-AS [6] and W7-X [2]).

In compact stellarator design the 3D magnetic field has an approximate symmetry direction (quasi-symmetry) to keep charged particle drift orbits well confined, the aspect ratio $R/\langle a\rangle$ is much less than that of currentless optimized stellarators (≤4.5 *vs* ≥10), and the bootstrap current can be used to generate some of the rotational transform (iota). Stellarator coils generate the remainder of the rotational transform and shape the plasma to achieve desired physics properties.

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The National Compact Stellarator Experiment (NCSX) [7] is designed as a quasi-axisymmetric stellarator (QAS, also examined in the CHS-qa study [8]), which can share advantageous properties with tokamaks, including high beta (>4%) in moderate aspect ratio ($R/\langle a \rangle \approx 4.4$ or less) geometries and turbulence stabilization with sheared flows. A successful experimental test of magnetic quasi-symmetry has already been carried out in the Helically Symmetric Experiment (HSX) [9]. The QPS experiment [10] is being designed to study a quasi-poloidal, very low aspect ratio compact stellarator configuration. The mission of NCSX is to test compact stellarator physics in a high-beta QAS configuration and to determine the conditions for high-beta disruption-free operation in order to evaluate its potential as a fusion concept.

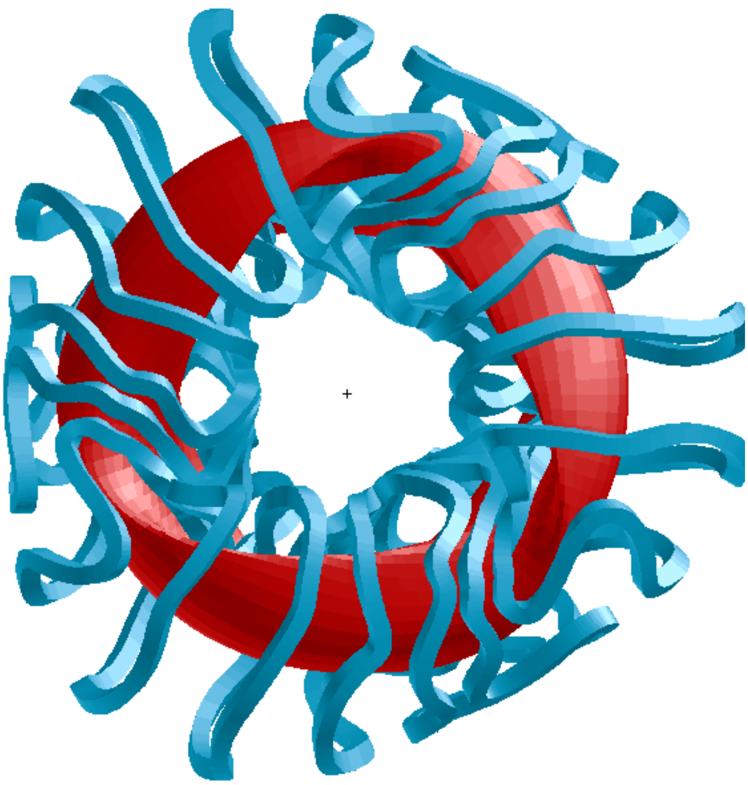

FIG. 1. NCSX modular coils and reference plasma. The coil system also includes toroidal, poloidal, and helical-field trim coils, not shown in the figure.

2. Coil Design

The NCSX uses modular coils (Fig. 1) to provide the main helical magnetic field. For equilibrium flexibility, there are also toroidal field coils, poloidal field coils, and helical-field trim coils. The physics basis for the coil design is a computed fixed-boundary three-period QAS reference plasma configuration [11] with β = 4%, assuming a moderately broad pressure profile typical of experiments and a consistent bootstrap current profile which generates

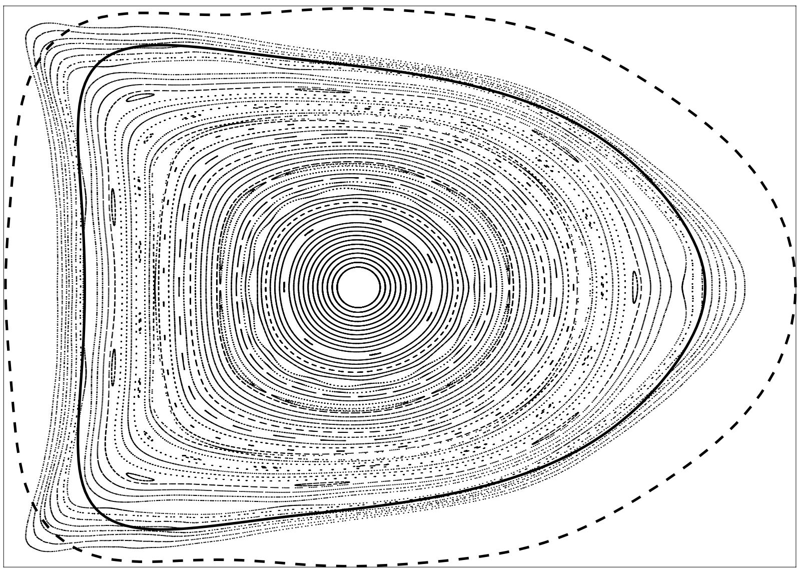

FIG. 2. Poincaré plots of PIES magnetic surfaces for target free-boundary PIES equilibrium at β=4.1% for finite-cross section model of NCSX coils. Dashed line: first wall; solid line: VMEC equilibrium boundary.

TABLE I. NCSX Configuration Physics Design Summary

Parameter or Property	**Achieved in β = 4% Target Equilibrium**	**Criteria**
Aspect ratio $R/\langle a \rangle$	≤ 4.4	Substantially lower than existing drift-optimized stellarator designs, e.g. HSX, W7-X.
Stability at $\langle\beta\rangle$ = 4%	Stable to ext. kink, vertical, Mercier modes	Sufficient to test stabilization of a sustainable toroidal plasma by 3D shaping.
Stellarator shear	iota = 0.39 (center), 0.65 (edge)	For neoclassical island reduction and healing. Monotonically increasing except very near the edge.
Large external iota fraction.	~0.75 from coils	Conservative for disruption-resistance.
Quasi-symmetry	effective ripple $\varepsilon_h \approx$ 0.1% (center), 0.4% ($r/a \approx 0.7$).	Low helical ripple neoclassical transport compared to axisymmetric; tolerable balanced neutral-beam-injected ion losses in high-beta scenarios.
Magnetic surface quality	total effective island width <10% of toroidal flux	For negligible contribution to losses. Based on PIES equilibrium calculations and estimated neoclassical and finite-$\chi_\perp\chi_\parallel$ corrections.

about one-fourth of the rotational transform at the edge.

Existing stellarator coil design methodologies were extended in order to design feasible coils for low-aspect-ratio plasma configurations with internal currents. An initial, filamentary modular coil solution is found by minimizing the root-mean-squared normal component of the magnetic field on the surface of the reference plasma, following the “reverse engineering” approach used in the W7-X design [12]. The coil geometry is then modified by a technique which couples the coil [13] and plasma [14] optimization processes to obtain coils which produce a free-boundary target equilibrium with physics properties of the reference plasma, rather than its shape, as well as engineering parameters such as coil spacing and curvature. The achieved physics properties are summarized in Table I. An effective ripple parameter ε_h, matched to tranport in the $1//\nu$ regime [15], measures the degree of quasi-axisymmetry. In the final step, the coil geometry is modified again using a technique to reduce the width of residual islands in the free-boundary equilibrium as calculated by the PIES code [16]. The coil design process leads to a free-boundary high-beta target equilibrium that has. good magnetic surfaces (Fig. 2) while preserving other physical and engineering properties.

Although their design is optimized for a single equilibrium, the NCSX coils can support a broad range of equilibria with favorable physics properties, as is critical for its physics mission. This is done by varying the currents in the toroidal field, poloidal field, and modular coil circuits. Since it is planned to initiate NCSX plasmas on vacuum magnetic surfaces, the existence of good vacuum configurations is critical. The nominal case with iota < 0.5 everywhere, is shown in Fig. 3. Good configurations with iota > 0.5 are also available, providing flexibility to either encounter or avoid tearing modes associated with the iota = 0.5 resonance during startup.

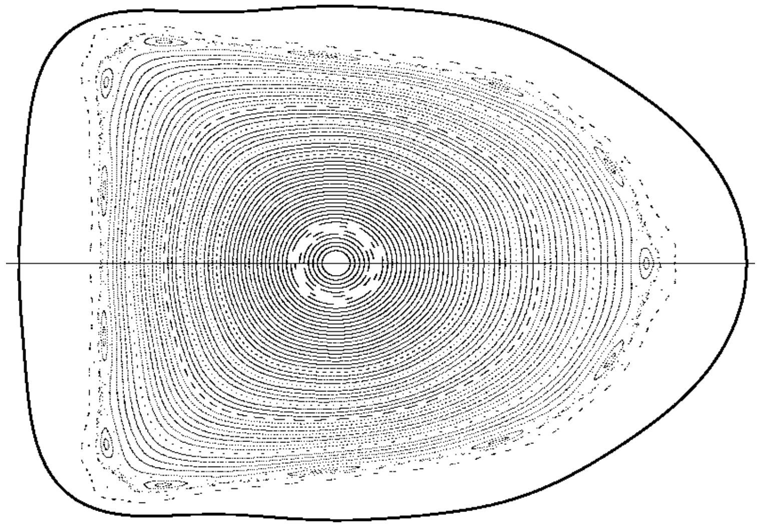

FIG. 3. Poincaré plots of PIES magnetic surfaces for vacuum configuration with iota of 0.43-0.46 (filamentary coil model). Dashed line: first wall boundary.

The coils provide a wide operating space in β and plasma current (I_P) for the reference pressure and current profile shapes and constant magnetic field (B = 1.7 T) at R = 1.4 m. As shown in Fig. 4, VMEC equilibria stable to external kink and ballooning modes and having low ripple ε_h can be made with β ranging from 0 - 4% and I_P from 0 to 100% of its reference value. Stable equilibria at higher beta (at least 6%) can be made with modest increase in ripple. The coils are also robust to variations in pressure and current profile shapes; while the reference current profile is hollow, stable equilibria with β = 3% and ε_h < 0.5% are also found with peaked profiles.

		Beta (%)						
		0	1	2	3	4	5	6
Current (kA)	0	0.82%	0.89%	0.79%				
	44	0.77%	0.68%	0.67%	0.61%	0.72%		
	88	0.71%	0.65%	0.51%	0.72%	0.60%		
	131	0.52%	0.46%	0.42%	0.41%	0.45%		
	174	0.37%	0.39%	0.36%	0.40%	0.45%	0.92%	

FIG. 4. Effective ripple (ε_h) at r/a≈0.7 for equilibrium conditions in plasma current-beta space for NCSX coils and reference profiles. Equilibria inside heavy boundary are stable to external kink and ballooning modes. Shaded equilibria are unstable.

The coils can vary equilibrium parameters to the physics properties are sensitive, providing the capability to test theoretical predictions. By reshaping the plasma, kink stability beta limits can be lowered from the nominal 4% to about 1% so stability limits to be studied over a range of beta values. While the design has been optimized to make the effective ripple at r/a≈0.5 very low, it can easily be increased by almost an order of magnitude, while maintaining stability, to test the dependence of neoclassical losses and confinement enhancement on the degree of quasi-symmetry. The rotational transform profile can be varied to study its effects on transport and stability. The external rotational transform can be varied at fixed shear from –0.2 to +0.1 about the reference profile. The global shear (ι_{edge} - ι_0) can be increased by a factor of 2 at fixed ι_0; limits on reducing the shear are still being studied.

Satisfactory magnetic surfaces have been calculated with PIES for several points in the NCSX operating space, but additional island width control may be needed. Planned resonant trim-coils can be added to provide additional control of island-producing resonances to reduce island widths or perform controlled island physics experiments.

3. Device Size and Performance

The NCSX device size (major radius R = 1.42 m), magnetic field range (B = 1.2-2.0 Tesla), pulse length (0.3-1.1 s) and plasma heating power (initially 3 MW) are set to produce the plasma conditions and profiles needed to test critical physics issues over a range of betas collisionalities. The device will be initially equipped with two (of four) existing 1.5-MW, 50-keV, 0.3 s neutral beam injectors, formerly used on the PBX-M experiment, arranged for balanced tangential injection to be able to balance rotational transform perturbations due to beam-driven currents. Monte Carlo beam slowing down calculations predict 24% hydrogen beam ion losses for balanced inection at at B = 1.2 T. With the initial 3 MW beam system, and assuming an enhancement factor of 2.9 times the ISS95 [17] stellarator confinement time scaling, or 0.9 times the ITER-97P L-mode tokamak scaling [18], plasmas with $\beta = 2.6\%$ and collisionality $\nu^*=0.25$ are predicted. With the full complement of hydrogen neutral beams (6 MW) and these same collisionality and confinement enhancement assumptions, the reference beta value of 4% is predicted. The required ISS95 enhancement factor is reduced to 1.8 by allowing the density to rise to the Sudo limit [19], with an attendant increase in collisionality.

The NCSX magnet system is designed for pulsed operating scenarios with magnetic fields up to 2.0 T for 0.2 s for low-collisionalty plasma studies. The magnet design also provides operating scenarios with plasma current up to 350 kA (providing a factor of 2 range for internal rotational transform flexibility) and pulse lengths up to 1.2 s (at B = 1.2 T) for experiments with pulse lengths long compared to current equilibration times. The neutral beam pulse length can be increased from 0.3 s to 0.5 s with modest changes and potentially to >1 s with technology improvements being developed by Culham Laboratory. Radiofrequency waves can be launched from the high-field side for long-pulse heating or to more directly heat electrons than with the neutral beams.

4. Discharge Evolution

Time-dependent modeling of NCSX discharge evolution has been carried out to demonstrate the existence of satisfactory paths from vacuum fields to a target high-beta state, consistent with the technical capabilities of the coils and heating systems. The coils provide Ohmic heating, current drive, external rotational transform, and plasma shape control. Neutral beams provide plasma heating and current drive, although the balance of co- and counter-injection is adjusted to minimize the latter. The bootstrap current provides significant internal rotational transform.

A simulation methodology based on the TRANSP code, a well-developed 1-1/2-D tokamak analysis code, was developed to model the time-evolution of the poloidal flux and iota profile. Because NCSX is quasi-axisymmetric, tokamak modeling tools can provide an accurate guide. An equivalent tokamak equilibrium is generated having the same external iota, major radius, aspect ratio, plasma volume, and toroidal flux as the reference NCSX equilibrium. In the simulation, the plasma shape, external iota profile (simulated as an externally-specified driven current profile), and density profile shape are kept constant in time. The plasma current, density, neutral beam heating power, and co-/counter- beam balance are programmed. The electron and ion thermal diffusivities are automatically adjusted to match an assumed global energy confinement scaling (minimum of neo-Alcator and ITER-97P L-mode scaling). The TRANSP code models poloidal flux diffusion, beam deposition and slowing down,

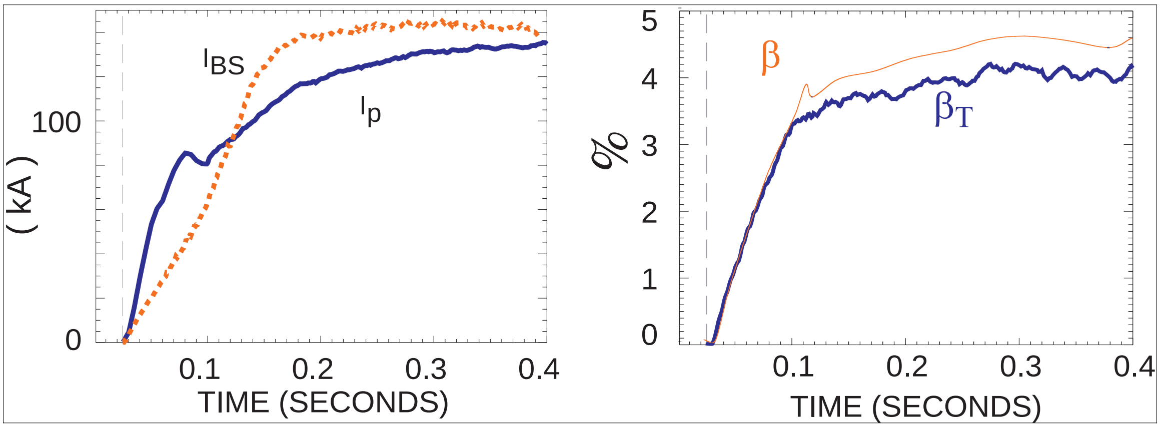

FIG. 5. Waveforms from time-dependent simulation of high-beta NCSX discharge. (a) Plasma current (programmed) and calculated bootstrap current. (b) Calculated beta values using tokamak (β_T) and stellarator (β) definitions.

neutral beam current drive, and power balance; and calculates the profiles of electron and ion temperature and pressure, fast ion pressure, current, and iota. The transformation to the actual NCSX geometry is accomplished by using the free-boundary VMEC code to calculate a time-series of equilibria with the actual NCSX coil geometry and the TRANSP-simulated profiles as input.

Results are presented for a simulation with B = 1.4 T and an available 6 MW of balanced neutral beam injection. Figure 5 shows time sequences of key input and output parameters of the 1-1/2-D modeling. Important aspects of discharge programming to obtain the target beta and a nearly stationary bootstrap current profile at the end of the discharge were: minimizing the Ohmic current during startup, adjusting the co-counter beam balance to approximately balance the Ohmic current profile, rapidly (~1.5 MA/s) increasing the current initially followed by clamping of the applied loop voltage, and modulating the neutral beam heating power to control the total pressure. At t = 303 ms of the TRANSP simulation, the average toroidal beta has risen to about 4% (corresponding to 4.5% when transformed to a stellarator equilibrium because of a different magnetic field normalization), peak electron and ion temperatures are 2.4 keV and 2.8 keV respectively, and the volume-averaged electron and ion collisionalities are ν_e*=0.2, ν_i*=0.1. The current (130 kA) is well matched to the bootstrap current (140 ms), and the current profile is found to be predominantly bootstrap. Converged VMEC equilibria were found for the sequence of profiles resulting from this simulation and the NCSX coils. These equilibria were found to be stable to ballooning, kink, and vertical modes and to have good quasi-axisymmetry (ε_h < 0.4% at r/a≈0.5) throughout the discharge.

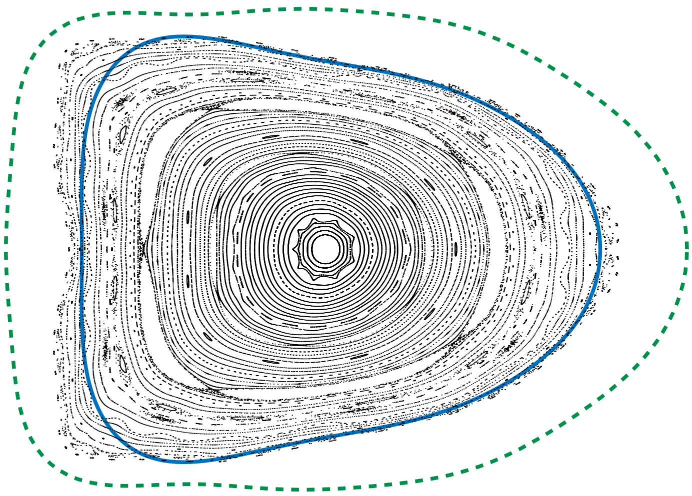

FIG. 6. Poincaré plots of PIES magnetic surfaces for equilibrium at t = 303 ms in discharge simulation.

Magnetic surface quality was analyzed using the PIES code for several time slices in this discharge. Results at t = 303 ms, when 99% of the current is bootstrap, are shown in Fig. 6. The m=5 island at t=303 ms, which extends over 6.3% of the cross-section, is predicted to be reduced to 3% by neoclassical effects which are not included in the PIES calculations. A number of small island chains are visible in Figure 5. When the island width is smaller than a critical value, diffusion across the island dominates diffusion along the field line, and the presence of the island has little effect on transport. The critical island width is about 2%-3% of the minor radius for the mode numbers of interest.

5. Power and Particle Handling

Control of impurities and neutral recycling is the main power and particle handling issue in the design of NCSX. For impurity control, low-Z materials (carbon) are planned for surfaces with high plasma-wall interactions. For neutral control, recycling sources and baffles will be arranged so as to inhibit neutral flow to the main plasma. Motivated in part by recent W7-AS divertor results showing improved edge control and plasma performance [20], the NCSX is designed so that a pumped slot divertor can be installed in the future. The basic requirement affecting the coils and vacuum vessel is providing sufficient connection length of field lines in the scrapeoff layer outside the last closed magnetic surface. Connection lengths longer than 100 m are sufficient to allow high separatrix temperatures and significant temperature drops along field lines to reasonably low target temperatures, and hence the establishment of a high recycling regime with a low impurity source at the target.

Scrapeoff layer field-line following calculations for magnetic configurations which have finite plasma pressure are made using the MFBE code [21]. In NCSX the field lines launched close to the last closed magnetic surface (here taken to be the VMEC boundary) make many toroidal revolutions close to it and do not exhibit very strong stochasticity. This is seen in Fig. 7, a Poincaré plot of scrapeoff field lines for the reference high-beta state and the NCSX coils. The field lines are launched from the midplane, half from the inside and half from the outside, from 0 to 10 mm outside the LCMS. Most lines remain with a 4-cm wide band conformal to the boundary (except near the tips of the banana-shaped cross sections, where the divertor would be located) long enough to complete 20 toroidal revolutions (~200 m) and provide the required connection lengths. These calculations are used as a guide in the placement of plasma-facing components with sufficient clearance to provide long connection lengths.

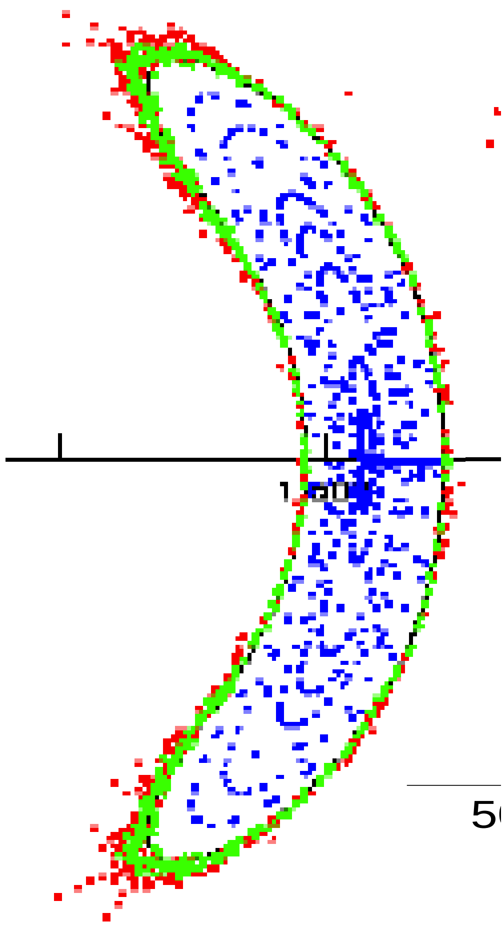

FIG. 7 Poincaré plots of scrapeoff field lines started at the inboard and outboard midplane within 0-1 cm of the nominal (VMEC) boundary and followed for 20 toroidal revolutions.

6. Experimental Plan and Diagnostics

The experimental goals of NCSX will be accomplished in a sequence of experimental phases. The needed diagnostics, which will be implemented as needed, are planned within this framework.

First plasma operation: Short Ohmic pulses will be used to test the ability to initiate the plasma and diagnose it with magnetic sensor arrays, fast visible cameras, and a 1 mm interferometer.

Field-line mapping: Magnetic surface properties will be measured in vacuum using a scanning electron beam source, scanning fluorescent probe, and a CCD detector. The probe is to be deployable without breaking vacuum and aims at a 2 mm spatial resolution.

Ohmic experiments: Ohmically heated discharges will be used to establish good discharge control and begin physics studies of global confinement scaling, density limits, vertical stability, effects of low-order rational surfaces on stability and disruptions, and plasma-wall interactions. A high-throughput, repetitively pulsed Nd:YAG Thomson scattering system, a soft x-ray tomography system, bolometer arrays, and visible and ultraviolet spectroscopic diagnostics will be added.

Auxiliary heating, confinement optimization, high beta, and long pulse:. The main NCSX research program will use auxiliary heating, starting at the initial power level of 3 MW. The design anticipates that the configuration of heating, fueling, control, diagnostic, and plasma-facing component systems will evolve as dictated by experimental results. Charge-exchange recombination spectroscopy and motional Stark effect measurements are key diagnostics which will be added; a diagnostic beam is planned to support these measurements. The full complement of diagnostics required for this program has been considered in the design of the NCSX. [22]. A large number of ports (96) are provided but the task of optimizing the diagnostic arrangement and port configuration for good access is in progress..

7. Summary

The NCSX is designed with the capabilities needed to assess the physics of compact stellarators. It will produce high beta equilibria with good physics properties and magnetic surfaces, have an ample operating space and flexibility, and provide access to high-beta target equilibria starting from vacuum. It provides the space and access provisions to augment the initial configuration with new capabilities, particularly diagnostics and plasma-facing components, to meet research needs.

[1] LHD reference.
[2] W7-X reference.
[3] TJ-II reference
[4] H1-NF reference
[5] Heliotron-J reference.
[6] W7-AS design reference.
[7] past NCSX reference(s)
[8] CHS-qa reference.

[9] HSX reference for results that symmetry matters.
[10] QPS reference.
[11] reference for LI-383?
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