CP1.033

Physics Design of NCSX

The National Compact Stellarator Experiment

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Princeton Plasma Physics Laboratory

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in collaboration with

Auburn, Wisconsin

Australia, Austria, Germany, Japan , Russia, Spain, Switzerland

Motivation: Build Upon Recent Advances in Understanding

Stellarators:

- Design for orbit confinement, good flux surfaces
- Numerical design to obtain desired physics properties
- Accurate construction of experiments with good properties

(e.g. W7AS, HSX, and superconducting LHD)

Tokamaks:

- Confirmation of ideal MHD equilibrium & stability theory; neoclassical theory; neoclassical tearing theory
- Importance of shear-flow & zonal (self-generated) flows for turbulence stabilization

Challenges:

– β_T ~ 5% steady state, good confinement without disruptions (~ARIES-RS)
– sustainment of current with minimum recirculating power

($\Rightarrow$ ~80% self-generated current boostrap)

Combine Best Features of Both

- Use flexibility of 3D shaping to combine best features of stellarators and tokamaks, synergistically, to advance both

  —Stellarators: *Externally-generated helical fields; no need for external current drive; generally disruption free.*

  —Advanced tokamaks: *Excellent confinement; low aspect ratio – affordable, high power density; self-generated bootstrap current*

The compact stellarator opportunity

NCSX Mission

Understand…

- Beta limits and limiting mechanisms in a low-A current carrying stellarator
- Effect of 3D fields on disruptions
- Reduction of neoclassical transport by QA design.
- Confinement scaling; reduction of anomalous transport by flow shear control.
- Equilibrium islands and neoclassical tearing-mode stabilization by choice of magnetic shear.
- Compatibility between power and particle exhaust methods and good core performance in a compact stellarator.
- Explore Alfvenic-mode stability in reversed shear compact stellarator

Demonstrate…

- Conditions for high-beta, disruption-free operation

Acquire the physics data needed to assess the attractiveness of compact stellarators. (adopted as 10-year goal by FESAC-1999)

Energy Vision: a More Attractive Reactor

Vision: A steady-state toroidal reactor with

- No disruptions
- No near-plasma conducting structures or active feedback control of instabilities
- No current drive ($\Rightarrow$ minimal recirculating power)
- High power density (~3 MW/m^2)

Likely configuration features (based on present knowledge)

- Rotational transform from a combination of bootstrap and externally-generated (how much of each?)
- 3D plasma shaping to stabilize limiting instabilities (how strong?)
- Quasi-axisymmetric to reduce helical ripple transport, alpha losses, flow damping (how low must ripple be?)
- Power and particle exhaust via a divertor (what topology?)
- $R/\langle a \rangle \sim 4$ (how low?) and $\beta \sim 4\%$ (how high?)

Design involves tradeoffs.

Need experimental data to quantify mix, assess attractiveness.

Two strategies for Orbit Confinement in 3D

3D shape of standard stellarators $\Rightarrow$ no conserved canonical momenta

orbits can have resonant perturbations, become stochastic $\Rightarrow$ lost

B is bumpy every direction $\Rightarrow$ rotation is strongly damped

- 'quasi-symmetry'
  - Boozer (1983) Drift orbits & neoclassical transport depends on variation of |B| within flux surface, not the vector components of B !
  - If |B| is symmetric in "Boozer" coordinates, get confined orbits like tokamak

  $\Rightarrow$ neoclassical transport very similar to tokamaks, undamped rotation

- Non-symmetric drift-orbit omnigeneity; "linked mirror" configurations
  - reduce: ∇BxB drift $\Rightarrow$ orbit width, Pfirsch-Schluter & bootstrap currents
  - Principle of W-7X, new German superconducting experiment (A=11)
  - Being explored at low aspect ratio in the design of QPS (ORNL)

Boozer coord: straight field-line coordinates, Jacobian $\propto 1/B^2$

Quasi-Axisymmetry Offers Innovative Solutions

Use 3D shaping to:

- Quasi-axisymmetry to close orbits, allow flow, get good confinement
  Take advantage of tokamak advances on transport control
  Use bootstrap to raise iota

- Aspect ratio ~ 4 $\Rightarrow$ n=0 toroidal Fourier terms large
  $\Rightarrow$ compatible with quasi-axisymmetry

- Passively stabilize external kink, vertical, neo-tearing, ballooning modes
  – expand safe operating area to $\beta \geq 4\%$,
  without need for conducting walls or feedback systems
  – prevent disruptions?

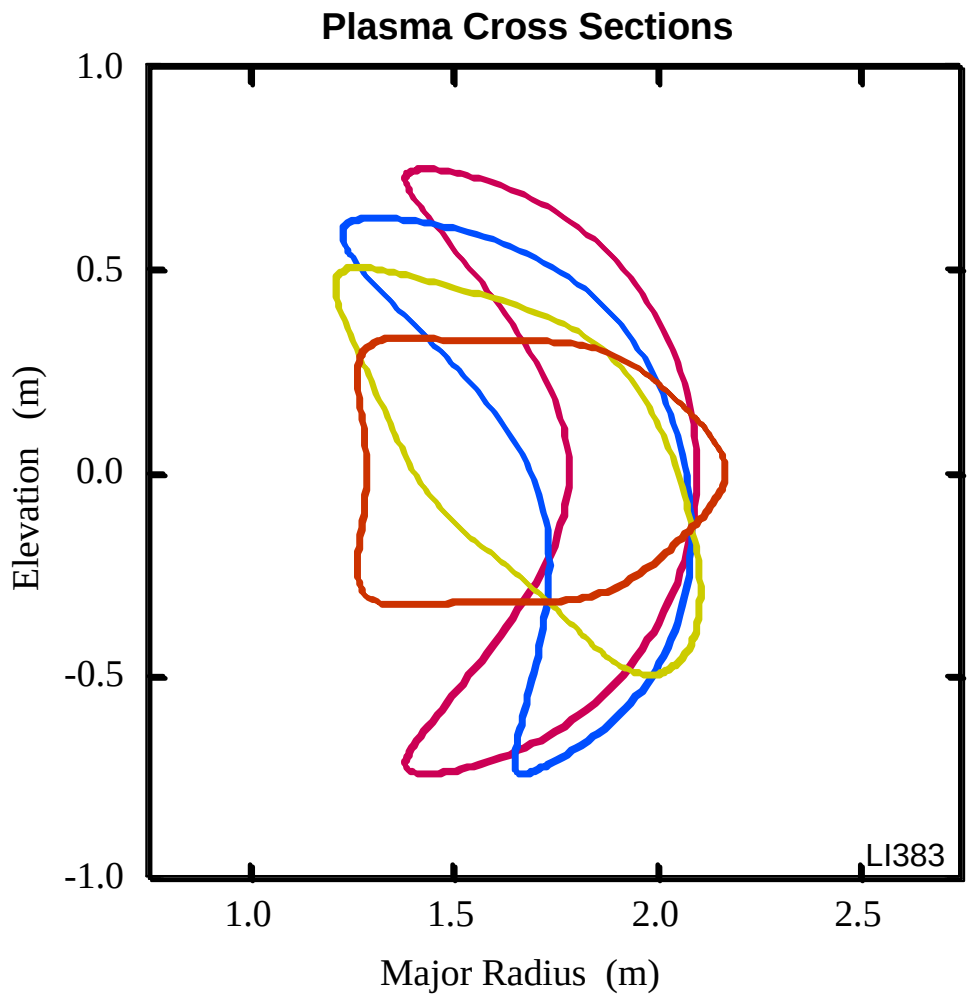
- Steady state without current drive. Control of iota (q) and shear via coils

$\rightarrow$ **NCSX Design Goals**

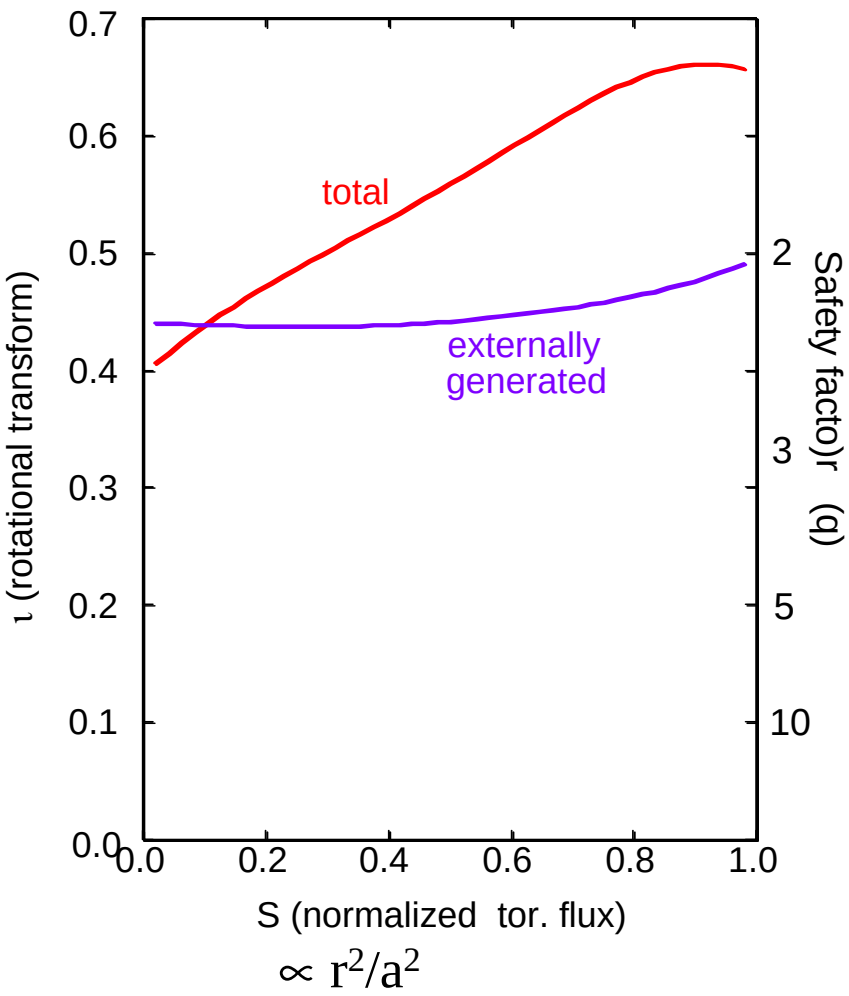
Using Advances in Theory and Numerical modeling; parallel computing
(NERSC, ACL/LANL, Princeton/PPPL)

NCSX Plasma Configuration Has Attractive Physics

- 3 periods, R/⟨a⟩=4.4, ⟨κ⟩~1.8
  <indented>
- Good magnetic surfaces.
- Quasi-axisymmetric: low helical ripple transport.
- Stable at β=4.1% to kink, ballooning, kink, vertical, Mercier, neoclassical-tearing modes
- ~75% of the magnetic transform (B_P) is from external coils at β=4%. Remainder from boostrap current
- Proposed device has R = 1.4 m, B ranging from 1.2 - 1.7T at full iota > 2 T at reduced iota



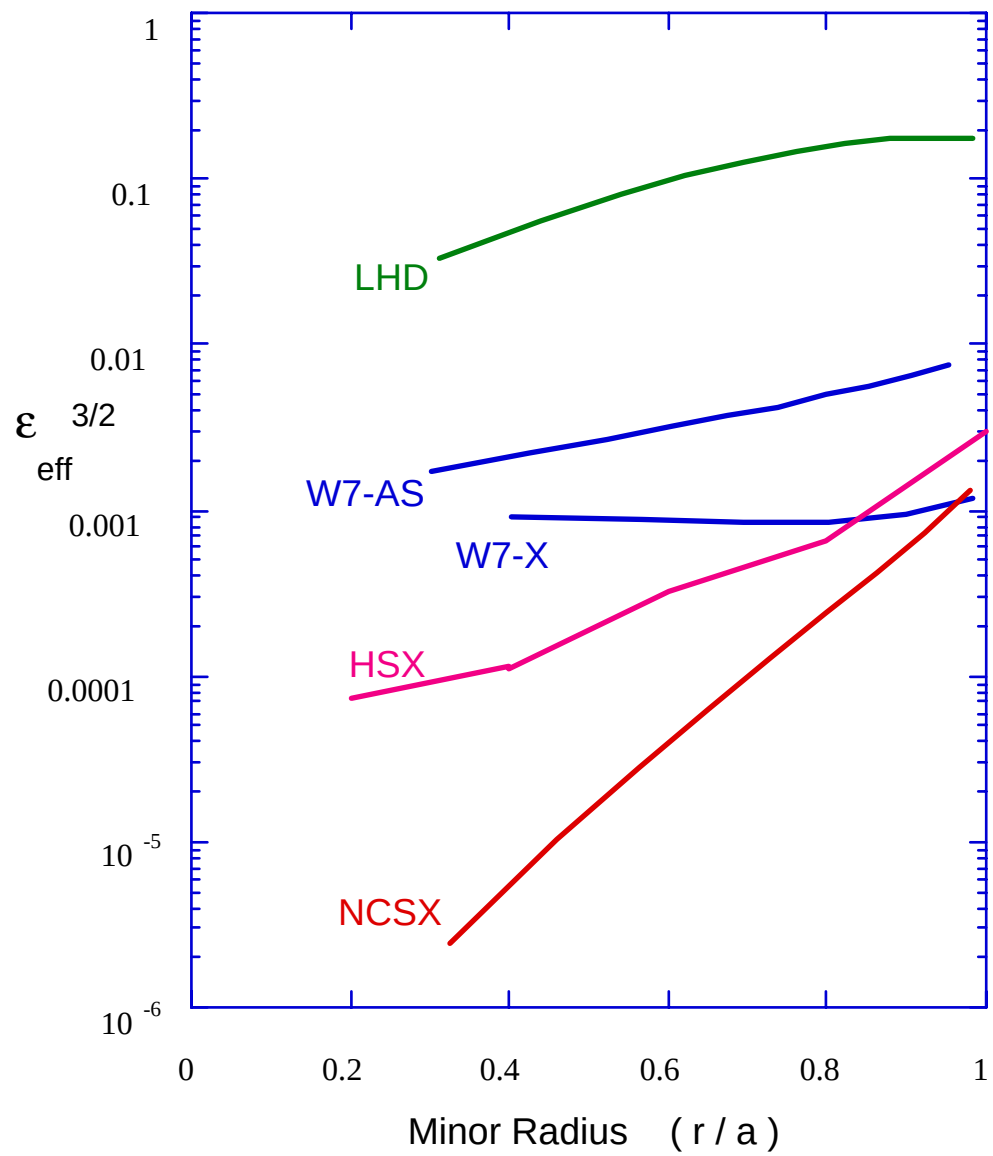
Hybrid Configuration Combines Externally-Generated Fields with Bootstrap Current



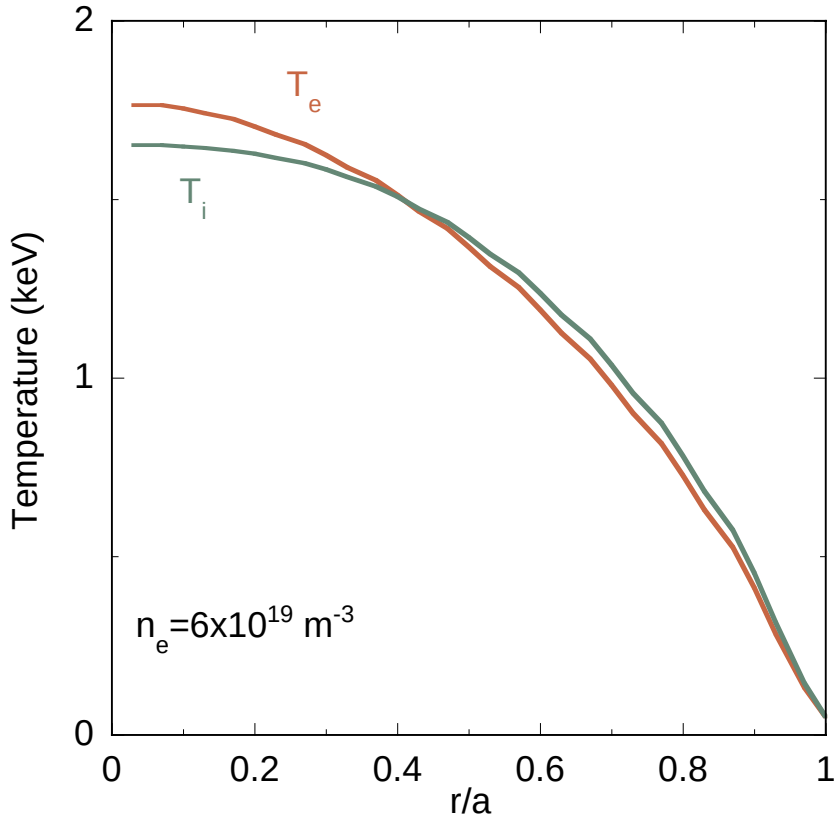
- Quasi-axisymmetry
  ⇒ tokamak like bootstrap current
- ~3/4 of transform (poloidal-B) from external coils ⇒ externally controllable
- at R=1.4m, B=1.2T, I_P = 125 kA;
  I_P^{Equiv} = 500kA; β=4.1%
  ⇒ β_N^{Equiv} = 2.5
- 'Reversed shear' stabilizes neoclassical tearing and trapped particle modes ⇒ reduced turbulence drive as in reversed shear tokamak regimes?

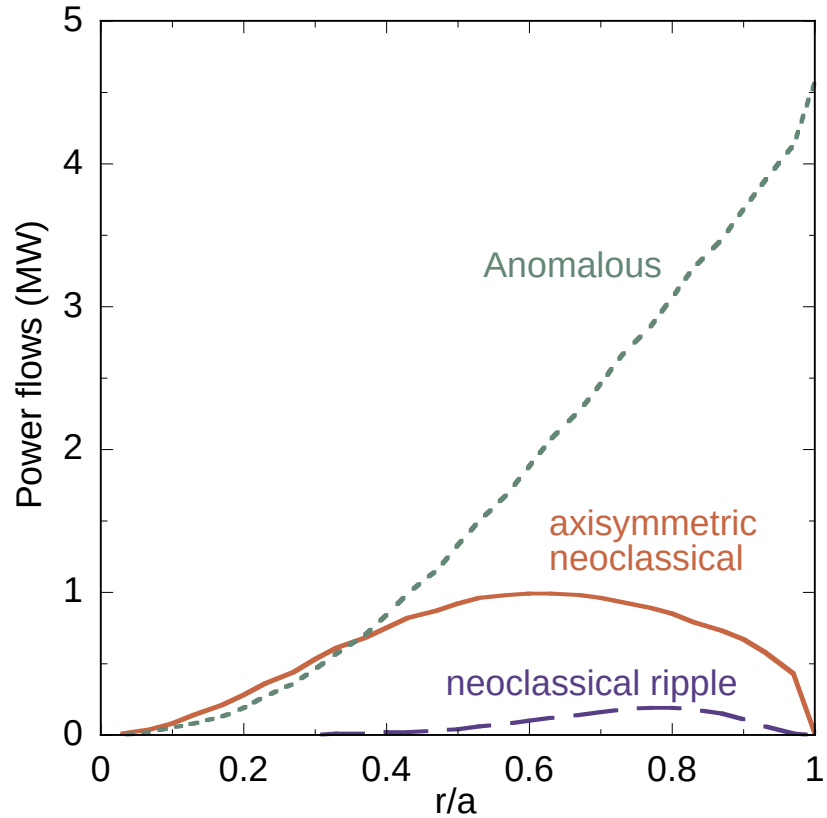
Quasi-Axisymmetric: Very Low effective ripple

- ε_{eff} from NEO code by Nemov-Kernbichler
- In 1/ν regime, neoclassical transport scales as $\varepsilon_{eff}^{3/2}$
- Edge ε_{eff} ~ 1.2%, < 0.1% in core (bug corrected in NEO)
- Allows balanced-NBI 24% loss at 1.2T, drops as B↑
- Should give low flow-damping
  - manipulation of flows for flow-shear stabilization
  - zonal flows like tokamaks
- Linear microstability being studied See E. Belli LP1.077



Low $\varepsilon_{h,eff}$ $\Rightarrow$ Low Ripple Transport



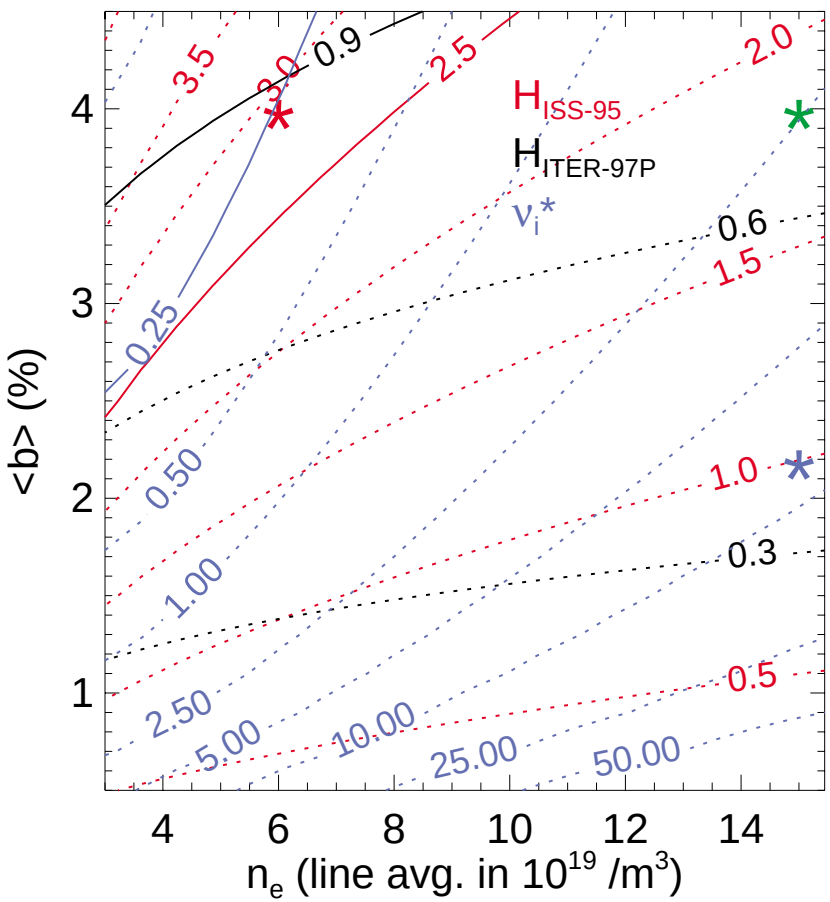


- Helical transport is negligible with self-consistent E_r
- β = 4%, ν^* ~ 0.25 with B=1.2 T, P_{inj}=6 MW, n_e = 6 x 10^{19} m^{-3} requires H_{ISS95}=2.9 or $H_{ITER\text{-}97P}$=0.9 B=1.7T gives access to ν^* ~ 0.1, $T_i(0)$~2.3 keV
- Shaing-Houlberg for helical transport, benchmarked with Monte-Carlo.
- Uniform anomalous χ used. Similar results obtained with Lackner-Gottardi

Wide Range of Plasmas Accessible

Contours of H_{ISS95}, $H_{ITER\text{-}97P}$, and min ν_{*i}

P_{inj}=6.0 MW



B = 1.2 T

- β=4%, ν_{*i} =0.25 requires H_{ISS95}=2.9, $H_{ITER\text{-}97P}$=0.9
- β=4% at Sudo 'density-limit' requires H_{ISS95}=1.8
- H_{ISS95}=1.0 gives β=2.2% sufficient to test stability theory

- 3MW gives β=2.7%, ν_{*i} =0.25 with H_{ISS95}=2.9;
  β=1.4% with H_{ISS95}=1.0 sufficient to test stability theory

LHD and W7-AS have achieved H_{ISS95} ~ 2.5

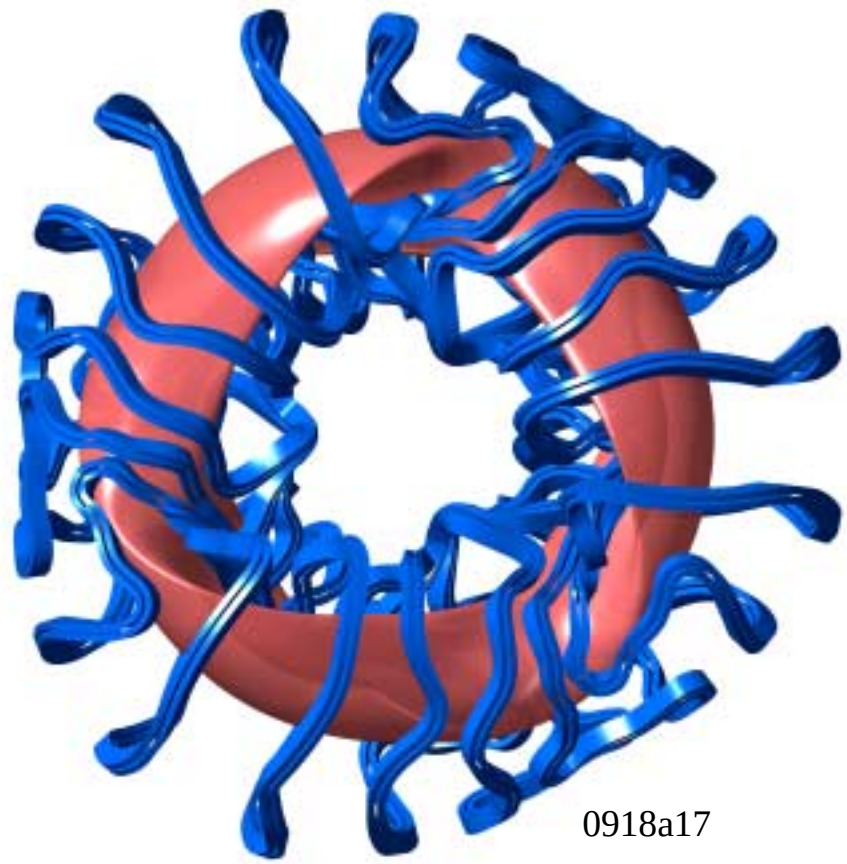
PBX-M obtained β = 6.8% with $H_{ITER\text{-}97P}$ = 1.7 and H_{ISS95} ~ 3.9

Coil Design

- Coils designed to match normal B on plasma boundary (similar to process on W7-X and HSX)
- Many coil topologies have been explored: helical, saddle, modular (distorted TF)
- Modular coil sets have been designed which reproduce the physics properties of the fixed boundary design, with either 18 or 21 coils.
- Still adapting the coil designs to meet engineering constraints
  - See H. Neilson CP1. 034

NCSX Modular Coils Provide Good Physics Capability

- Modular coils best preserve physics properties of reference plasma:
  - stable at reference β (4%).
  - Good magnetic surfaces.
- Also include Poloidal Field coils and weak Toroidal Field, for flexibility



- Stable to $\beta > 6.5\%$ with some increase in ripple
- Passive disruption stability ?

equilibrium maintained even with total loss of β or I_P.

See H. Neilson, CP1.034

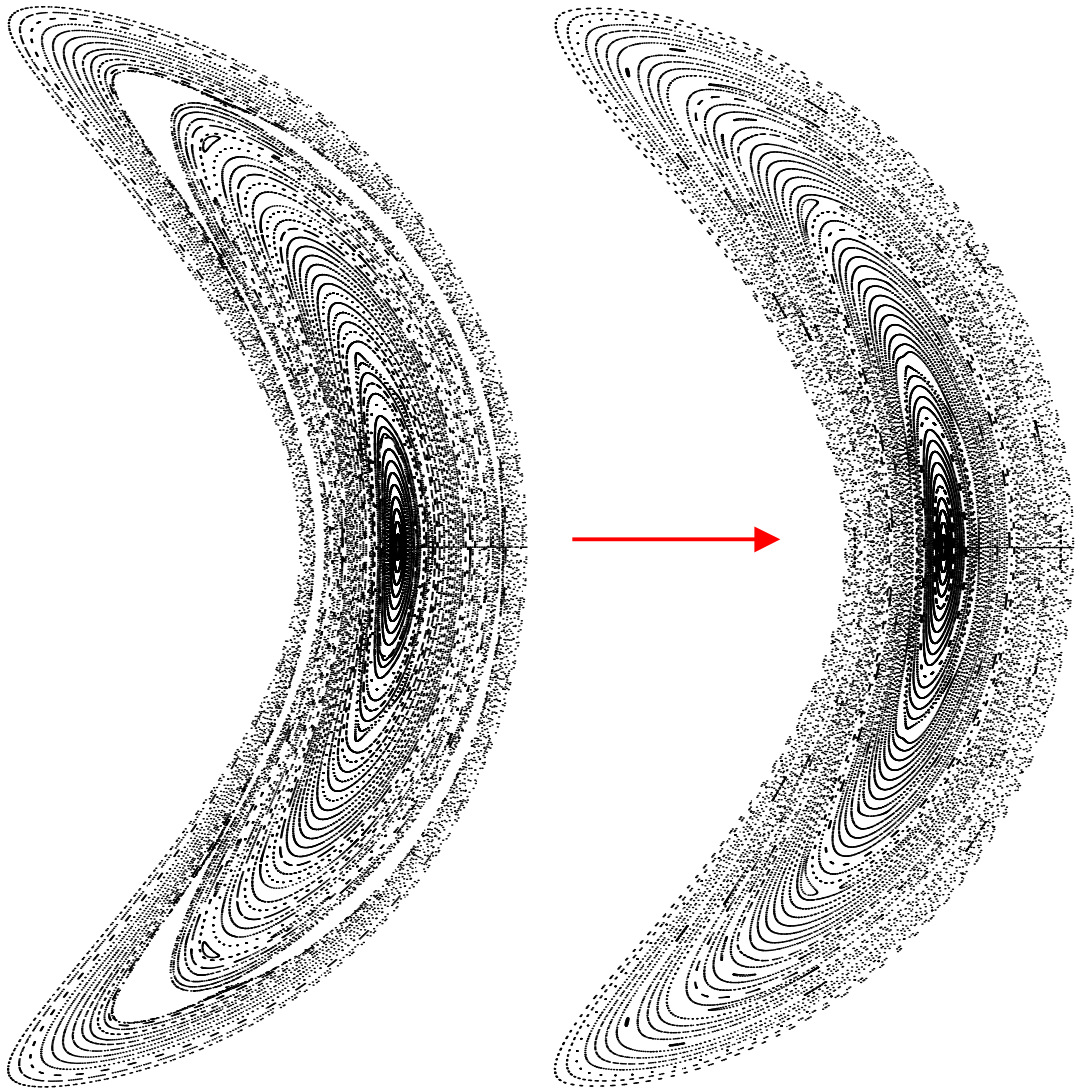
Multiple Methods used to Produce Good Flux Surfaces

Flux surface quality is a significant issue for 3D systems.

NCSX design uses multiple methods to attempt to ensure good flux surfaces:

- Explicit design to eliminate resonant fields, in both fixed boundary target plasma, and in coil designs
- ‘Reversed shear’ configuration $\Rightarrow$ neoclassical healing of equilibrium islands and stabilization of tearing modes
- Trim coil arrays targetting low-order resonances

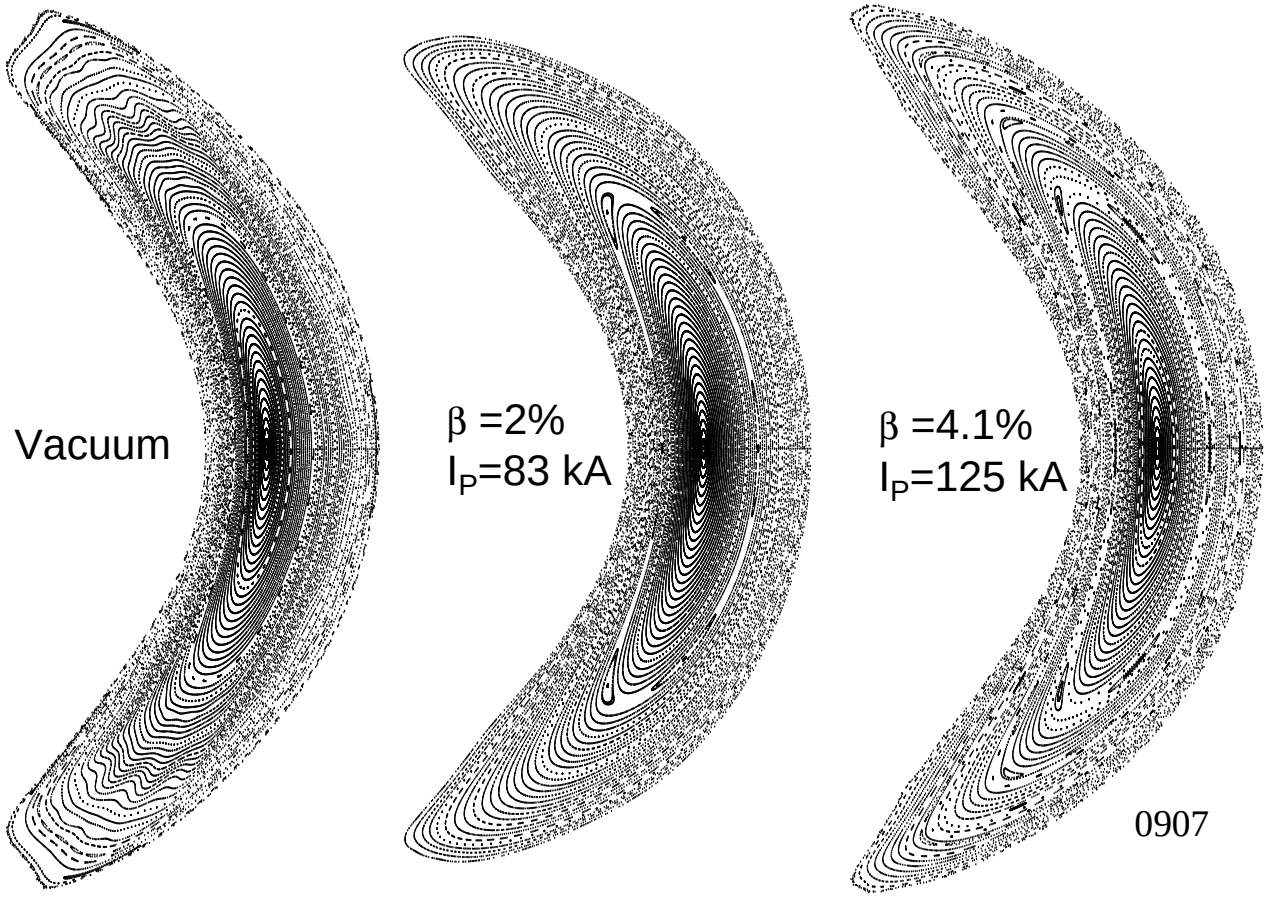
Island Removal Method



- Calculate coupling between plasma boundary shape and island widths by perturbation, using PIES
- Invert coupling matrix to find (small) shape modification to remove islands
- Modification had no effect on calculated stability or transport
- In experiment, neoclassical effects should heal islands

See S. Hudson, LP1.075 and A. Reiman, LP1.076

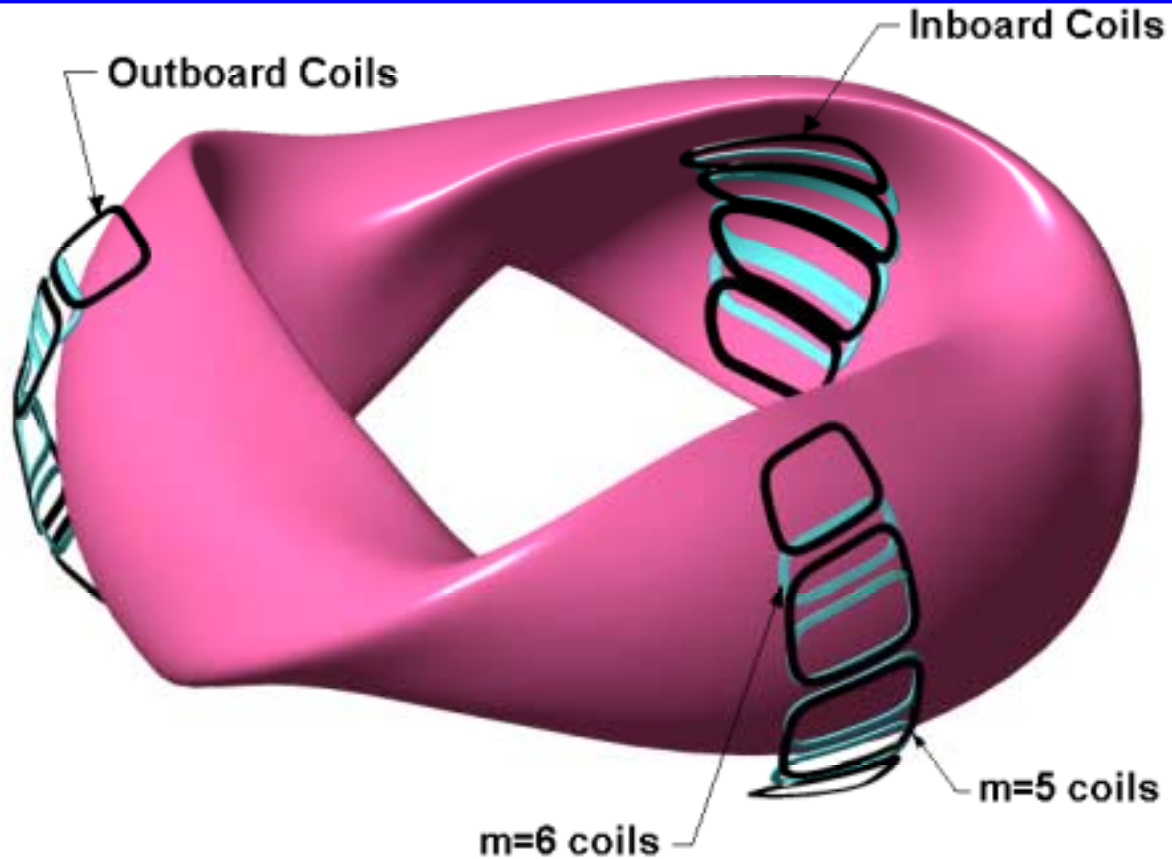
Coils Produce Good Flux Surfaces



- Free-boundary equilibria (PIES)
- I_P values for B_T=1.2 T
- Coils designed to produce good surfaces at full current.
  Island in middle case can be eliminated with trim coils.

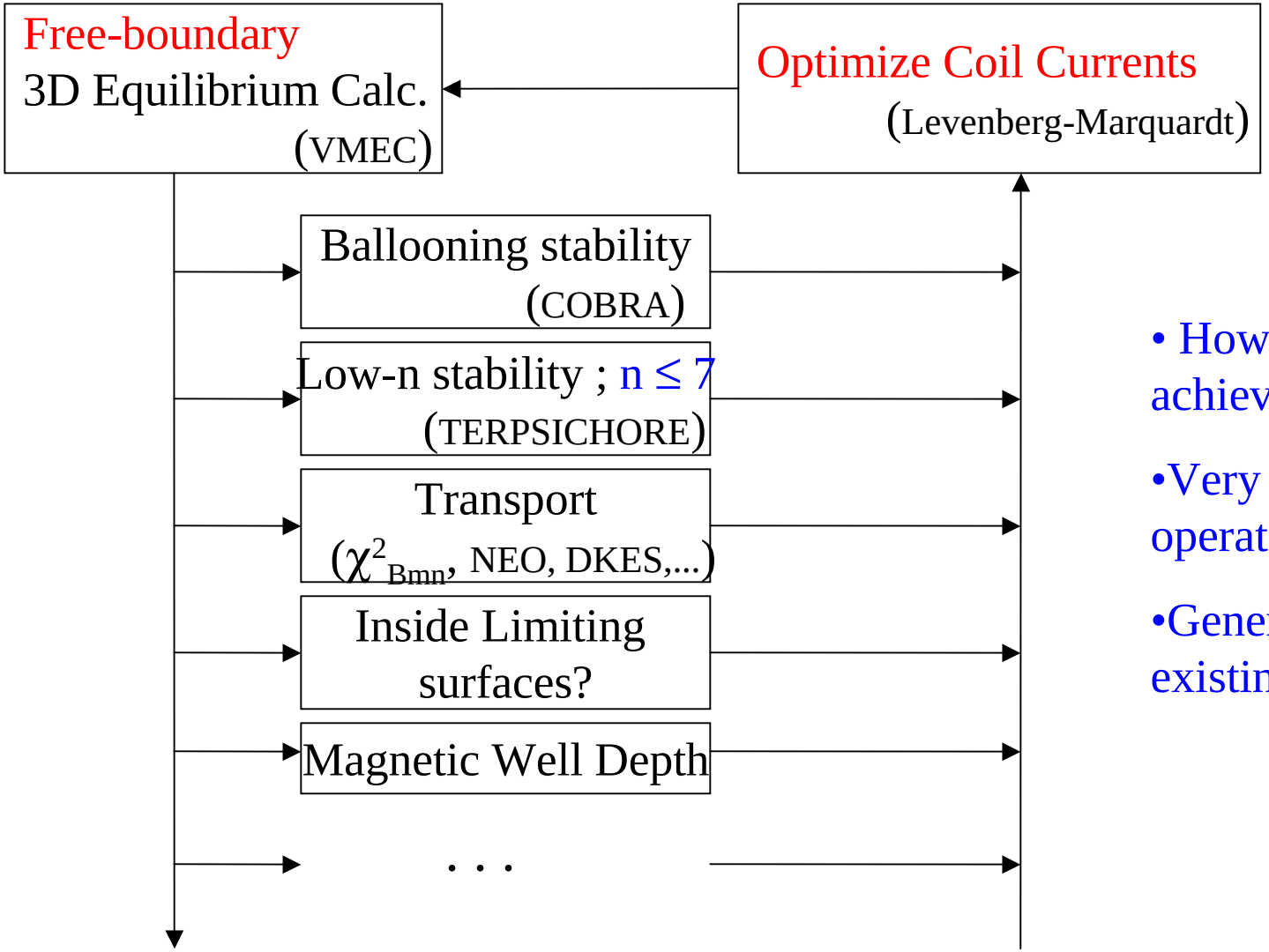
See S. Hudson, LP1.075
and A. Reiman, LP1.076

Trim Coil Array Controls Equilibrium Islands



- For suppression of equilibrium islands over wide range of iota and shear profiles
- For controlled generation of islands to test neoclassical tearing theory
- Tested on vacuum and finite β configurations.

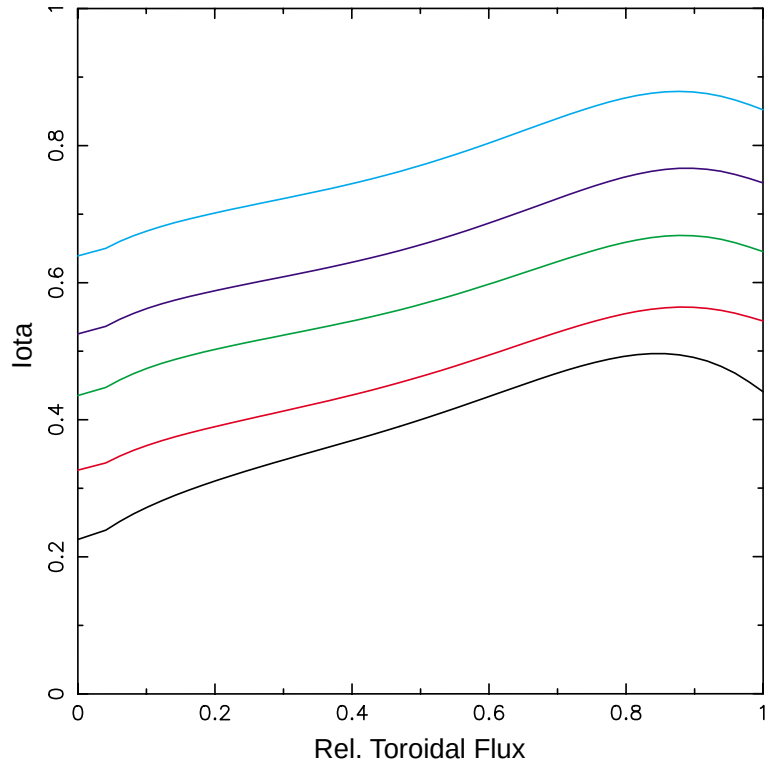
Free-Boundary Optimizer used to Assess Flexibility



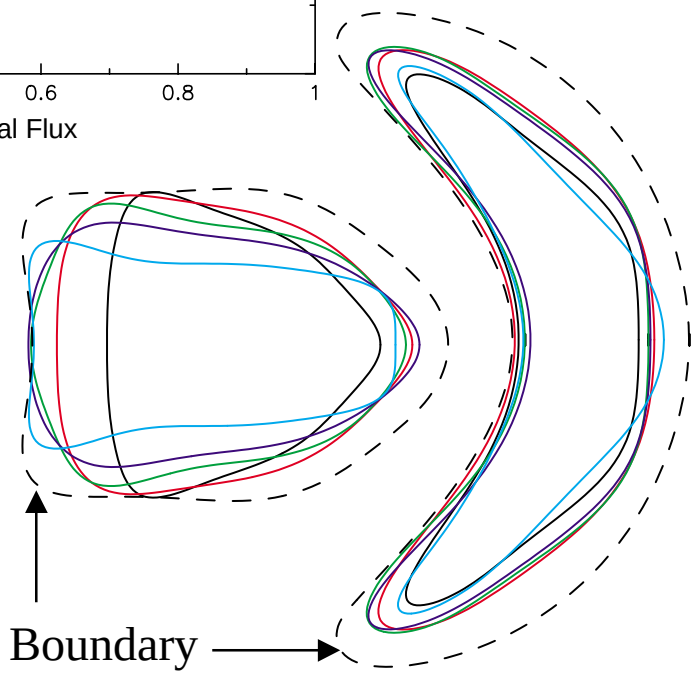
- How to use coils to achieve physics goals?
- Very similar to how one operates an experiment
- General tool, useful on existing experiments

Modular Coils are Flexibile

- External rotational transform controlled by plasma shape at fixed plasma current & profile.
- Can adjust to avoid iota=0.5, or hit it
- Can externally control shear
- Can stabley accommodate wide range of p,j profiles
- Can use to test stability, island effects
- Results shown for 1017 coil design, similar results obtained for newer designs

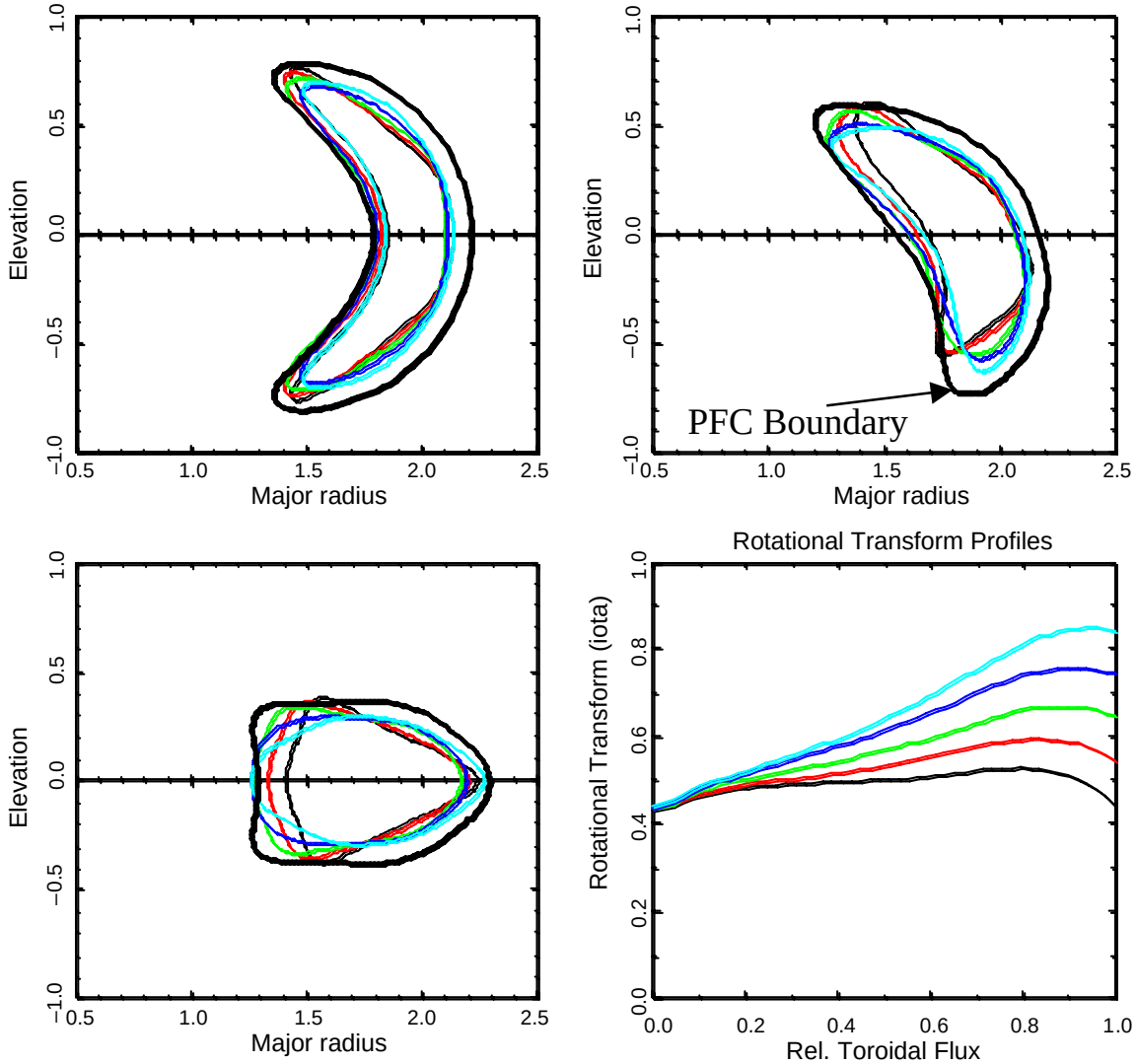


β=0, full current



Modular Coils give External Shear Control

- Coil-generated rotational transform decouples shear from plasma current profile
- Can control magnetic shear at fixed plasma current and profiles
- Allows controlled study of shear effects , e.g., kink-stabilization physics; turbulent transport.

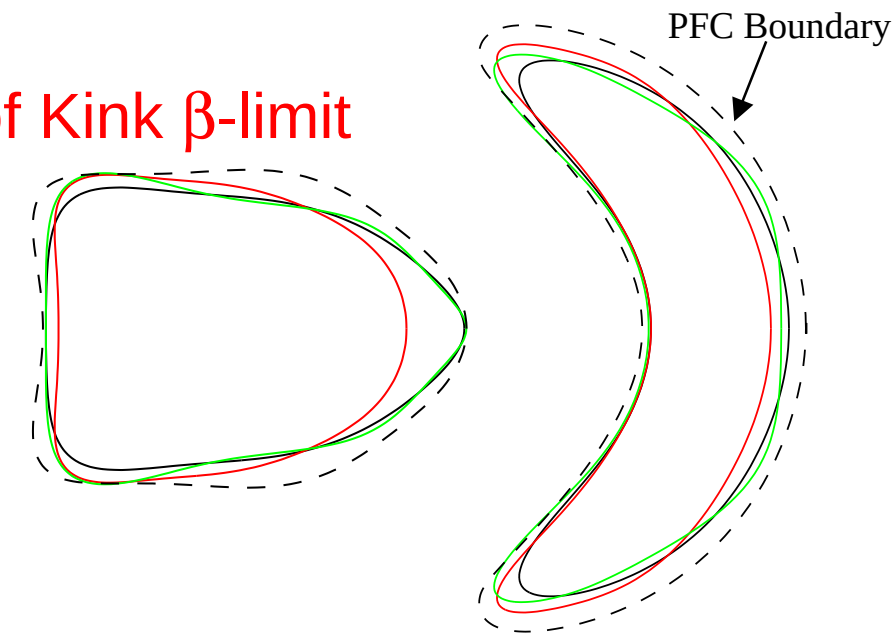


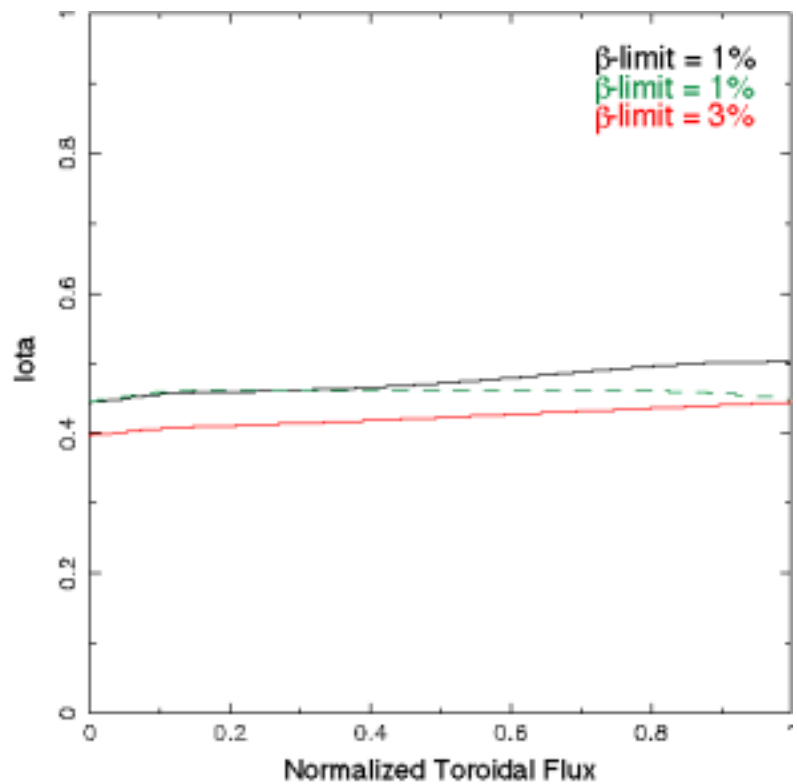
β=0, full current

Coil Flexibility Gives Control of Kink β-limit

- External-kink marginally stable β changed from 3% to 1% by modifying plasma shape
  - either at fixed shear or fixed edge-iota !
- Free-boundary equilibria, fixed pressure and current profiles
- Useful for testing understanding of 3D effects in theory & determining role of iota-profile

- Similarly, can find stable equilibria with effective ripple varying by factor ~ 5. For testing transport optimization & flow damping

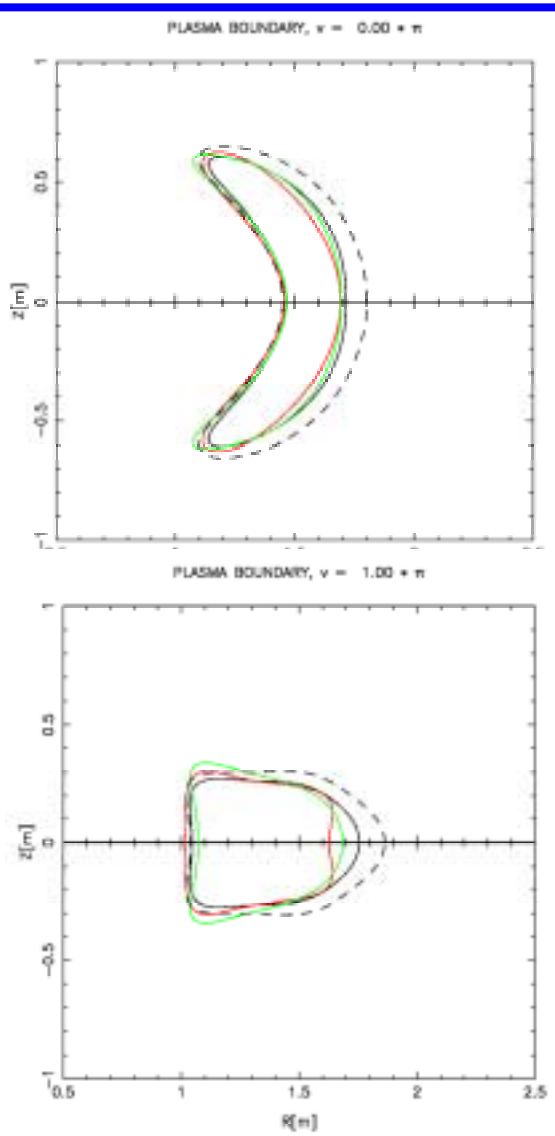




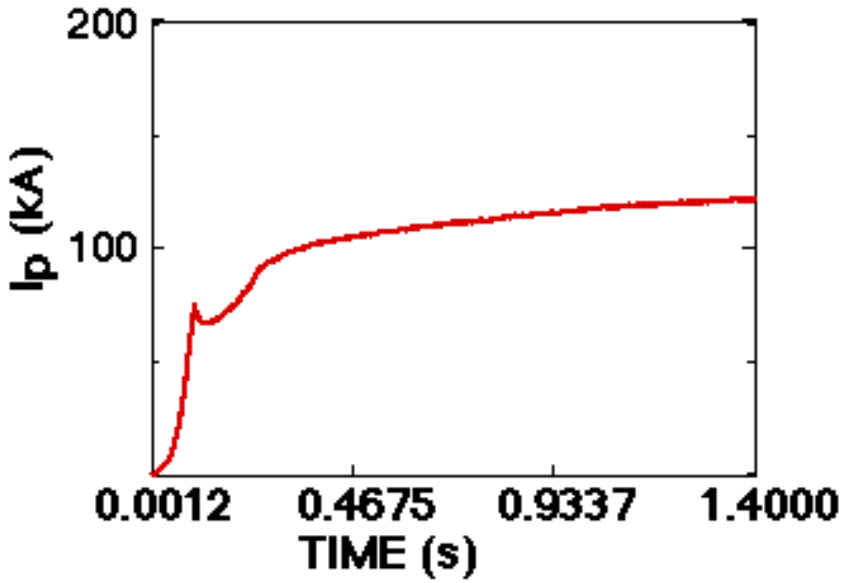
Equilibrium Maintained even with Loss of I_P or β

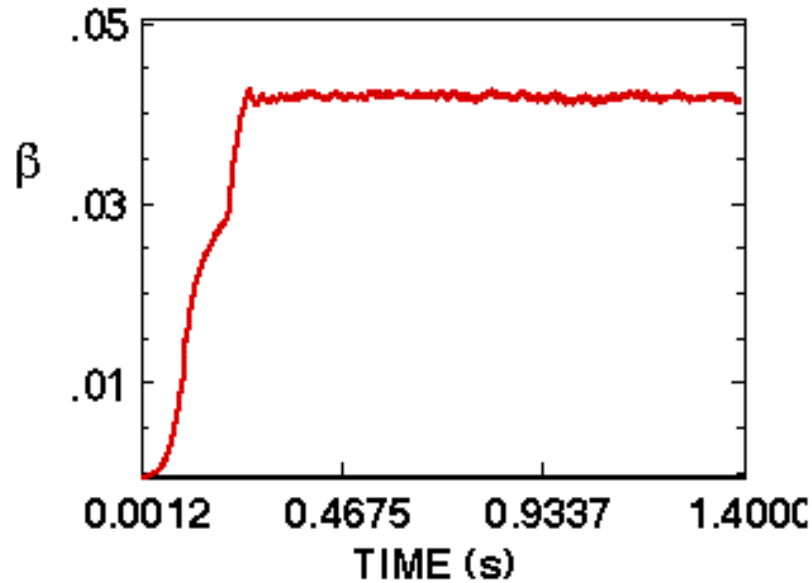
- Total loss of I_P or β only causes a small shift in equilibrium (few cm), for fixed coil currents.
- For comparable tokamak,

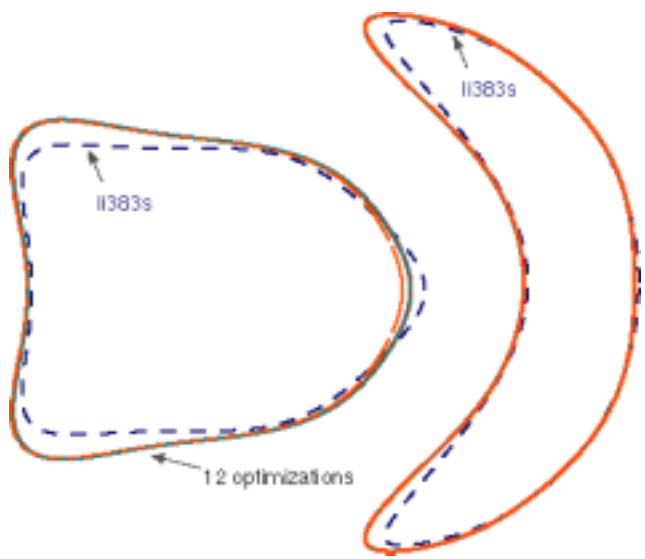
  loss of $\beta \Rightarrow$ radial shift of ~ 30cm. Similar shift for ~ 20% drop in I_P.
- Any NCSX disruptions will not lose radial equilibrium, should give unique insight into tokamak disruption dynamics.
- Possibility of passive disruption stability!



Modeling of Evolution Shows Stable Access to High-β







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- Stable evolutions calculated for high and low temperatures (confinement)
- Stable evolution with
  - iota(a) crossing 1/2
  - iota(a) always > 1/2

See E. Lazarus, CP1.035

Motivation and Goals of the Plasma Boundary Program

NCSX

General Goal:

- **Support the mission and objectives of NCSX and provide the necessary boundary conditions for improved confinement, high beta, and pulse length.**

Power Handling:

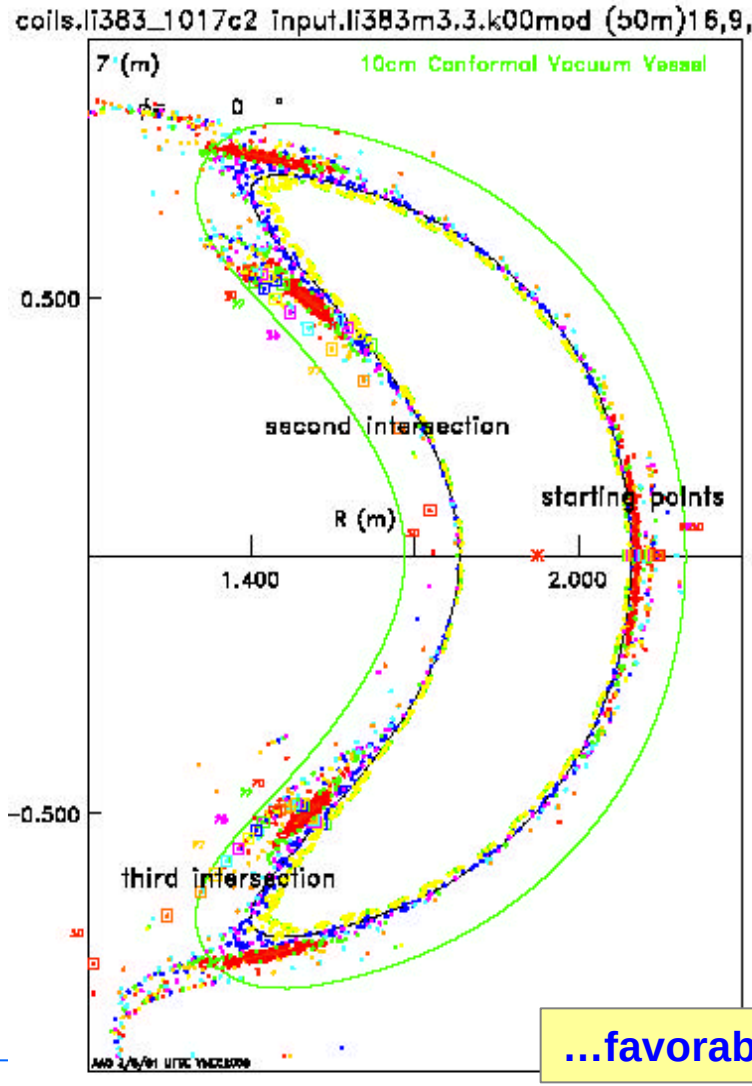
- **Initial PFC design for 3-6 MW and 0.5s. Vacuum vessel shaped to allow long connection lengths (>100m between LCMS and divertor plates). Design of divertor plates guided by "footprint" plots for a range of configurations (range in beta and iota).**

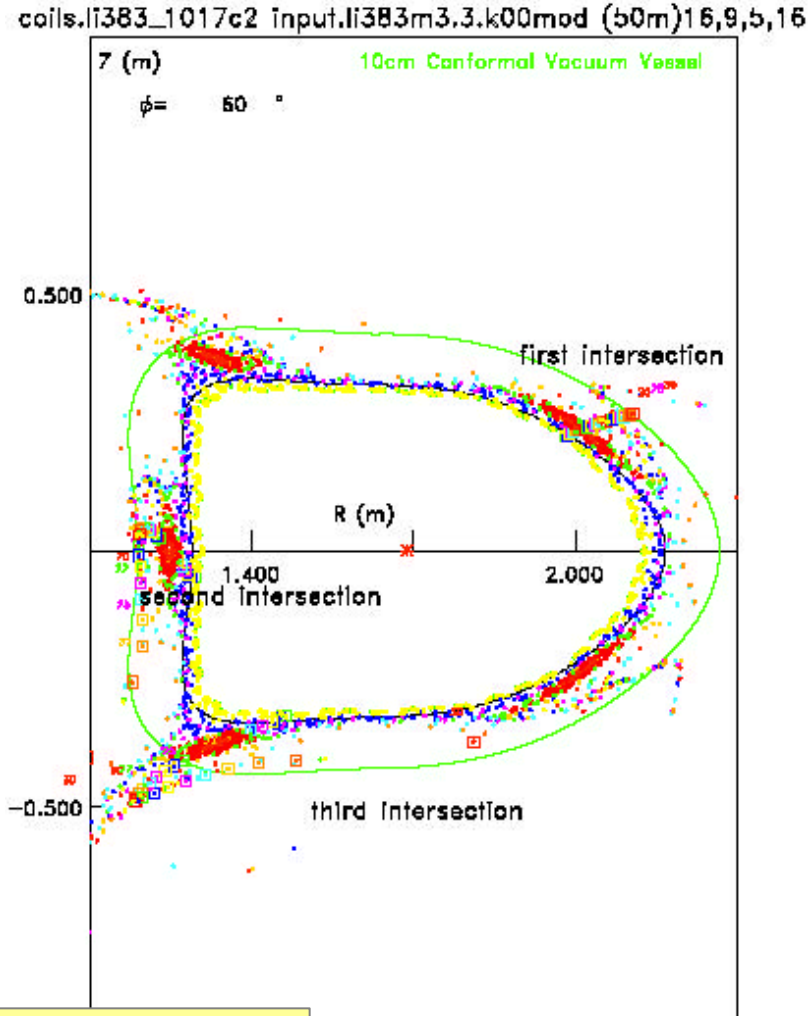
Particle Control:

- **Special emphasis on divertor baffle design to confine neutrals in the divertor area and minimize neutrals densities in the f=0 midplane.**

Poincaré Plots Including Trim Coils for Island-Healing (Inside) Reveal More-Pronounced Island Structure Outside the LCMS

NCSX





…favorable for island divertor!

See A. Grossman CP1.036

Phased Approach for Plasma-Facing Components: Future Upgrades

NCSX

Phase I :

- **3 divertor baffles = conformal wall armor around f = 0° cross-section and**
- **3 inboard limiters 'centered' around f = 60°**

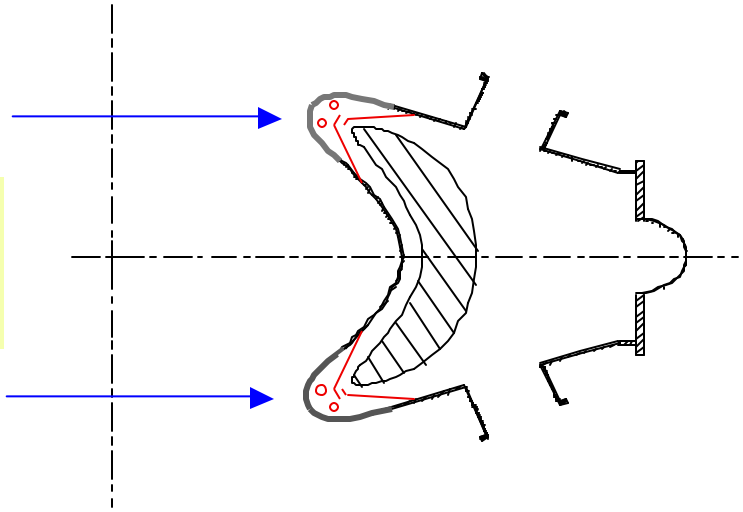
Phase II

- **baffles optimized for recycling control and impurity control, based on neutrals and impurity transport modeling.**

Phase III

- **divertor upgrades include optimized divertor baffling and divertor pumping.**

Neutrals control via optimized baffles and divertor pumping



Modelling Efforts Make Use of International and Domestic Developments in the Stellarator Program

NCSX

- **Magnetic Boundary Topology**

  VEMEC: equilibrium with nested surfaces inside LCMS
  MFBE: finite-beta, free-boundary, with toroidal currents
  Gourdon: field-line tracing code for Poincaré plots

- **Neutrals modeling**
  - start with 2-D and DEGAS 2 with given plasma background; this provides first estimates of neutrals; later full 3-D

- **3-D plasma edge code development**
  - collaboration with Greifswald (BoRiS-code)
  - we will adapt the BoRiS-code to the NCSX geometry and perform self-consistent edge plasma calculations

- **Turbulent transport**
  - 3-D turbulence code for tokamaks: BOUT-code; Greifswald group plans to help modify BOUT for stellarators

See A. Koniges CP1.035

NCSX Research Advances Fusion Science in Unique Ways

- Can limiting instabilities, such as external kinks and neoclassical tearing modes, be stabilized by external transform and 3D shaping? How are the non-linear dynamics and disruptions affected? How much external transform is enough?

- Can the collisionless orbit losses from 3D fields be reduced by designing the magnetic field to be quasi-axisymmetric? Is flow damping reduced?

- Do anomalous transport reduction mechanisms that work in tokamaks transfer to quasi-axisymmetric stellarators? How much effective-ripple is too much?

- How do stellarator characteristics such as 3D shape, islands and stochasticity affect the boundary plasma and plasma-material interactions?

NCSX provides unique knobs to understand toroidal confinement fundamentals: rotational transform, shaping, magnetic symmetry.

Plans for Proceeding

- **Proposed milestones:**

– Conceptual design review (CDR), April, 2002.
– Start Title I design, October, 2002.
– Start fabrication, October, 2003.
– First plasma, March, 2007

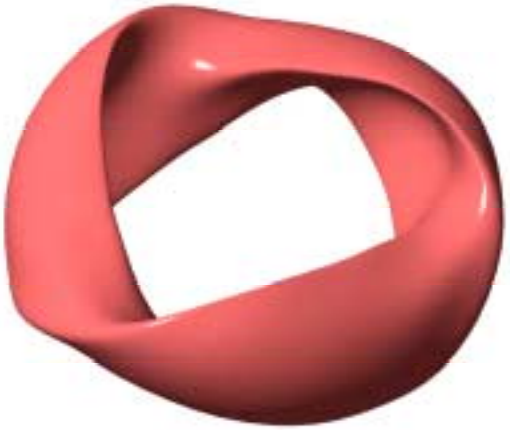
Bottoms-up cost and schedule will be developed for CDR.

- **NCSX is using a proven project management approach**

– similar to NSTX and TFTR D&D; both ahead of schedule and on budget.
– Follows DOE project management guidelines and orders

- **NCSX will continue national team approach through all phases**

– Combines the best talents and experience of DOE Labs and Universities
– Integrated team led by PPPL and ORNL, with numerous collaborators; similar to NSTX.

Conclusions

- NCSX is an exciting opportunity for unique fusion science.
  - Stabilize high-β modes with 3D shaping.
  - Transport in low-collisionality quasi-axisymmetric system.
- NCSX provides innovative solutions to make magnetic fusion more attractive.
  - Combine best characteristics of stellarators and tokamaks.
  - Possibly eliminate disruptions; intrinsically steady state
- Strong linkages with all of magnetic fusion science, complementing other toroidal confinement research programs.
- Physics basis for NCSX is sound, attractive configuration identified
  - passive stability to kink, ballooning, vertical, Mercier, neoclassical tearing with $\beta > 4\%$
  - very good quasi-axisymmetry
  - possible passive disruption stability
  - Robust, flexible coil system
- Good plans are in place for going forward.