Physics Issues in the Design of High-Beta, Low-Aspect-Ratio Stellarator Experiments

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High-beta, low-aspect-ratio ("compact") stellarators are promising solutions to the problem of developing a magnetic plasma configuration for magnetic fusion power plants that can be sustained in steady-state without disrupting. These concepts combine features of stellarators and advanced tokamaks and have aspect ratios similar to those of tokamaks (2-4). They are based on computed plasma configurations that are shaped in three dimensions to provide desired stability and transport properties. Experiments are planned as part of a program to develop this concept. A β = 4% quasi-axisymmetric plasma configuration has been evaluated for the National Compact Stellarator Experiment (NCSX). It has a substantial bootstrap

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current and is shaped to stabilize ballooning, external kink, vertical, and neoclassical tearing modes without feedback or close-fitting conductors. Quasi-omnigeneous plasma configurations stable to ballooning modes at β = 4% have been evaluated for the Quasi-Omnigeneous Stellarator (QOS) experiment. These equilibria have relatively low bootstrap currents and are insensitive to changes in beta. Coil configurations have been calculated that reconstruct these plasma configurations, preserving their important physics properties. Theory- and experiment-based confinement analyses are used to evaluate the technical capabilities needed to reach target plasma conditions. The physics basis for these complementary experiments is described.

I. Introduction

A critical issue for magnetic fusion energy (MFE) is the need to develop a high-beta magnetic plasma configuration that can be sustained in steady-state without disrupting. Interest in steady-state magnetic confinement systems is motivated by their economic benefits for MFE power plants: continuous power output without the need for energy storage, and good reliability through minimizing the thermal cycling of plasma-facing components. However, there are challenging requirements. The core plasma must have good plasma energy confinement and require recirculation of only a small fraction of the plant’s fusion power output to sustain the plasma. For this reason the use of inefficient means of sustainment such as non-inductive current drive must be avoided to the extent possible. The frequency of unscheduled plasma terminations (e.g., due to disruptions) must be minimized, and their effects mitigated, to avoid long unscheduled outages. High plasma beta (5% or more) and low aspect ratio (4 or less) advance the goal of making fusion devices as economically attractive as possible while reducing development costs.

Currently the major steady-state MFE research thrusts are the stellarator and the advanced tokamak (AT). The three-dimensional conventional stellarator relies entirely on magnetic fields generated by coils to sustain it. This eliminates the need for plasma current drive, but care must be taken in the configuration design to avoid self-generated currents. These designs have large plasma aspect ratios (5-12) and project to conservative power plant designs with relatively low power density. The axisymmetric AT uses the bootstrap current to sustain a configuration with lower aspect ratio (≤4) and higher power density than the stellarator, using plasma profile control, conducting structures close to the plasma, and feedback control of unstable modes to avoid disruptions.

High-beta, low-aspect-ratio stellarators, or "compact stellarators" (CS), offer a promising alternative solution that combines features of stellarators and advanced tokamaks. The CS uses the self-generated bootstrap current to sustain a configuration with AT-like plasma aspect ratio and beta values. The CS uses three-dimensional stellarator magnetic fields from coils to provide some of the confining poloidal magnetic flux and to stabilize the plasma by three-dimensional shaping without the need for active plasma controls or conducting structures close to the plasma. Theoretical studies, aided by sophisticated numerical analyses, have established basic physical characteristics of two CS plasma optimization approaches [1], the quasi-axisymmetric (QA) stellarator and the quasi-omnigeneous (QO) stellarator, each having a set of advantages. Experi-ments are needed to develop the physics of these complementary approaches and clarify their relative merits, and the necessary facilities are now being designed. The National Compact Stellarator Experiment (NCSX) will demonstrate disruption-free operation near beta limits and test the physics of the QA approach, while the smaller Quasi-Omnigeneous Stellarator (QOS) will test QO physics properties at lower beta values.

II. NCSX Physics Basis

Quasi-axisymmetric stellarators [2,3] possess tokamak-like magnetic symmetry as experienced by charged particles in the system. Neoclassical transport and energetic-particle losses are reduced by reducing the non-axisymmetric components of ripple in the magnetic spectrum. The bootstrap current magnitude is comparable to that in an advanced tokamak. The NCSX will support experiments aimed at demonstrating disruption-free operation near beta limits and understanding the physics properties of QA stellarators: beta limits and limiting mechanisms, equilibrium islands and neoclassical tearing-mode stabilization, reduction of neoclassical transport by QA configuration design, reduction of anomalous transport by flow-shear control, and compatibility with stellarator power and particle exhaust methods. The design is optimized around a reference high-beta QA plasma configuration. The machine and plasma heating system are able to generate the reference plasma and produce the required equilibrium fields accurately enough to preserve its important physical properties.

Here we describe the physics of a design for NCSX which re-uses existing major components (toroidal and poloidal field coils and neutral-beam injectors) from the former PBX-M [4] tokamak facility, has a major radius $R_0 = 1.45$ m, an average plasma radius $a = 0.42$ m, and a nominal magnetic field strength $B_0 = 1.2$ T.

A. Quasi-axisymmetric Plasma Configurations

This paper will focus on the physics properties of a reference QA plasma configuration [5] shown in Fig. 1. The configuration has three periods and an aspect ratio (3.4) compatible with the geometry of the existing TF coil set. A reference pressure profile with β = 4% (volume-averaged) and a bootstrap-like current profile are assumed in generating this configuration. The boundary shape has been chosen to provide a substantial externally generated rotational transform (>50% of the total transform at the plasma edge) with sufficient shear that the total rotational transform (ɨ, or 1/q) profile is monotonically increasing. This gives good ballooning and kink stability properties [6,7], and provides adequate quasi-axisymmetry for the assumed profiles.

Figure 2 shows the ɨ profile of the reference configuration, as well as the vacuum transform generated by the externally-generated three-dimensional fields alone. Since the transform generated by the current decreases near the edge, the externally generated transform has been designed to have enough shear to produce a total ɨ with positive shear even in this edge region. The monotonically increasing ɨ profile gives a perturbed bootstrap current effect that strongly opposes the formation of magnetic islands. This stabilizing effect is the opposite of the unstable neoclassical tearing mode that has been seen in tokamaks. [8]

Our configurations differ in shape from other stellarators in their strong axisymmetric components of ellipticity and triangularity. These shape components are used to produce good ballooning stability properties. [9,10] The ballooning beta limit for the reference configuration discussed in this paper is about 4%. The plasma core region is calculated to be in the second stability regime for ballooning, and consequently there is considerable ballooning stability margin which could allow the ballooning beta limit to be further increased by peaking the pressure profile.

The reference configuration is marginally stable to external kink modes at $\beta \approx 3.9\%$. The external kink mode has been stabilized by a combination of externally generated shear and an appropriately designed three-dimensional corrugation of the boundary. The potential use of externally generated shear to stabilize kink modes was suggested in several early papers. [11,12] Calculations for NCSX confirm that the external kink in quasi-axisymmetric configurations can be stabilized by this method. [13] However, when externally generated shear alone is used to stabilize the kink, in combination with the requirement for a monotonic $\iota\!\!\!-$ profile, the $\iota\!\!\!-$ in the interior is forced to undesirably low values for neoclassical transport. The development of a second kink stabilization scheme– a three-dimensional corrugation of the plasma boundary with little associated shear– to augment externally generated shear is therefore key to generating attractive kink-stable configurations for NCSX. [5,14]

The stabilizing corrugation is illustrated in Fig. 3, which shows the cross-section of the reference configuration at several values of the toroidal angle, as well as the cross-section of a closely related kink-unstable configuration. The kink unstable configuration has approximately the same $\iota\!\!\!-$ profile, so that the externally generated shear has a substantial stabilizing effect on the kink, but not sufficient to stabilize the mode. A key feature of the corrugation used to complete the stabilization is an outboard indentation, which can be seen in Fig. 3. In the following section we describe the capability of the NCSX coil design to control this feature, varying the degree of indentation to test its effect on kink stabilization.

The external kink mode preserves stellarator symmetry (symmetry under $\varphi \rightarrow -\varphi$, $\theta \rightarrow -\theta$), but not the stellarator's periodicity. It is also necessary to consider modes, corresponding to vertical instabilities in tokamaks, that preserve the periodicity, but not the stellarator symmetry. (All modes can be constructed by a combination of these two types.) Periodicity-preserving modes are found to be robustly stable in kink-stabilized NCSX configurations without a nearby conducting wall. [15] This makes it possible to use stronger axisymmetric shaping than in a tokamak, where the shaping is limited by vertical stability considerations.

Figure 4 shows a calculation of growth rates of periodicity-preserving modes for a set of equilibria interpolating between the reference configuration and a corresponding tokamak. The tokamak boundary is defined by keeping only the axisymmetric($n = 0$) Fourier components defining the boundary of the reference configuration. (See Ref. [16] for a discussion of the Fourier representation of the flux surfaces and boundary in VMEC.) The nonaxisymmetric Fourier components are linearly varied to produce the intermediate configurations. The tokamak case is unstable to a vertical mode, as is generally the case for shaped tokamaks in the absence of a conducting wall. Figure 4 shows that the growth rate of the mode decreases as the three-dimensional deformation is introduced, and the mode is stabilized about 60% of the way to the reference configuration. The existence of such a large stability margin suggests the possibility of applying much stronger axisymmetric shaping to improve performance; this is currently being studied.

While producing a substantial rotational transform and the favorable stability properties described above, the three-dimensional boundary shape of the reference configuration also provides approximate quasi-axisymmetry, meaning that the non-axisymmetric Fourier coefficients of the magnetic field strength in Boozer coordinates [17] are strongly suppressed relative to the axisymmetric term, even though the shape is three-dimensional. For example, even though the m = 2, n = 1 component of the boundary shape has a significant role in producing the desired externally generated shear, the m = 2, n = 1 component of the magnetic spectrum is made small (less than 3.5% of the axisymmetric component) to reduce the neoclassical transport. The adequacy of the level of quasi-symmetry achieved is judged on the basis of calculated transport.

The sensitivity of the stability properties to changes in profiles has been analyzed to assess their robustness. [18] Peaking the pressure profile raises the beta limit for both the kink and the ballooning mode, while a broader pressure profile is more unstable. A broader current profile raises the beta limit for both modes. Decreasing the magnitude of the current also improves stability to the kink mode, but causes ballooning stability to deteriorate somewhat.

B. Coils

Coils for NCSX are configured to reconstruct the reference plasma equilibrium with sufficient accuracy to preserve its kink stability and neoclassical transport properties. Other requirements derive from experimental considerations. To satisfy NCSX physics goals, flat-top pulse lengths at nominal high-beta conditions must be long enough (≥0.5 s) for profile relaxation. For the coil and conductor designs under consideration for NCSX, the pulse length is limited by conductor current density, so reducing this quantity is a key objective of the coil design process. Practical considerations impose further constraints. In order to adapt the design to a fixed axisymmetric background field provided by the PBX-M toroidal field (TF) coil set, saddle coils (coils with no net toroidal or poloidal current) are used to generate the non-axisymmetric fields. Space between the plasma and coils must be allowed to provide room for power and particle handling structures and diagnostics.

The NESCOIL [19] code is used to establish an initial set of saddle coils, by first generating sheet-current solutions on a winding surface surrounding the plasma with a uniform standoff distance of 18 cm. Reconstruction accuracy is targeted by minimizing the normal component of the magnetic field, averaged over the surface of the reference plasma. A genetic algorithm is used to generate a small number of discrete coils by selecting constant-current-potential contours from the sheet-current distribution. The selection method simultaneously targets both low normal component of the magnetic field and low conductor current density. To calculate the latter with sufficient accuracy, the effect of the winding surface curvature and winding-pack geometry details are taken into account. The results strongly favor reducing the number of coils to reduce the conductor current density enough to satisfy minimum pulse length requirements.

The NCSX saddle coil configuration shown in Fig. 5 has 10 coils per period and large unobstructed regions on the outboard side which provides access for heating and diagnostics. Each coil can be independently powered, if necessary, and the mutual inductance between saddle coils and the TF or PF coils is low; these are desirable for plasma control. These coils reconstruct the plasma with a mean value of magnetic field normal to the surface equal to 1.2% of the average magnetic field in the surface, and a maximum value equal to 6.3%, and they have a current density 80% of the maximum allowable. At this value of the field error, the plasma boundary shape is matched with an average deviation of 1.2 cm and a peak deviation of 4.4 cm (compared to an average plasma radius of 42 cm). This reconstruction adequately preserves the neoclassical transport properties of the original configuration. To examine the coil set flexibility to control the kink stability of the plasma, use is made of the sensitivity of the kink mode to the outboard indentation described in Sect. II.A. A single pair of coils (in each period) in the Fig. 5 coil set is effective at controlling the outboard plasma indentation. This is demonstrated in Fig. 6, where cross sections of reconstructed plasmas are shown for two values of the current, differing by 10%, in the relevant coil pair, with all other coil currents are held fixed. A kink-unstable configuration is made marginally stable by this adjustment, with only a slight deterioration of the neoclassical confinement. The dependence of the kink eigenvalue is monotonic with coil current, which implies that this coil pair is a good controller of the kink stability. Thus, by adjusting the currents in this single coil group we gain control of the outboard indentation, and hence an important experimental flexibility to test understanding of kink stabilization mechanisms.

C. Operating Scenarios

The transport properties of NCSX candidate configurations have been evaluated to project accessible beta values, estimate required heating powers, and quantitatively assess the neoclassical transport optimization. The projections have focussed on the expected performance using the existing 50-keV neutral beams from PBX-M. Separate numerical models of neutral-beam ion orbit loss and thermal neoclassical transport are used. These are combined with global confinement scalings to project plausible plasma parameters.

The beam ion orbiting and slowing-down is modeled using two separate three-dimensional Monte Carlo simulation methods [20,21] and the high-beta equilibrium. The codes differ in their treatment of the collision operator, but show the same variations. The neutral beam deposition profile is calculated by the TRANSP code using the oblate cross-section geometry as a 2D approximation. While the beam ions are deposited onto passing orbits, most of the losses are due to pitch-angle scattering to trapped orbits. The pitch angles of lost ions are broadly distributed throughout the trapped region, indicating stochastic orbits as the primary loss mechanism. A small amount of loss appears to be due to stochastic passing orbits [22].

The thermal plasma neoclassical confinement is calculated by Monte Carlo simulations using the gyrokinetic toroidal code (GTC) [23,24], similar to previous models of axisymmetric neoclassical transport. The model simulates the full ion distribution function and the perturbation (δf) of the electron distribution function from a Maxwellian. The radial electric potential is taken as $e\Phi = T_i(0)\Psi$ (where Ψ is normalized poloidal flux), presuming operation in the ion root. The calculated electron neoclassical losses are negligible, compared with the ion losses, as expected for a quasi-symmetric configuration. Eliminating the electric potential reduces the ion energy confinement time by ~25%, indicating that the electric field terms are relatively weak. The calculated ion neoclassical confinement scales approximately as B^2.

The calculated beam-energy loss and thermal neoclassical confinement are combined with the ISS95 global energy confinement scaling [25] (to model anomalous transport) in a zero-dimensional model of the plasma performance, depending on the plasma size, density, magnetic field, heating power, and confinement enhancement over the scaling law. The confinement time is assumed to be the minimum of that given by the scaling law or half of the neoclassical value (i.e. presuming the anomalous losses will at least equal the neoclassical losses). The density is constrained by the Sudo density limit [26]. This model is used to search for parameter choices that minimize the heating power needed to achieve the design β limit, or to search for the minimum confinement enhancement needed to achieve the β limit. Table 1 shows the results from such a search for parameters giving β = 4%, with 6.9 MW of H^0 NBI and a confinement enhancement of 2.3 times ISS95 scaling. The minimum power to achieve a specified beta typically is equally constrained by the global scaling and by (twice) the neoclassical confinement, due to their different scaling with B. This confinement enhancement is similar to the best achieved by W7-AS, at a very different aspect ratio (R/a = 11). For comparison, similar sized PBX-M plasmas achieved β = 6.8% with 5.5 MW of NBI at B = 1.1 T, with a confinement enhancement of 1.7 times ITER-89P scaling or ~3.9 times ISS95 scaling [4]. The ITER-89P and ISS95 scaling multipliers differ because the scaling laws differ in their dependence on aspect ratio and plasma shaping. RF heating is being investigated for NCSX as a possible supplement to the neutral beam power, if needed, or as an alternative to neutral beams to avoid energetic ion losses.

III. QOS Physics Basis

Low-R_0/a quasi-omnigeneous (QO) stellarators [21] obtain reduced neoclassical transport and high-beta stability using an approach that is complementary to the QA approach. In a QO design, energetic particles are confined by approximately aligning their bounce-averaged drift orbits with the magnetic surfaces. [27] Distinguishing QO configuration features are a large helical deformation of the flux surfaces and a “bumpy” (mirror) term in the magnetic field spectrum, which lead to magnetic configurations that are relatively insensitive to beta and have a low bootstrap current, typically ~1/10 that in a comparable axisymmetric system. The QOS experiment is planned to test these properties. The low bootstrap current and insensitivity to beta should allow configurations that are robust against current-driven instabilities, vertical instabilities, and disruptions. The QO configurations studied here have some general similarity to the drift-optimized Wendelstein 7-X (W7-X) “helias” configuration [28], but there are four significant differences. First, a factor of 3-4 smaller plasma aspect ratio means a given plasma radius can be obtained with a correspondingly smaller major radius. The lower aspect ratio would lead to a factor of 3-4 larger 1/R component of the magnetic field and a larger drift off a flux surface, but the structure of the optimized magnetic field produces a 1/R term similar to that in W7-X. Second, the non-zero bootstrap current affects the rotational transform profile $\iota(r)$ and the shear, which could introduce magnetic islands and affect MHD stability. Third, a factor of three larger helical field component allows $\iota > 1/2$ at low aspect ratio without the need for a large plasma current. Fourth, the mirror component of the magnetic field varies strongly with radius, which produces a poloidal B drift that reduces transport and leads to closed drift surfaces for trapped particles, even at low beta would lead to a larger 1/R component of the field and a larger drift off a flux surface, so the structure of the magnetic field is chosen to give the same 1/R term as W7-X.

QO-optimized configurations with three and four toroidal field periods and R_0/a from 3 to 4.8 have been studied. The ι profile has low shear and the average value of ι varies from 0.55 to 0.9 in this sequence of QO configurations. Figure 7 shows the spatial Fourier spectrum of the $|B|$ components for a three-field-period R_0/a = 3.6 QO configuration where the on-axis field is normalized to 1. The largest components are the helical, axisymmetric "1/R" term (but a factor of 4 smaller than for an equivalent tokamak, resulting in a smaller toroidal curvature drift), and mirror terms. The spectrum of smaller compensating field terms is that needed to satisfy the QO physics constraints at low aspect ratio. Figure 8 shows the plasma boundary for this reference configuration and the modular coil set that creates it. The shading indicates contours of constant $|B|$. Changing the current in the corners of the coil set ±50% allows varying the aspect ratio from 2.9 to 4.6. Auxiliary coils would be used for plasma shaping and additional rotational transform control.

A study of the reference QO configuration with $\iota\!\!\!-(0) = 0.56$ and $\iota\!\!\!-(a) = 0.64$ demonstrates its relative insensitivity to changes in beta. Fixed-boundary VMEC [16] equilibrium calculations showed little outward shift of the magnetic axis as beta increased (7.4% of a at $\beta = 2\%$ and 17% at $\beta = 6\%$) [29]. There is only a small (<8% decrease) change in $\iota\!\!\!-(r)$ due to the small bootstrap current at the reference 2% beta [30]. The bootstrap current is not in the direction to stabilize neoclassical islands and tearing modes, but is thought to be sufficiently small to have little effect. Similar low- R_0/a QO configurations have been found in which the bootstrap current is in the stabilizing direction, producing a small increase in $\iota\!\!\!-(r)$ [31]. A magnetic well and monotonically increasing $\iota\!\!\!-$ out to the plasma edge exist in both the vacuum and finite-beta configurations. Ideal ballooning modes set the critical β for QO stellarators rather than kink modes because of their relatively small bootstrap current. Ballooning instabilities are local modes that are driven unstable by the presence of pressure gradients in regions of bad local curvature. Their stabilization is governed by local quantities including the local shear and curvature. A fast 3-D ideal MHD code was developed to evaluate the ballooning growth rate on a prescribed set of flux surfaces and initial points for a given equilibrium [32]. Use of this code to target ballooning stability in the configuration optimization process has doubled the achievable value of β while preserving good transport properties. Figure 9 shows how the configuration in Fig. 8 (stable at β=2%) was modified to be stable at $\beta > 4\%$.

The fundamental QO nature (minimized deviation of bounce-averaged drift orbit surfaces from magnetic surfaces) holds even for the vacuum configuration at low R_0/a. This leads to neoclassical energy confinement times τ_E that are typically 3-5 times the ISS95 scaling value for the machine parameters (R_0 = 1 m, B = 1 T) being considered for QOS. The values for τ_E are obtained from a Monte Carlo calculation of the rate of loss through the outer surface of ions and electrons where the particles are distributed across the plasma cross section according to assumed density and temperature profiles. A electric potential profile $e\phi(r) = T_e(r)$ is assumed in these calculations. The calculated values for the particle diffusivity D and heat diffusivity χ decrease with decreasing density (and collisionality ν^*) and do not exhibit the $1/\nu^*$ transport associated with ripple-induced losses. [33]

IV. Future Work

The physics development for NCSX has led to a reference design with a practical engineering embodiment which demonstrates attractive physics properties of QA stellarators. Future physics development goals include configuration flexibility to robustly support the full range of equilibria needed for startup and physics studies. For the QOS, future work includes the physics development of a reference plasma and modular coil configuration optimized for physics goals, and assessment of flexibility for creating a range of plasma configurations.

V. Conclusions

Experimental results on tokamaks and stellarators and recent rapid developments in theoretical understanding have opened a promising path to a magnetic plasma configuration for fusion power applications that can be sustained in steady-state without disrupting. The freedom to shape a toroidal plasma in three dimensions can be used to stabilize disruption-causing instabilities. The bootstrap current can be used to sustain a compact stellarator plasma configuration with tokamak-like physics properties. The available toroidal physics knowledge base and the theoretical understanding embodied in numerical design tools provide a satisfactory basis for designing experimental facilities needed for the next steps in developing the physics of compact stellarators. The NCSX studies demonstrate a methodology for designing a high-beta quasi-axisymmetric stellarator stable to ballooning, external kink, vertical, and neoclassical tearing modes, consistent with practical considerations of experimental physics and engineering. The QOS studies show that there is a range of quasi-omnigeneous configurations available to provide a basis for experiments to test key properties of the low-aspect-ratio QO approach: transport reduction and configuration insensitivity to beta.

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Figure Captions

Fig. 1. Reference plasma configuration evaluated for NCSX.

Fig. 2. Total rotational transform profile and externally-generated contribution for configuration C82.

Fig. 3. Reference configuration (right) with kink-stabilizing surface corrugation, compared with an uncorrugated, kink-unstable configuration (left).

Fig. 4. Growth rate of periodicity-preserving modes (the vertical mode in the tokamak limit) for a family of configurations in which the non-axisymmetric components in the boundary shape specification are linearly interpolated between the NCSX reference configuration and a tokamak.

Fig. 5. NCSX saddle coil design for reference plasma configuration. (Background TF and PF coils not shown.)

Fig. 6. Reconstructed plasma cross sections, varying the current in the longest saddle coil pair in Fig. 5 by 10%. Dashed configuration (b) is kink unstable. Solid configuration (a) is made marginally stable by the deeper indent on the outboard side.

Fig. 7. Magnetic field structure (magnetic field spectrum versus normalized toroidal flux) for QO plasma configuration in Fig. 8.

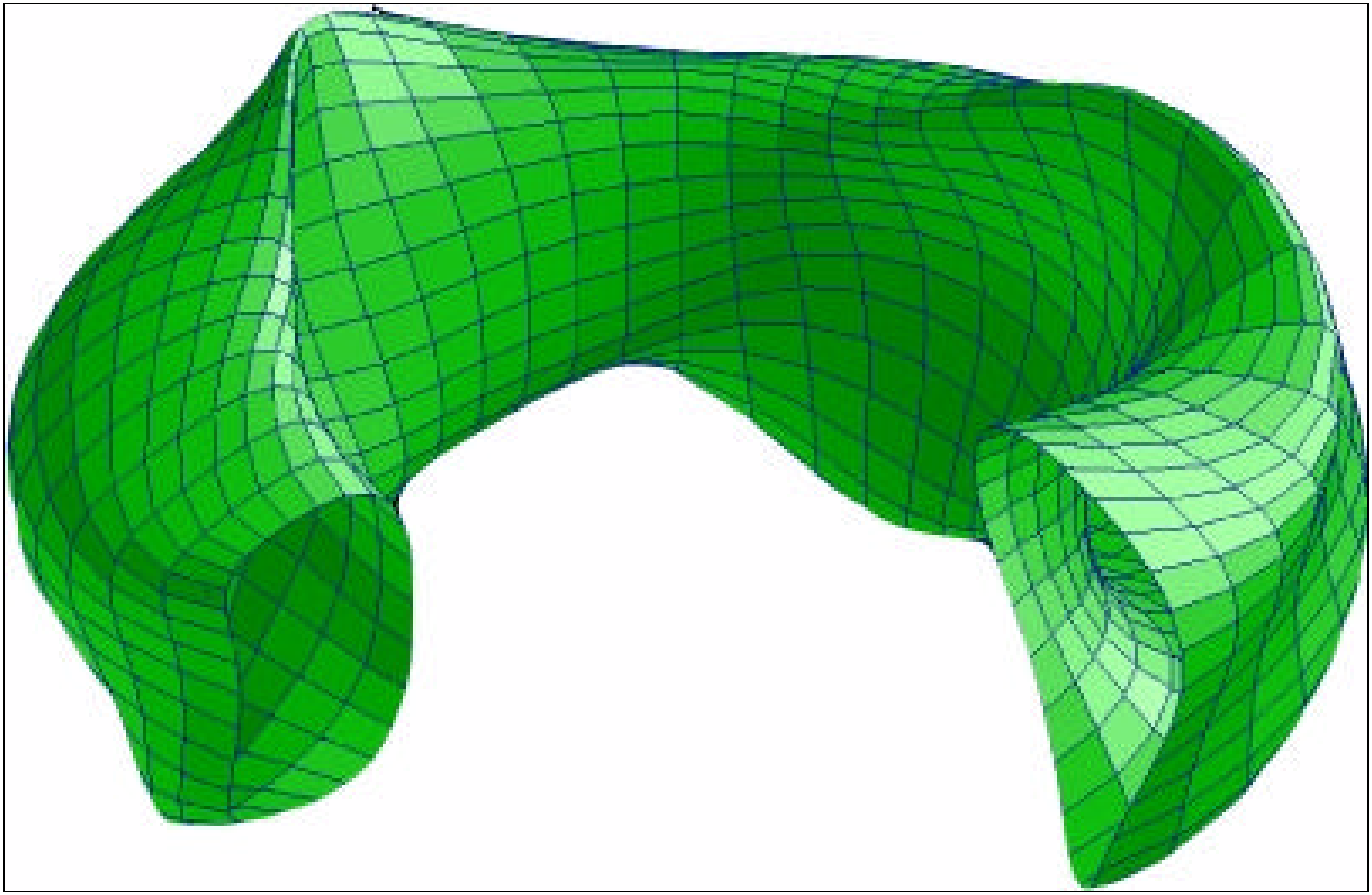
Fig. 8. Plasma surface and coils for a three-period QO plasma configuration.

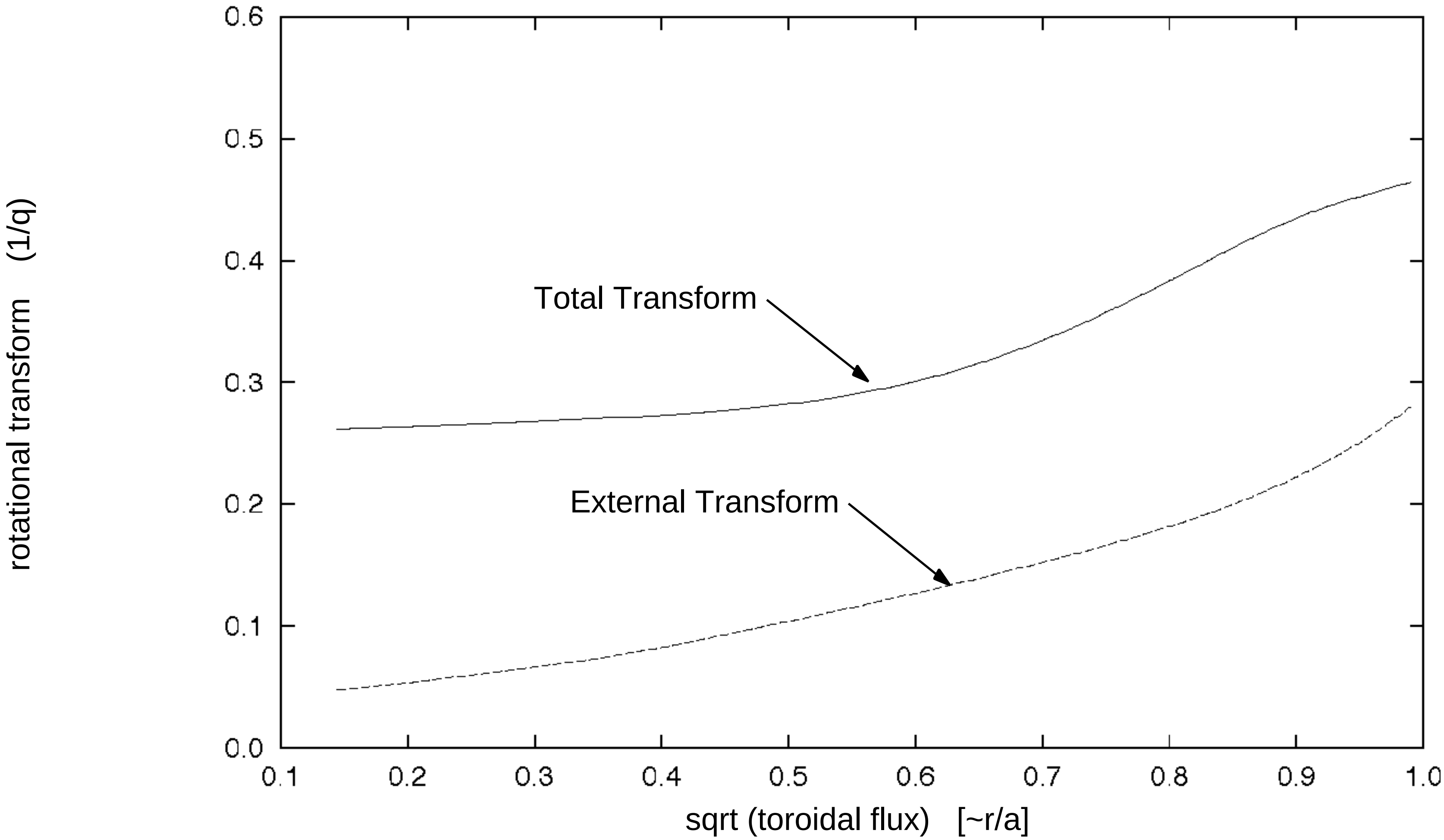
Fig. 9. Plasma surface changes to reduce the ballooning growth rate.

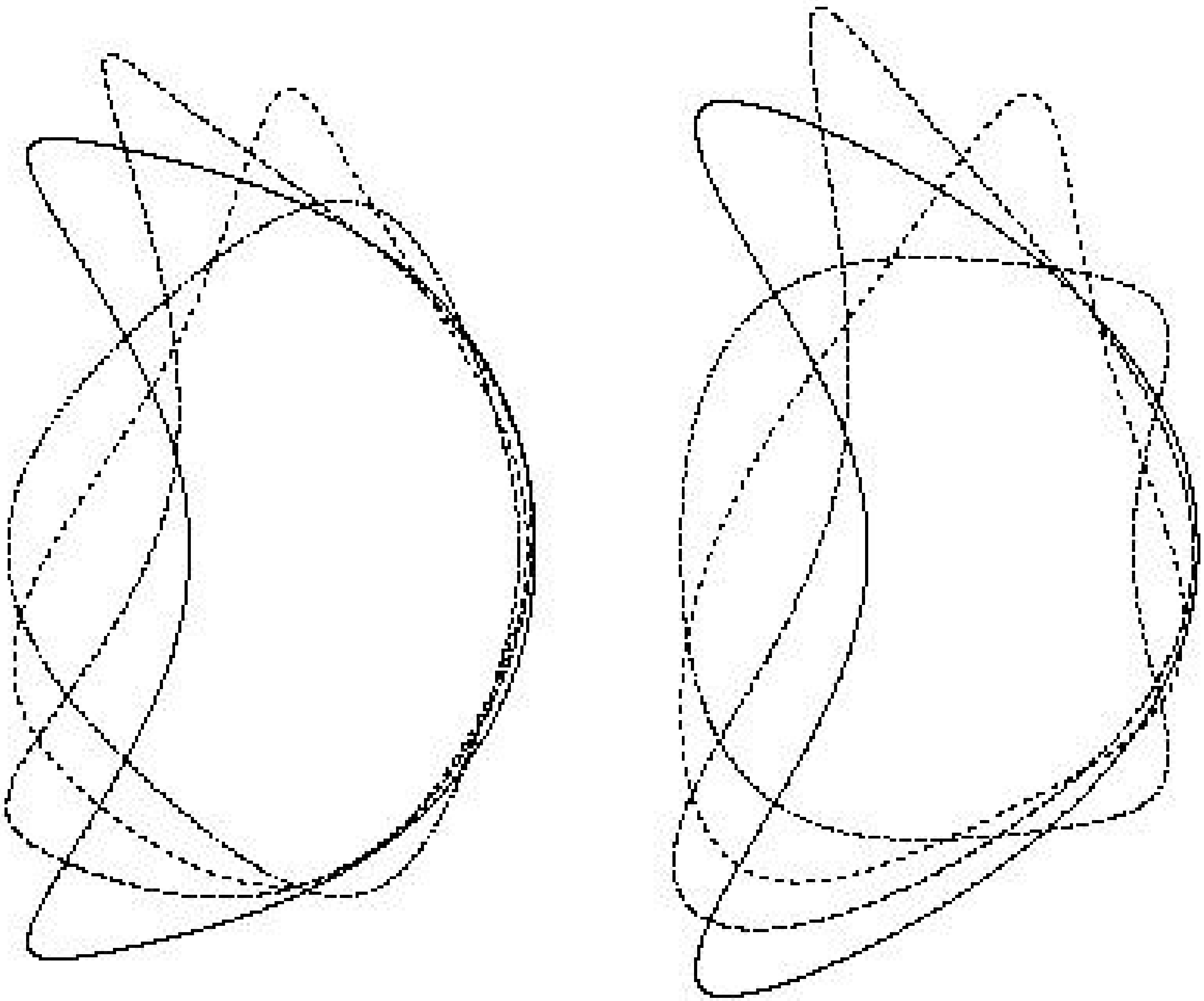
Table 1. Parameters for NCSX Operating Points.

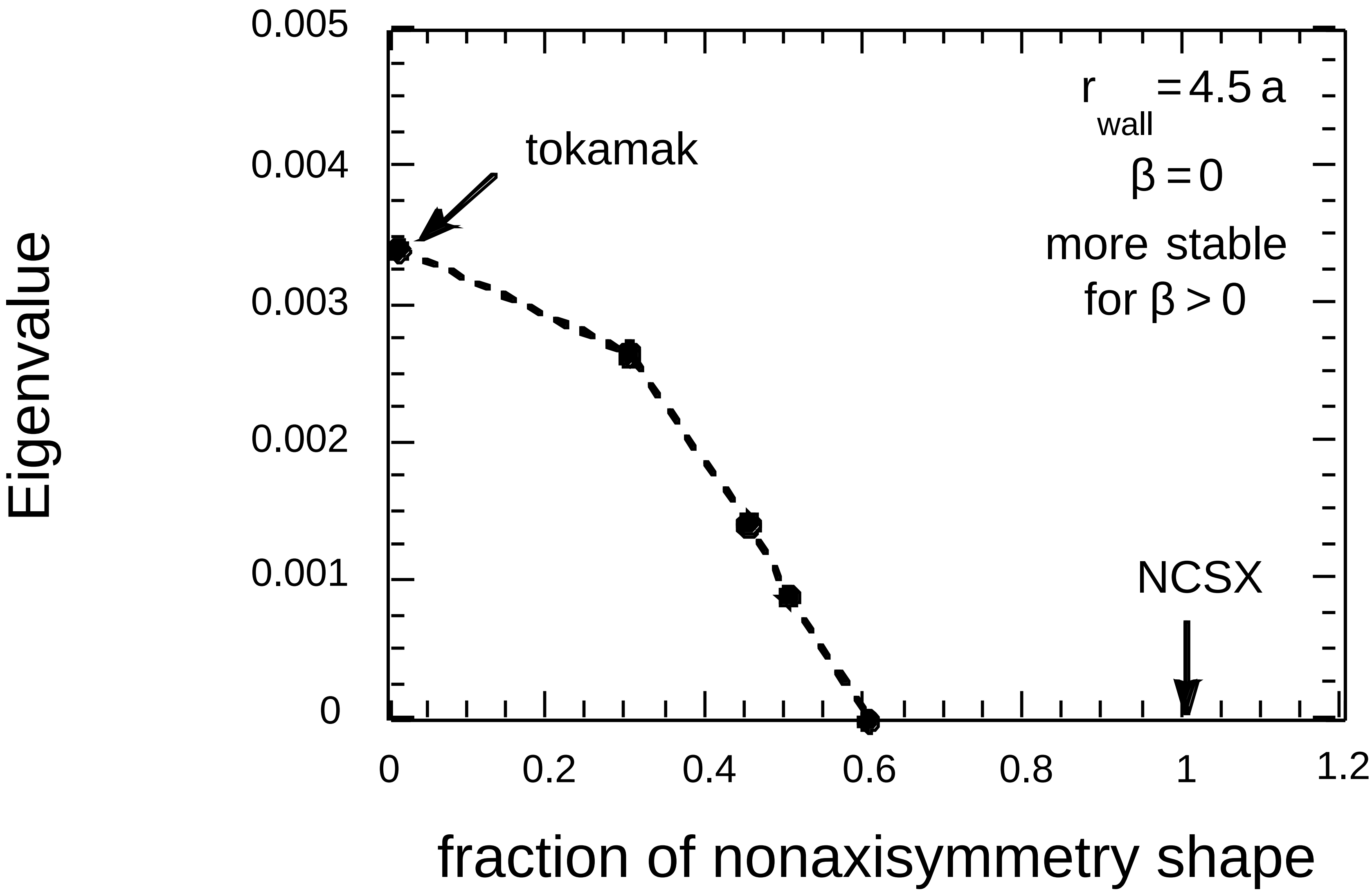
Scenario	$\beta = 4\%$
Magnetic field, B (T)	1.5
Injected power, P (MW)	6.9
Volume-avg. beta β (%) (*)	4.0
Volume-avg. density, n (10^{19} m^{-3})	11.3
Central temperature, T_0 (keV)	2.0
Collisionality parameter (nR/T^2)	4.2
τ_E (ms)	53

*Includes 10% beam beta.

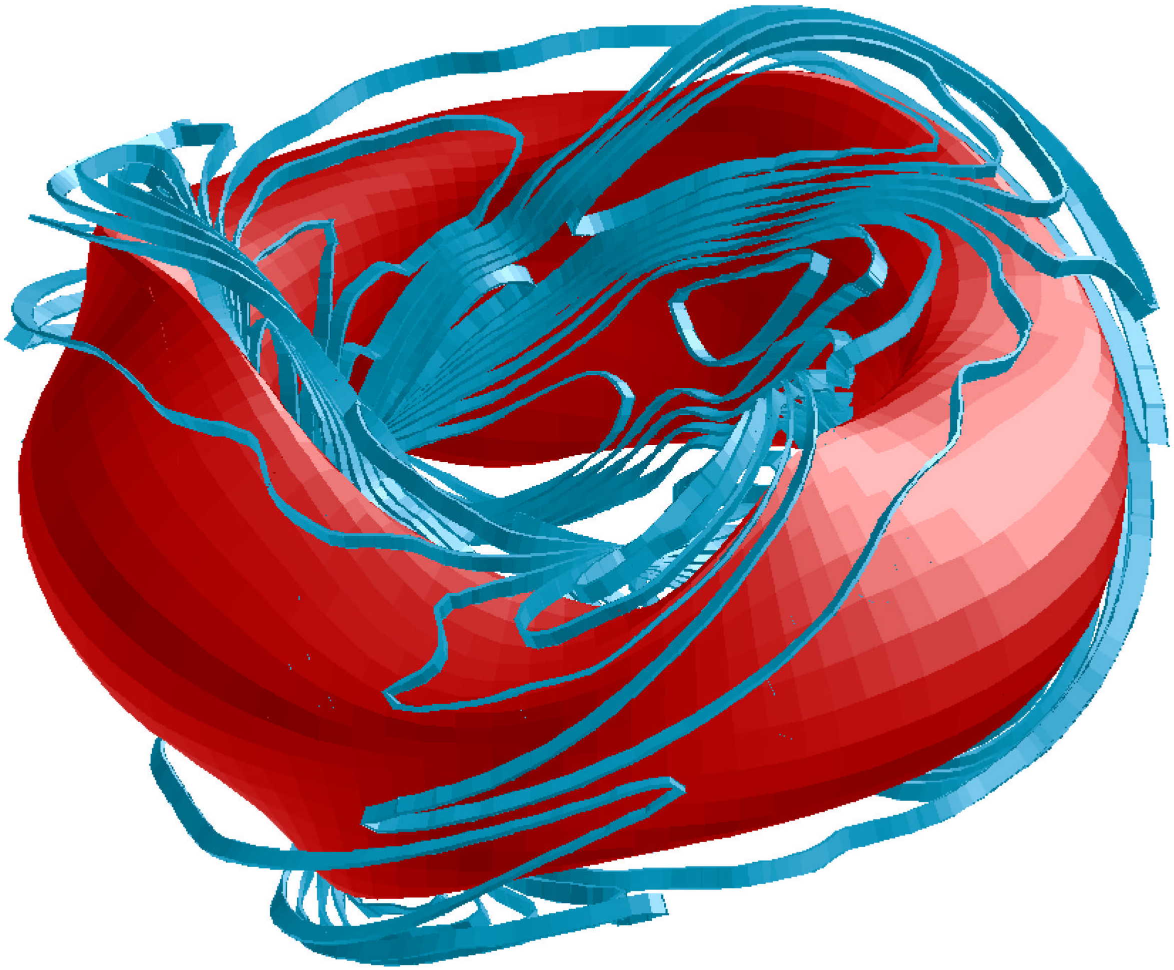


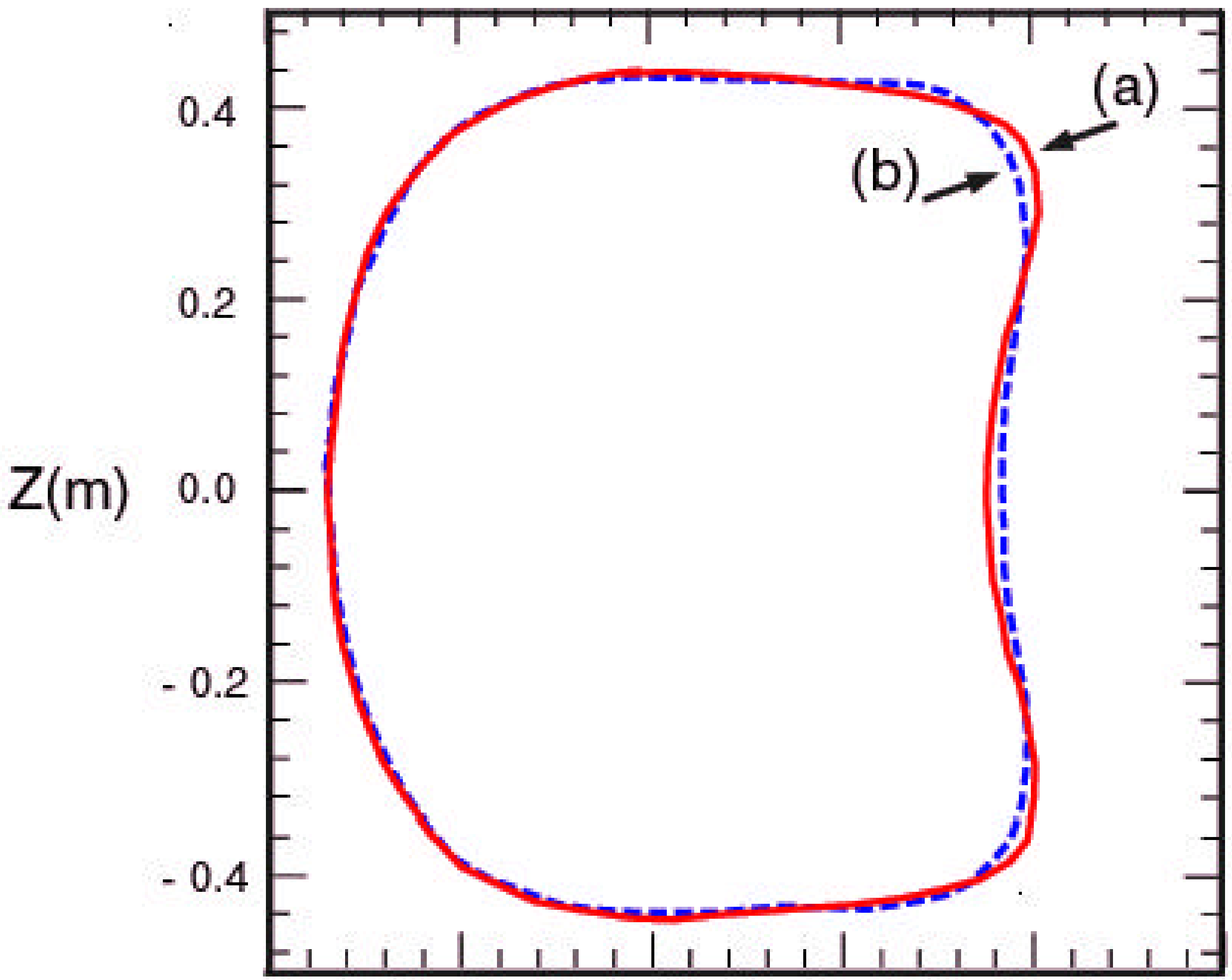

0.6
0.5
0.4
0.3
0.2
0.1
0.0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1.0
Total Transform
External Transform
rotational transform (1/q)
sqrt (toroidal flux) [~r/a]

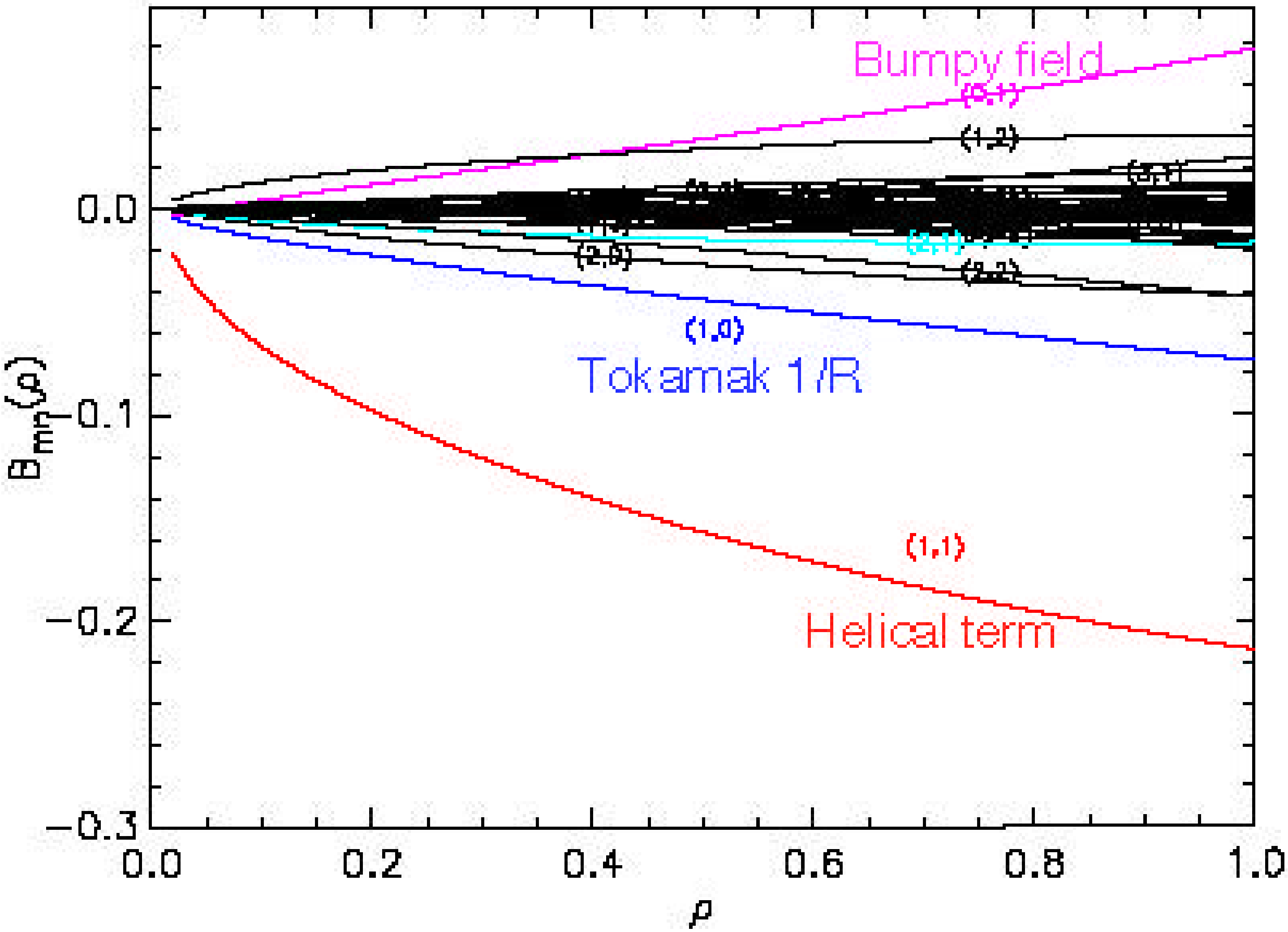




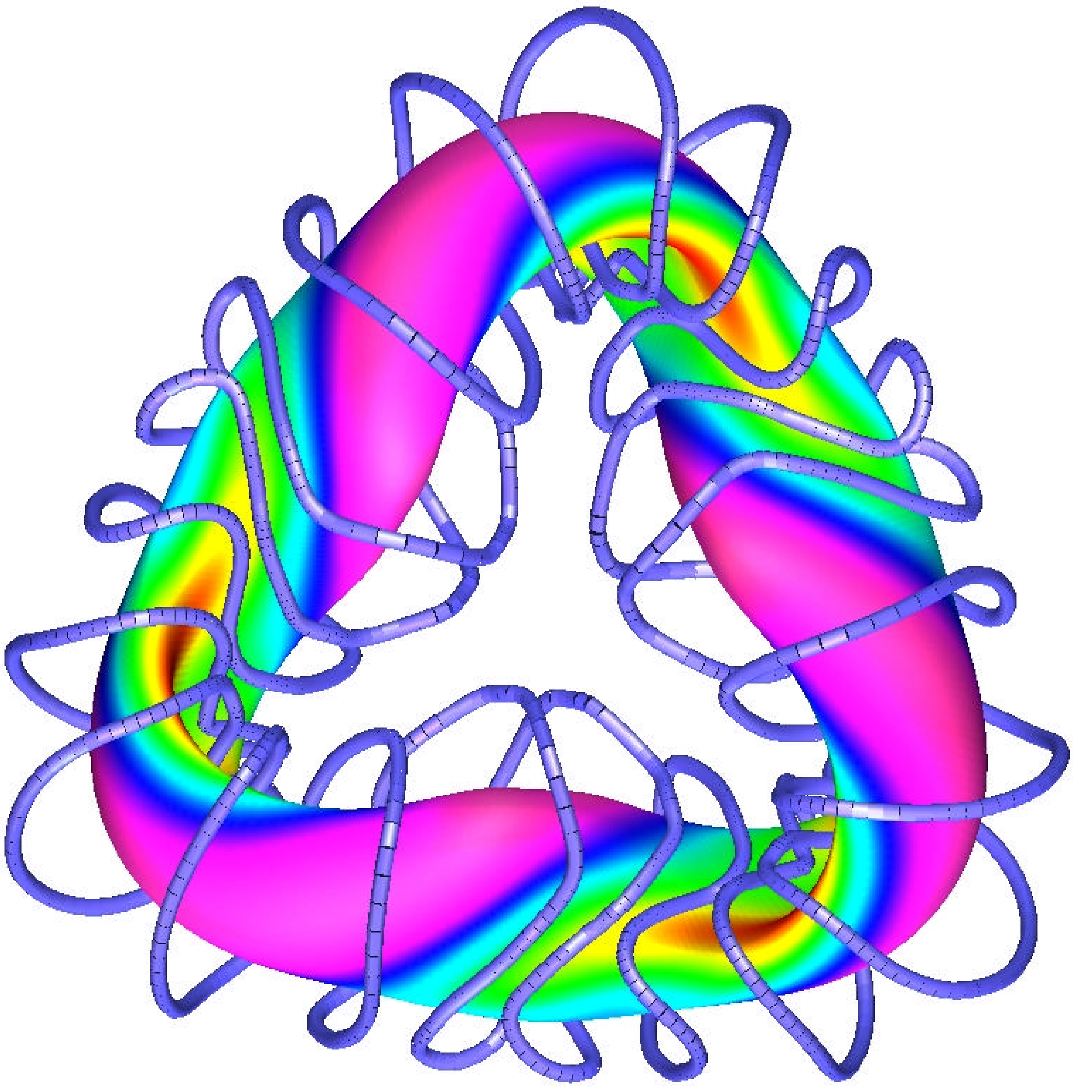
0.005
0.004
0.003
0.002
0.001
0
Eigenvalue
tokamak
r wall = 4.5 a
β = 0
more stable
for β > 0
NCSX
0
0.2
0.4
0.6
0.8
1
1.2
fraction of nonaxisymmetry shape

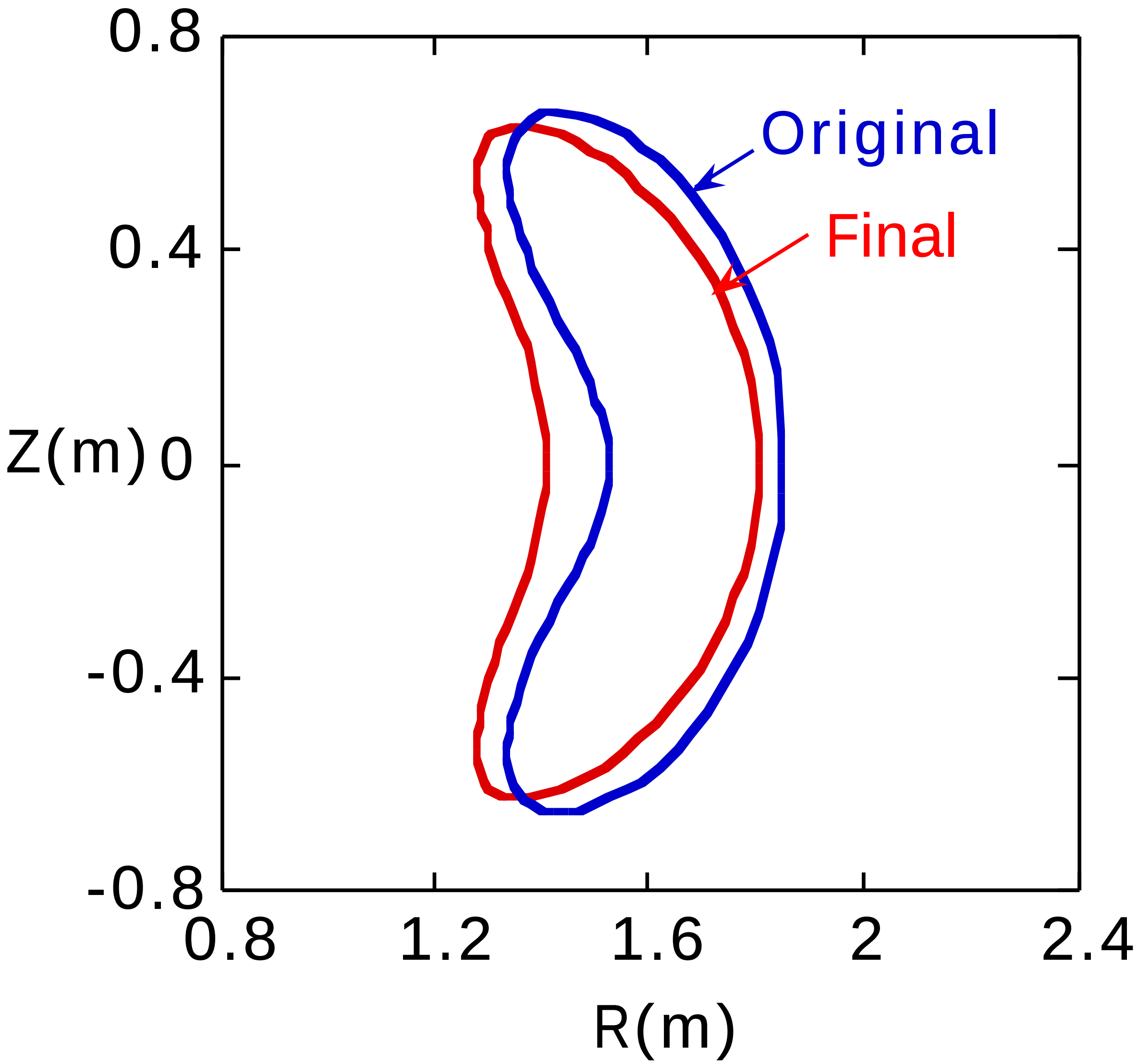



Z(m)
0.4
0.2
0.0
- 0.2
- 0.4
(a)
(b)



Bumpy field
(0,1)
(1,2)
(3,1)
(2,1)
(2,0)
(2,2)
(1,0)
Tokamak 1/R
(1,1)
Helical term
$B_{mn}(\rho)$
0.0
−0.1
−0.2
−0.3
0.0
0.2
0.4
0.6
0.8
1.0
ρ





Original
Final
0.8
0.4
Z(m) 0
-0.4
-0.8
0.8
1.2
1.6
2
2.4
R(m)