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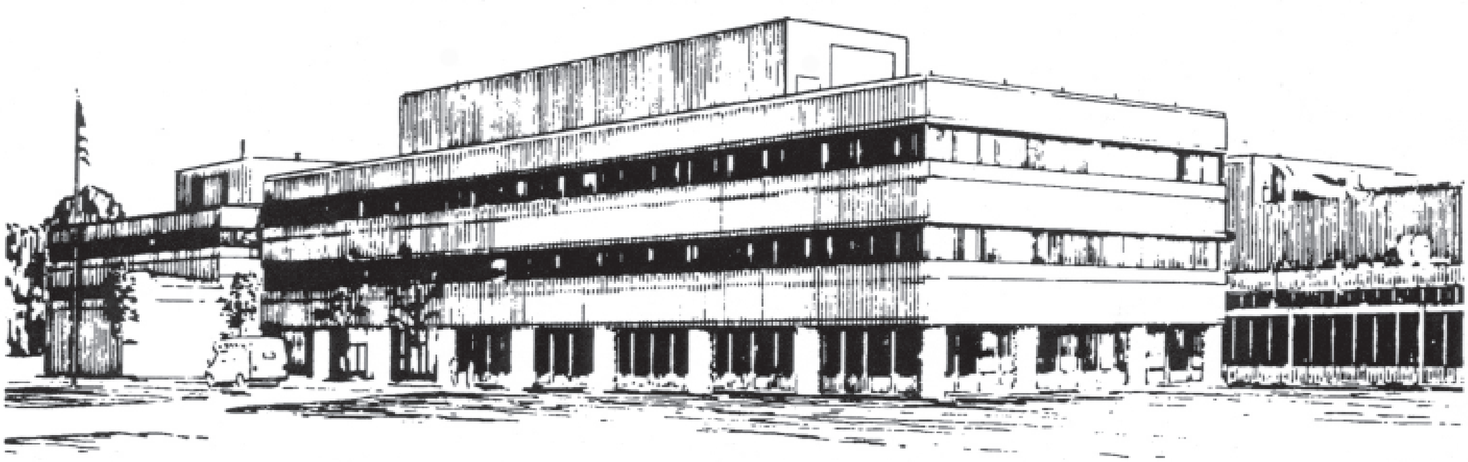

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Physics Design of the National Compact Stellarator Experiment

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Abstract. Compact quasi-axisymmetric stellarators offer the possibility of combining the steady-state low-recirculating power, external control, and disruption resilience of previous stellarators with the low-aspect ratio, high beta-limit, and good confinement of advanced tokamaks. Quasi-axisymmetric equilibria have been developed for the proposed National Compact Stellarator Experiment (NCSX) with average aspect ratio ~4.4 and average elongation ~1.8. Even with bootstrap-current consistent profiles, they are passively stable to the ballooning, kink, vertical, Mercier, and neoclassical-tearing modes for $\beta > 4\%$, without the need for external feedback or conducting walls. The bootstrap current generates only 1/4 of the magnetic rotational transform at $\beta = 4\%$ (the rest is from the coils). Transport simulations show adequate fast-ion confinement and thermal neoclassical transport similar to equivalent tokamaks. Modular coils have been designed which reproduce the physics properties, provide good flux surfaces, and allow flexible variation of the plasma shape to control the predicted MHD stability and transport properties.

1. Introduction

Historically, the major challenge for stellarators has been to provide acceptable drift-orbit confinement, allowing low neoclassical transport losses and adequate fast-ion confinement. One of the two strategies identified [1] to provide adequate drift-orbit and neoclassical confinement is called 'quasi-symmetry', which is based upon work by Boozer [2] showing that drift-orbit topology and neoclassical transport depends only on the variation of |B| within a flux surface, not on the dependence of the vector components of B. (The other strategy, which is to minimize the poloidal variation of the magnetic field strength while also minimizing the Pfirsch-Schlüter and bootstrap currents [3]was developed for the design of the Wendelstein 7-X, with aspect ratio 10.6, where the orbit confinement was specifically optimized to minimize the bootstrap current.) Quasi-symmetry was used by J. Nuehrenberg [4,5] and P. Garabedian [6] to develop stellarators that, while three-dimensional in Euclidean space, have a direction (either helical or toroidal) of approximate symmetry of |B| in (Boozer) flux coordinates. Quasi-symmetric configurations have drift-orbits similar to equivalent symmetric configurations, and thus similar neoclassical transport. Rotation in the quasi-symmetric direction is also undamped, as in a symmetric configuration. The first experimental test of quasi-symmetry is the Helically Symmetric eXperiment (HSX) now operating at the University of Wisconsin [7].

In parallel, there have been advances in the understanding and control of tokamak plasmas. There has been a general confirmation of ideal MHD equilibrium and stability theory and neoclassical transport theory. Methods for stabilizing and manipulating turbulent transport (particularly for ions) have been developed, allowing the elimination of anomalous ion-thermal and particle transport, and reduction of anomalous electron thermal transport.

There is a general understanding of the importance of flow-shear stabilization as a mechanism for stabilizing ion turbulence. In addition, there are a theoretical predictions that undamped turbulence-generated flows (zonal flows) [8] are significant in saturating turbulent transport at the levels observed.

Quasi-axisymmetric stellarators offer novel solutions for confining high-β plasmas by combining the best features of tokamaks and stellarators. They offer the possibility of combining the steady-state low-recirculating power, external control, and disruption resilience of the stellarator with the low aspect ratio, high beta-limit and good confinement of the advanced tokamak. Using the 3D shaping freedom available in a stellarator, configurations can be designed that are MHD stable without nearby conducting structure, require no current drive at high-β, and have good orbit confinement. The quasi-axisymmetry gives good orbit and neoclassical confinement, similar to equivalent tokamaks, and reduced damping of toroidal rotation. The reduced damping may allow manipulation of the radial electric field via driven rotation and the full development of zonal flows, similar to tokamaks. In addition, quasi-axisymmetric plasmas have significant bootstrap current, reducing the rotational transform required from the external coils. The rotational transform profile produced by the 3D shaping and bootstrap current can be designed to monotonically increase towards the plasma edge, like the core region of a 'reversed shear' advanced tokamak. This is predicted to stabilize neoclassical tearing modes, reduce equilibrium islands, and stabilize trapped-particle driven modes. Quasi-axisymmetry is well suited for the design of low aspect ratio configurations, since low aspect ratio forces the n=0, m>1 Fourier coefficients of the magnetic field strength to be large, where n and m are the toroidal and poloidal mode numbers, respectively. Low aspect ratio configurations are attractive in order to minimize the cost of near term experiments and the capital cost of possible future power plants.

The quasi-axisymmetric stellarator is the basis for the design of the National Compact Stellarator Experiment (NCSX) [9]. The NCSX is proposed to study the physics of the beta limit in compact stellarators, the role of 3D shaping and externally generated transform in disruptions, and the ability to operate reliably without disruptions at the beta-limit with low collisionality and bootstrap current consistent profiles. It will also test quasi-axisymmetric reduction of neoclassical transport, the residual flow damping, effects on turbulence, and the ability to induce enhanced confinement. Compact quasi-axisymmetric stellarators are also being investigated for the design of the CHS-qa experiment [10].

2. Configuration Design & Stability

The NCSX target plasma configuration is designed by adjusting the 3D plasma boundary shape to be ideal MHD stable at β =4.25%, have a monotonically increasing ƚ profile (decreasing q profile), and be consistent with the bootstrap current while optimizing the quasi-axisymmetry of the magnetic field. The bootstrap current is evaluated using an axisymmetric calculation, which was found to be an accurate approximation based on benchmarking against three-dimensional Monte Carlo calculations [11, 12] that Figure 1 shows the plasma boundary shape for the adopted configuration. It has 3 field periods and an average aspect ratio $\langle A \rangle = R/\langle a \rangle = 4.4$, where R is the major radius and $\langle a \rangle$ is the average minor radius. This configuration is calculated to be passively stable to the ballooning, low-n external and internal kink, and vertical instabilities up to β = 4.1%, without need for a conducting wall or active feedback systems. The rotational transform profile increases from 0.4 on axis to 0.66 near the plasma edge, dropping to 0.65 at the plasma edge. Approximately 1/4 of the edge rotational transform is from the bootstrap current and 3/4 is from the external coils. No external current drive is required. The bootstrap current is substantially smaller than in an equivalent advanced tokamak, due to the large transform from the coils. Due to the dominance of the coil generated rotational transform, the equilibrium is less sensitive to the pressure profile shape than in an advanced tokamak. This should allow control of the equilibrium via the external coils.

The kink stability is calculated to be due to the high rotational transform, reduced bootstrap current density, and the spatial variation of the local magnetic shear [13].The passive vertical stability appears to be due to the substantial rotational transform produced by the external coils. Due to the rising rotational transform profile, neoclassical tearing modes are theoretically stable over all but the plasma edge. Favorable neoclassical energy confinement and fast-ion orbit confinement are calculated as a result of the good quasi-axisymmetry (low helical ripple) in this design.

The toroidally averaged shape is similar to an advanced tokamak, with average elongation of 1.8 and an inside indentation of 9%. For this average shape, an equivalent current I_P^{Equiv} can be defined as the current required to match the stellarator edge rotational transform in a tokamak with the same average shape. The NCSX is envisioned to have R = 1.42 m, $\langle a \rangle$ = 0.33 m, and B up to 1.7 T (at full external rotational transform). For these parameters, I_P^{Equiv} = 0.71 MA.

The quality of the flux-surfaces for this configuration has been evaluated using the PIES equilibrium code [14]. Figure 2(a) shows the calculated fixed-boundary equilibrium flux surfaces for the equilibrium of Fig. 1, showing a significant n/m=3/5 island, with a width of ~10% of the minor radius and a smaller n/m=3/6 island in the core. These calculations do not include neoclassical-healing effects, from suppression of the bootstrap current in the island. An analytic estimate of the neoclassical-healing gives an expected island width of < 5% of the plasma minor radius.

The PIES code was used to calculate small (≤ 4 mm) perturbations in the resonant Fourier components of the plasma boundary shape needed to remove the 3/5 island [15]. The resulting equilibrium, see Fig. 2(b), confirms that the 3/5 island was removed, though a small residual 6/10 island remains that was not targeted. The plasma stability and transport characteristics of the perturbed equilibrium have been analyzed and found to be unchanged from the original design.

Confinement

The degree of quasi-axisymmetry can be characterized by the effective ripple strength $\varepsilon_{h,eff}$ [16], calculated numerically to match the $1/\nu$ transport regime. As shown in Fig. 3, the effective ripple rises exponentially to ~1.2% at the plasma edge. The toroidal spectrum of the ripple is dominated by low-order perturbations, n=3 and 6, which reduces its effect on confinement relative to typical tokamak ripple with n ~ 20. This low $\varepsilon_{h,eff}$, together with the relatively-high rotational transform results in acceptably low fast ion losses, as calculated by Monte-Carlo simulations [17] using the full 3D magnetic field. The calculated energy losses of 40 keV H-neutral beam ions with B=1.7 T is ~15% for co-tangential injection and 23% for counter-injection. These counter-injection losses are similar to those for a tokamak of similar size and are low enough that balanced neutral beam injection can be envisioned to control the beam-driven current and to allow control of rotation. The calculated alpha-particle losses in a projected reactor are <20%, including collisional effects, depending on the final size.

The predicted plasma transport is dominated by the anomalous transport and the toroidal neoclassical losses. The predicted ripple thermal transport is negligible in all cases studied. As an example, Fig. 4 shows the predicted radial power flow and temperature profiles for a β = 4% plasma with a collisionality ν^* =0.25 at the half radius, using B=1.2T, 6MW of neutral beam injected power, an average density of 6×10^{19} m^{-3} , and spatially uniform anomalous transport coefficients. Note that the ripple-neoclassical flux is insignificant and that the anomalous transport dominates over most of the profile. Simulations using the Lackner-Gottardi model give similar results. The calculations indicate that achieving β = 4% with these parameters requires a global confinement time 2.9 times the ISS-95 scaling, somewhat higher than the best achieved on LHD and W7-AS. Since this configuration is designed to have tokamak-like drift orbits and the simulations predict tokamak-like transport, it is reasonable to compare this confinement to tokamak global scalings. If we use I_P^{Equiv} to evaluate the tokamak scaling, the required confinement to achieve β = 4% and ν^* =0.25 is ~0.9 times the ITER-97P prediction. For comparison, similar sized PBX-M plasmas achieved β = 6.8% with 5.5 MW of heating and B = 1.1T achieving a confinement of 1.7 times the ITER-97P prediction or ~3.9 times the ISS-95 prediction. Since ISS-95 and ITER-89P have different parametric dependencies, the confinement multipliers must be expected to vary separately. For the same conditions, except B = 1.7 T, central temperatures of 2.3 keV, ν^*=0.1 are predicted.

3. Coil Design & Flexibility

Of the wide range of coil topologies and designs that has been explored, the modular coils (shown in Fig. 5) best reproduce the physics properties and good flux surfaces of the target equilibrium. These coils will be used with a set of poloidal field coils, to control plasma position and average shape, and a weak toroidal solenoid to allow variation of rotational transform. The modular coil design shown has 6 coils per period, with 3 different coil shapes. The coils as designed allow tangential neutral beam injection and tangential diagnostic views. None of the coils is centered on symmetry cross sections, allowing good diagnostic access there. The coils produce equilibria that approximate the original fixed-boundary plasma shape subject to engineering constraints on coil characteristics, such as bend radii and coil-separation distances. To test the adequacy of a coil design, the coil currents are re-optimized to achieve the original plasma criteria (MHD stability and quasi-axisymmetry) based on free-boundary equilibrium calculations. Free-boundary optimized equilibria have been found that reproduce the key properties of original fixed-boundary design.

The coils must not introduce large islands or stochastic field regions. The coil shapes are perturbed to remove resonant fields that produce islands, as calculated in a free-boundary PIES equilibrium. The algorithm is similar to that used for removing the fixed-boundary equilibrium islands, above. The resulting coils are calculated to be able to produce equilibria with good flux surfaces over a wide range of conditions.

For experiments, the coil system must robustly handle a variety of pressure and current profiles and be flexible to handle the discharge evolution and generate a variety of configurations for physics studies. In studies to test whether a coil design can produce equilibria with specified properties, it is assumed that the current in each coil type can be independently controlled (preserving stellarator symmetry). Figure 6 shows a study of the accessible range of edge rotational transform with the modular coils, for fixed pressure and plasma current, while optimizing quasi-axisymmetry and constraining the plasma to be within a design vacuum vessel. For the cases shown, the ripple magnitude is no more than 1.8 times the original optimized configuration. A wide range of operation is available, including configurations where the transform profile is entirely above or below 1/2. This flexibility will allow control of the edge rotational transform, if needed, preventing it from passing through 1/2 during the discharge evolution, thus avoiding the tearing modes and disruptions observed in hybrid operation of W7-AS [18]. A similar study varied the magnetic shear down to approximately shearless at full plasma current by varying the modular coil currents. In these cases, the quasi-symmetry was degraded by up to a factor of 5 (for the shearless case). A set of trim coils is planned to control low-order resonant field components and the islands they generate, to allow operation over a large range of magnetic transform profiles.

Accessible free-boundary equilibria have been found with substantially improved or degraded quasi-axisymmetry, for fixed plasma profiles and ɨ profile. Similarly, the kink-beta limit can be varied by at least a factor of 3 just by varying the plasma shape via external coil currents, either at fixed edge ɨ or with a fixed shear profile. This ability to manipulate the plasma characteristics will enable controlled experiments for comparison with theoretical predictions and testing models. Surprisingly, the β -limit for free-boundary equilibria has not been found yet. Stable equilibria with $\beta > 6.5\%$ have been found with some degradation of the quasi-symmetry, but without yet making use of profile optimization.

The evolution of the plasma current from vacuum through an Ohmic current-ramp to equilibration with the bootstrap current has been simulated using an assumed temperature evolution. By assuming early auxiliary heating to increase the temperature, as used in reversed-shear tokamak experiments, broad current profiles were predicted which equilibrate with the bootstrap current in ~0.4 sec. Simulation of the evolution from vacuum to $\beta = 4.25\%$ showed reasonable coil-current variations and that kink-modes were calculated to be stable throughout the evolution. Stable evolution scenarios have been found where ι(a) either crosses 1/2 or is always above 1/2. In all cases studied, Δ' analysis indicates that the current profiles are stable or marginally stable to tearing instabilities.

In simulations of unidirectional tangential neutral beam injection, the beam driven current strongly changed the core rotational transform. For co-tangential only injection, the central rotational transform rapidly goes above one, producing a tokamak-like shear profile, which is unstable to neoclassical tearing modes. From these simulations, balanced co- and counter-injection will be required to obtain the optimized current profile. Variations away from balanced injection could be used to control the central magnetic shear.

4. Conclusions

A novel compact quasi-axisymmetric stellarator has been designed for NCSX, combining features from optimized stellarators and advanced tokamaks, and offering a possible path to steady-state reactors without current drive or disruptions. Extra design flexibility from 3D shaping has been used to passively stabilize the kink, vertical, ballooning, Mercier, and neoclassical tearing modes at $\beta > 4\%$ without need for external conducting walls or feedback, while maintaining good orbit confinement. The calculated confinement characteristics are similar to an equivalent tokamak. This NCSX design demonstrates the power of the recent advances in experimental and theoretical understanding and numerical modeling, and illustrates the possibilities available for magnetic confinement with 3D shaping.

Figure Captions:

FIG. 1. Plasma boundary shape in four poloidal cross-sections separated 20^0 toroidally.

FIG. 2. Poincare plot of flux-surface structure: (a) original target plasma of Fig. 1, (b) with boundary perturbed to remove islands(fewer flux surfaces plotted)

FIG. 3. Radial profile of effective helical ripple.

FIG. 4. (a) Predicted T_e and T_i profiles, and (b) Radial power flows for B =1.2 T, P =6 MW.

FIG. 5. Optimized modular coils for the equilibrium of Fig.1 .

FIG. 6. Free-boundary optimized variations of the rotational transform profile using modular coils, maintaining approximate quasi-symmetry, with fixed plasma pressure and current profiles.

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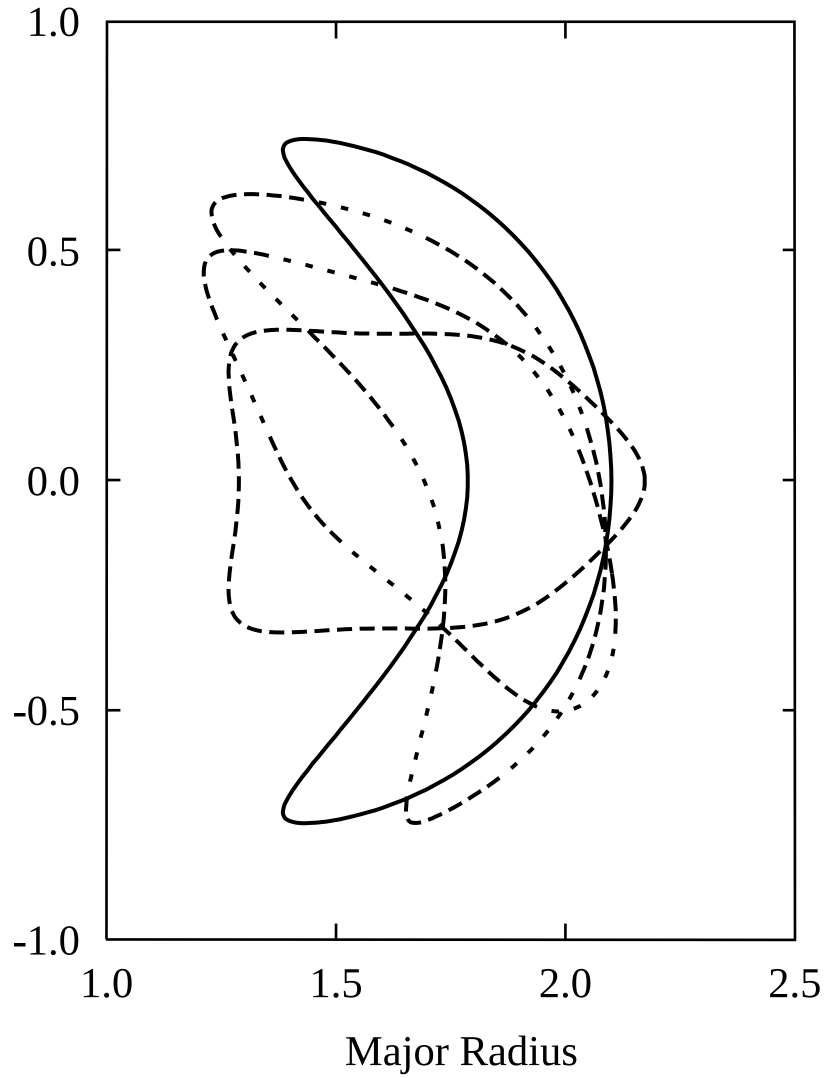


Fig. 1

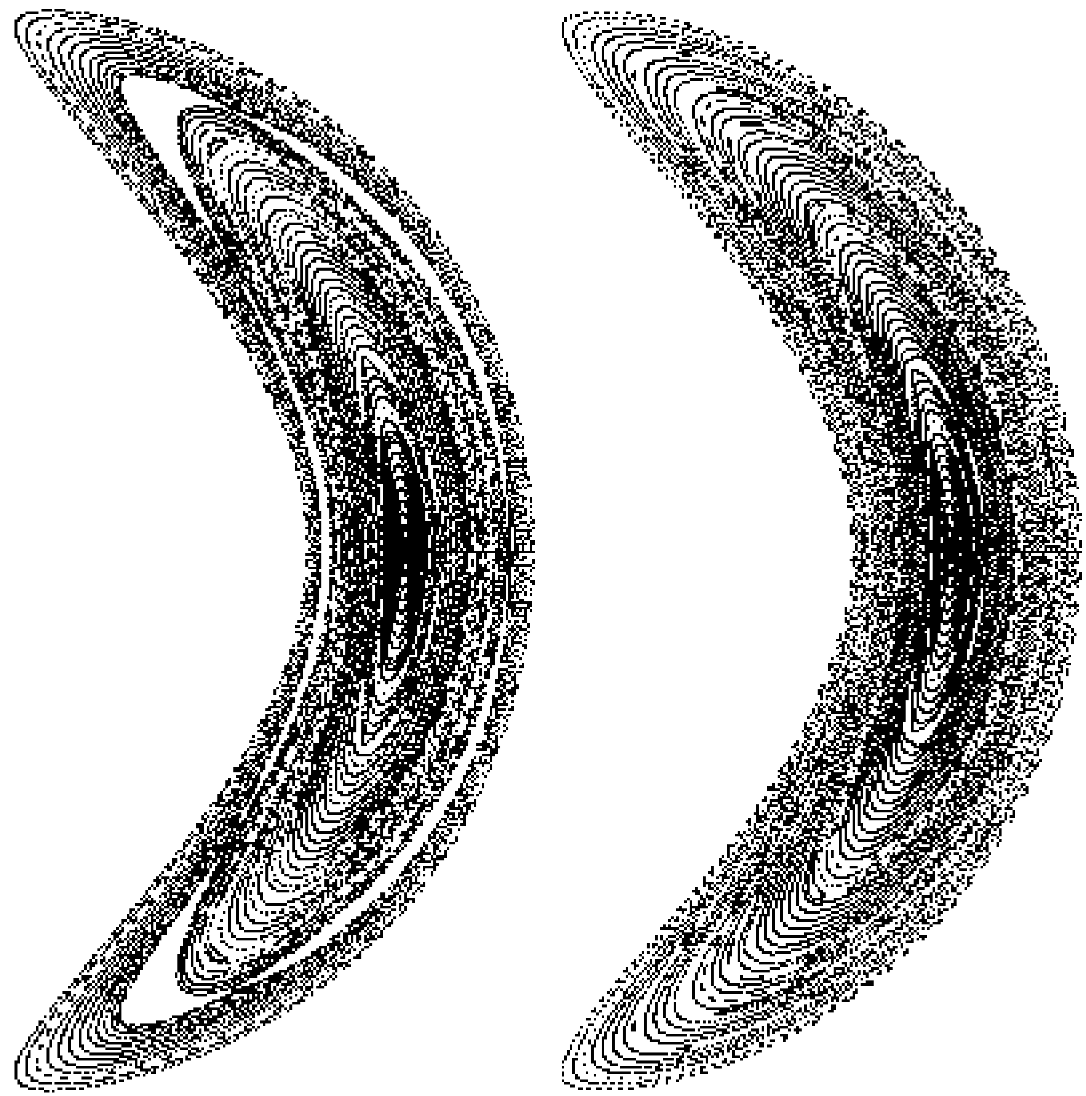


Fig. 2

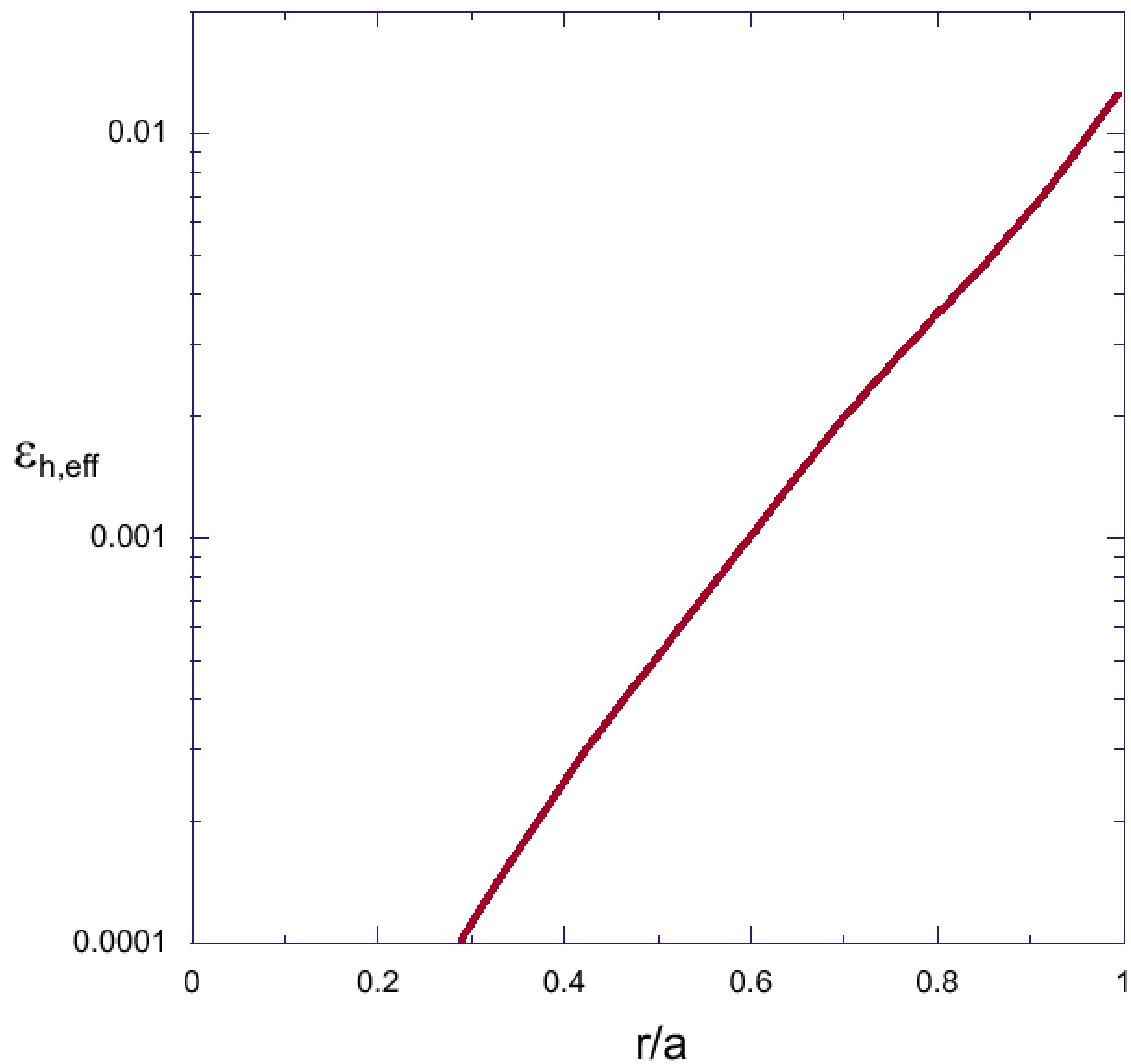
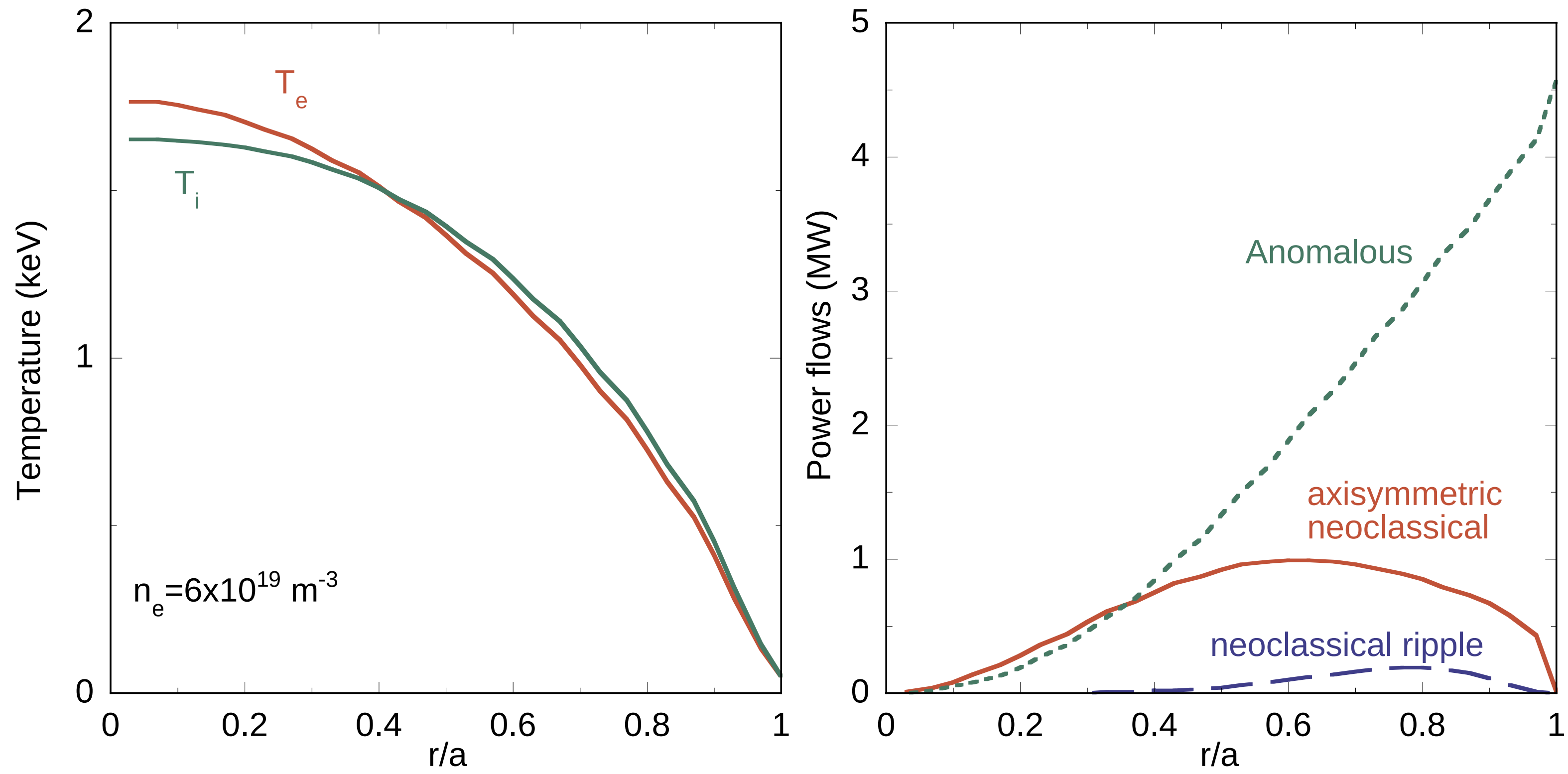


Fig. 3



Temperature (keV)
2
1
0
T_e
T_i
n_e=6x10^19 m^-3
0
0.2
0.4
0.6
0.8
1
r/a
Power flows (MW)
5
4
3
2
1
0
Anomalous
axisymmetric
neoclassical
neoclassical ripple
0
0.2
0.4
0.6
0.8
1
r/a

Fig. 4

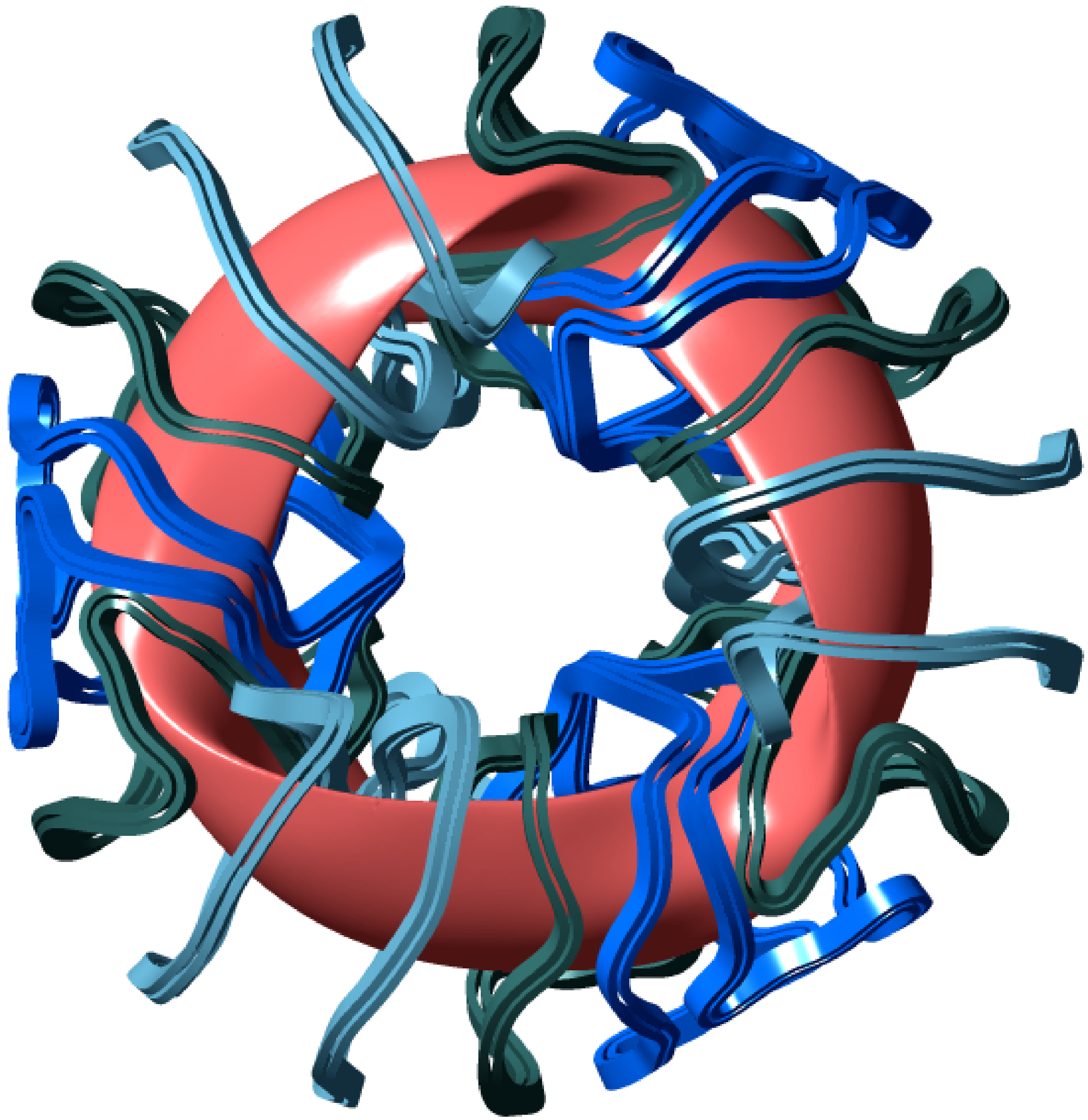


Fig. 5

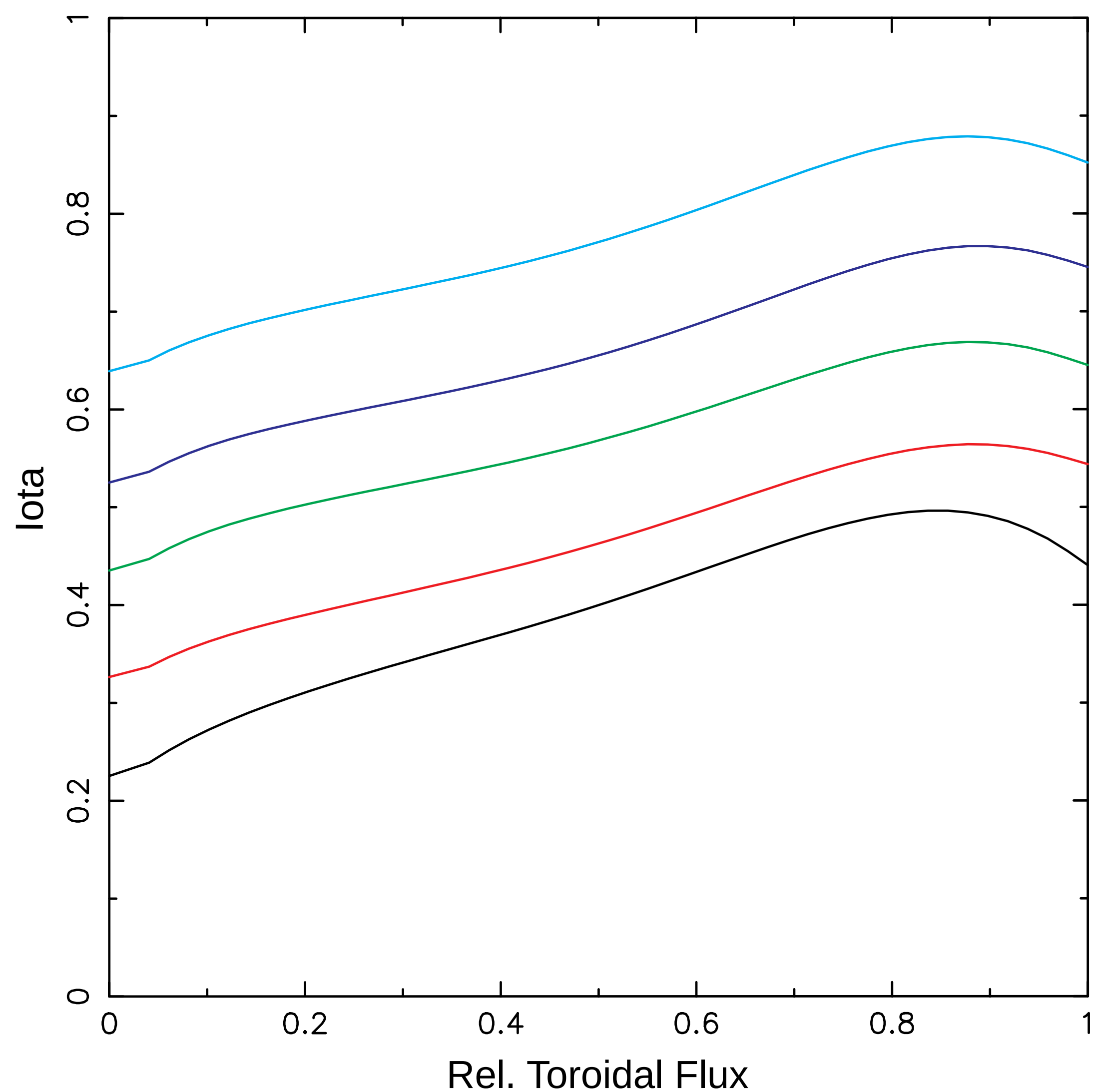


Fig. 6

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