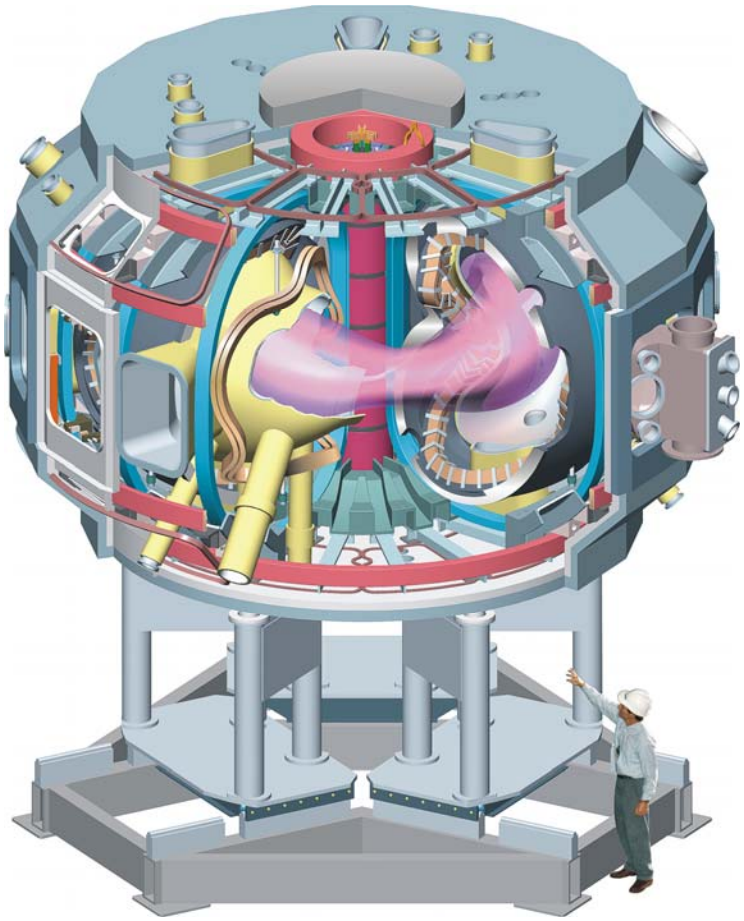
The Role of NCSX In the World Fusion Program

M.C. Zarnstorff
For the NCSX Team

FESAC Scientific and Programmatic Review of NCSX
15 September 2007

Outline

- Motivation, Strategy, and Goals
  - No disruptions. No external current drive. Higher pressure. QA to build on ITER.
- NCSX Unique Characteristics
  - 3D QA shaping, PoP scale experiment
  - Confinement, stability, surfaces, divertor, energetic particles.
- NCSX Research Plan
  - Near and longer term
- Implications of the Loss of NCSX



Motivation: What must we learn for DEMO?

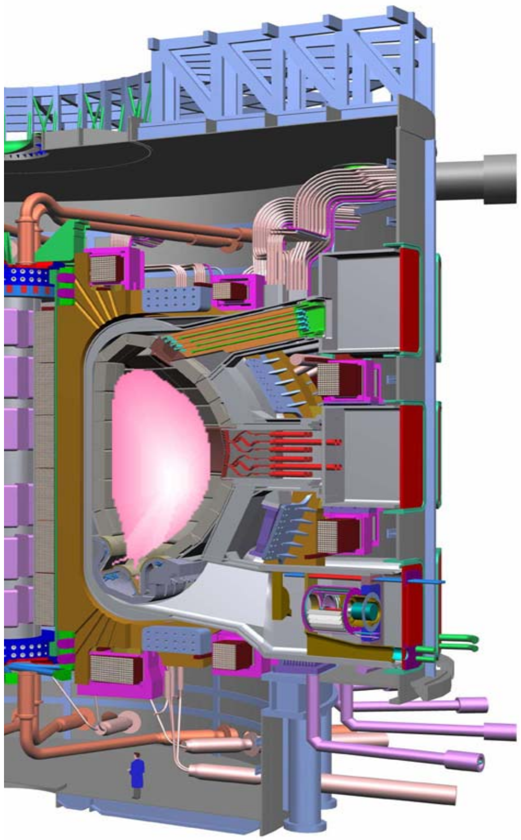
ITER: 500 MW for 400s, gain > 10

DEMO: ~2500 MW, continuous, gain > 25,
~ same size and field.

This requires plasmas with:

- Higher pressure, by at least factor of 2.2
- Less externally driven current
  No inductive current
- Essentially no disruptions or ELMs
- Stable confinement of α-particles
- And: high heat flux PFCs, T-breeding cycle, long-lived materials…

3D Shaping offers solutions to these plasma issues



ITER (~ 2016)

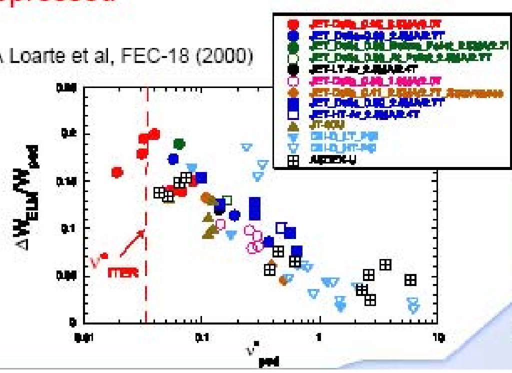
"Disruptions Will Essentially Have to be Eliminated"

EFDA
EUROPEAN FUSION DEVELOPMENT AGREEMENT

Power Exhaust/ Impurities III

- **Transient events are of even greater significance than in ITER**
  - availability and first wall lifetime considerations set severe limitations on frequency and magnitude of pulsed events
- **Disruptions will essentially have to be eliminated**
  - typical estimates in literature set frequency at 0.1 -1 per year
  - issues:
    - thermal quench: ~ 1GJ
    - current quench: ~ 1GJ
    - runaway electrons: >10MA if not suppressed
- **ELMs too will essentially have to be eliminated**
  - ELM-enhanced erosion might already set PFC lifetime limits in ITER
  - ⇒ ELM control/ suppression techniques

A Loarte et al, FEC-18 (2000)



D. Campbell et al.
IAEA 2006, Chendu
"Critical Issues for Tokamak Power Plants"

Aries studies: disruptions must be < 1 per year

- Breeding blanket designs have first-wall thickness of 1.5 – 5 mm insufficient thermal mass to avoid melting, doesn't stop runaways.
- *How to guarantee no disruptions at high beta in bootstrap sustained configurations?*

NCSX Strategy: Use 3D QA shaping Combine Advances of Stellarators and Tokamaks

Tokamaks:

- Compact $\rightarrow$ cost-effective, project to high power density
- Importance of flows (including self-generated) for turbulence stabilization
- 'Reversed shear' to reduce turbulence, increase stability, suppress islands
- ITER results at the reactor scale (particularly ρ^* scaling).

Stellarators:

- 3D shaping, from externally-generated helical fields
  - No external current drive.
  - Robust stability. Generally disruption-free.
- Quasi-axisymmetric shaping (QA) to keep tokamak-like field structure
- Optimization of 3D QA shaping to obtain desired properties
  - Increased stability, good flux surfaces at high-beta
  - Very good QA: very low effective ripple, orbits and thus neoclassical transport very similar to tokamaks, low rotation damping

Goal: Steady-state high-β, good confinement without disruptions

NCSX Research Goals

Acquire the physics data needed to assess the attractiveness of
3D QA shaping to control high-beta plasma stability, confinement, sustainment

Understand effect of 3D QA shaping on

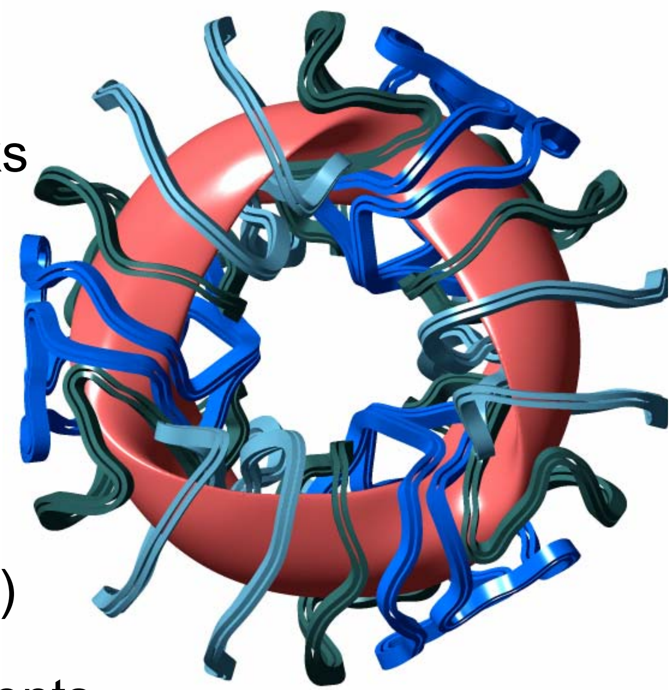
- Pressure limits and limiting mechanisms
- Disruptions and operating limits.
- Transport and confinement with low QA ripple.
  Relationship between NCSX and tokamak transport.
- Equilibrium islands and tearing-mode stabilization, role of reversed magnetic shear.
- Divertor operation, compatibility with good core performance.
- Energetic-ion stability and confinement

Determine degree of 3D QA shaping required to achieve

- High β, good confinement, compatible with steady state, without disruption risk
- Can the design constraints be simplified, with improved understanding?
  $\rightarrow$ Simpler engineering design

What is the best design for DEMO, in continuum between AT and NCSX ?

NCSX Designed to Integrate Attractive Properties of Low Aspect Ratio QA & Address Goals

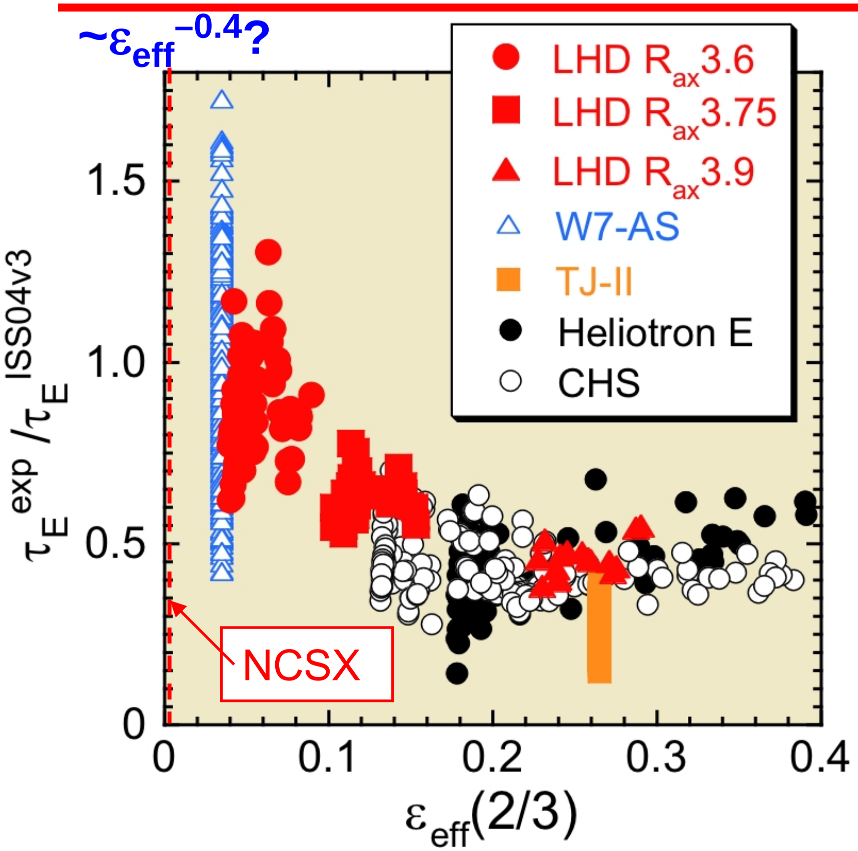


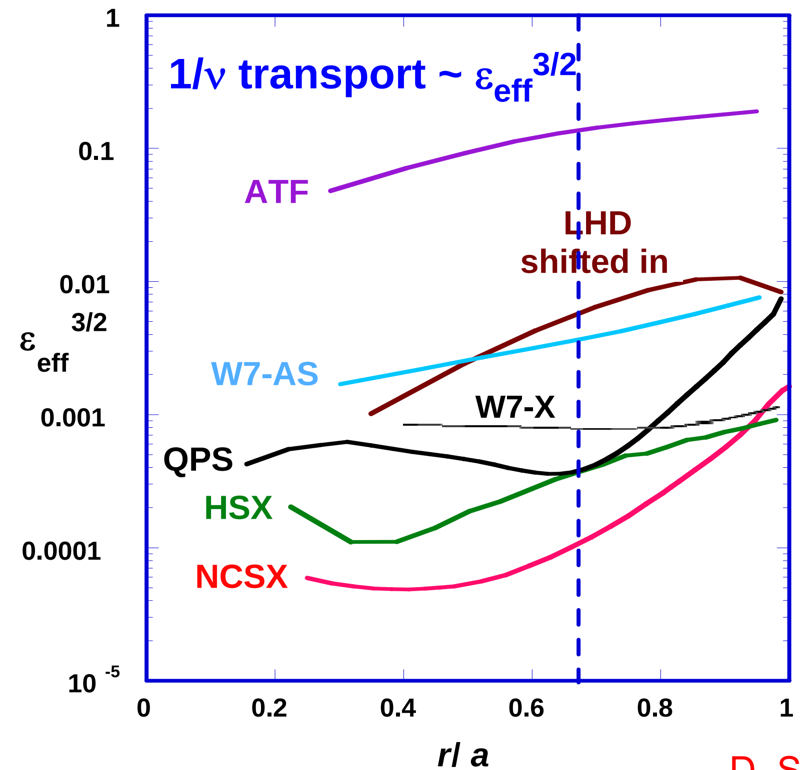
- 3 periods, $R/\langle a \rangle = 4.4$, $\langle \kappa \rangle \sim 1.8$, $\langle \delta \rangle \sim 1$
- Quasi-axisymmetric: transport similar to tokamaks
  ripple thermal transport insignificant.
- Passively stable at β=4.1% to kink, ballooning,
  vertical, Mercier, neoclassical-tearing modes
  (steady-state AT β limit ~ 2.7% without feedback)
- Stable for at least β > 6.5% by adjusting coil currents
- Passive disruption stability: equilibrium maintained
  even with total loss of β or I_P

G.-Y. Fu
L.P. Ku
A. Reiman
M. Zarnstorff

These characteristics are unique in the world fusion program
NCSX is only experiment studying QA and synergy with tokamak

NCSX Has Lowest Ripple of All Stellarators





D. Spong

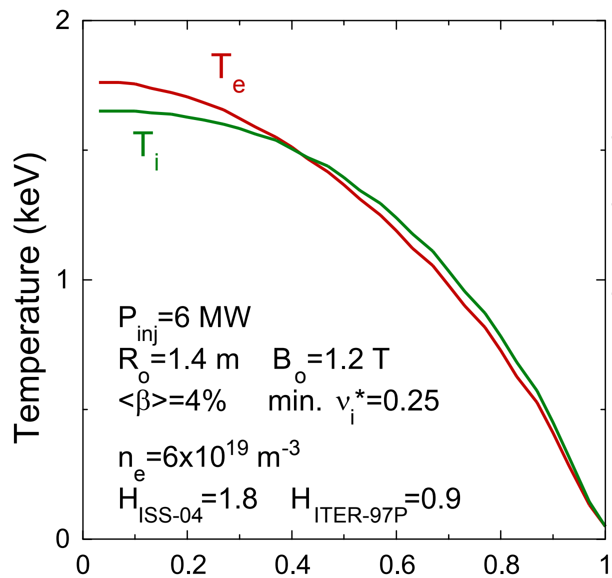
- Low ripple reduces rotation damping.
  Produces tokamak-like zonal flow damping (Mynick).
- NCSX can access ε_{eff} a factor of 3 lower, or a factor of 10 higher
- New global confinement scaling for stellarators (ISS04v3) found strong dependence on ripple magnitude. Must involve anomalous transport also.

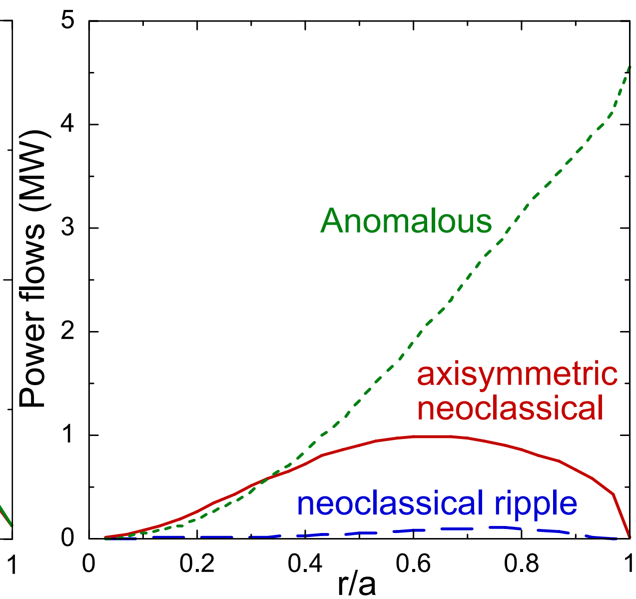
Low Ripple $\Rightarrow$ Negligible Ripple Transport

Ion-root $1/\nu$ regime:

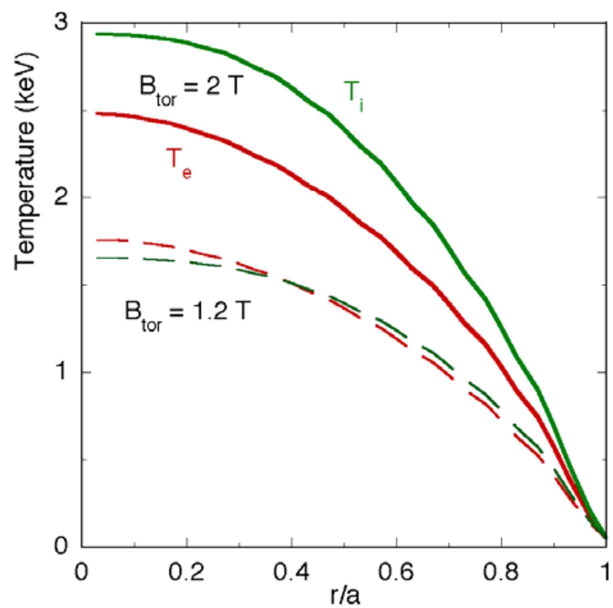
$Q_{ripple} \propto \varepsilon_{eff}^{1.5}\, T^{4.5}$

Negligible for the standard NCSX.





- 2T gives access to higher temperatures, lower collisionality
  $\nu_i^* \sim 0.1$
- 2T, 6MW and $n_e = 4\times10^{19}\ m^{-3}$ projects to $T_i\,(0) > 4$ keV
  $\nu_i^* \sim 0.04$

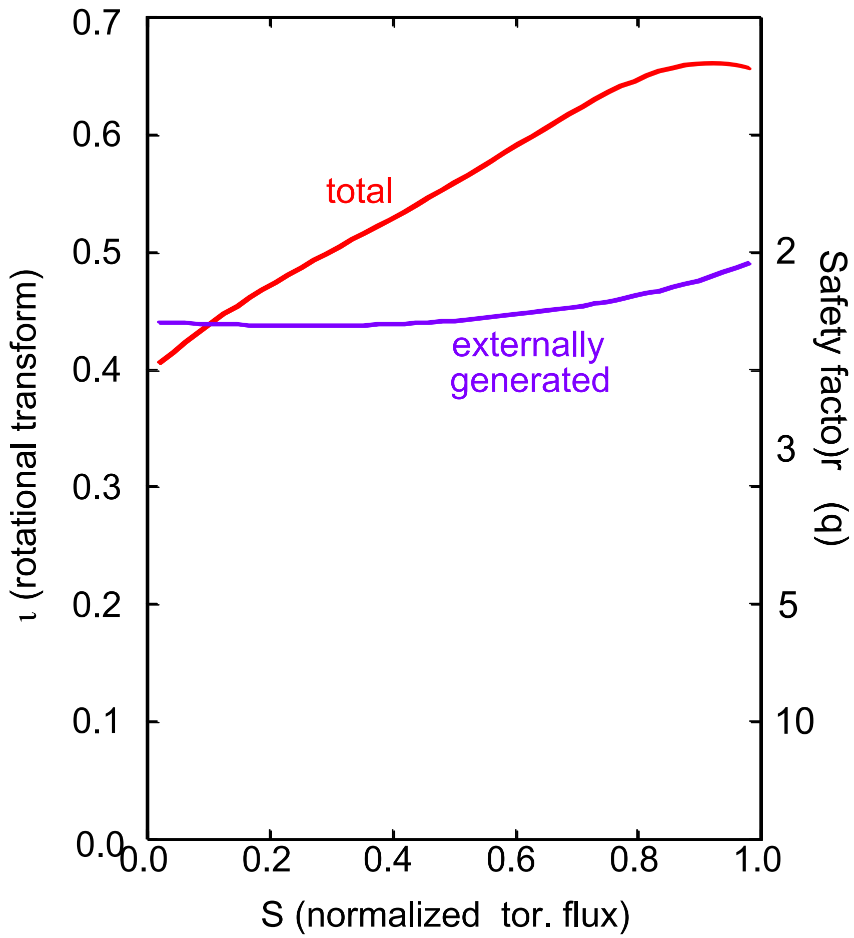


$n_e = 6\times10^{19}\ m^{-3}$

P_{inj} = 6MW

D. Mikkelsen

NCSX has 'Reversed Shear' Across Profile



Radial Coordinate^2

- QA $\Rightarrow$ tokamak like bootstrap current
- ~3/4 of transform (B-poloidal) from external coils $\Rightarrow$ externally controllable

$\Rightarrow$ much less non-linear than AT

- Rotational transform rising to edge key for stabilizing trapped particle and neoclassical tearing instabilities

Explored locally on tokamaks, but cannot be achieved across whole plasma using current.

- Zonal flow damping scales as q^2 (Rosenbluth & Hinton; Mynick)
$\Rightarrow$ lower in NCSX by factor of ~4 than toks.

Turbulence Growth Decreases for Higher ∇p Similar to Reversed Shear Tokamak

- Designed for 'reversed shear' to help stabilize turbulent transport

- Linear ion-scale turbulence growth rates calculated by FULL-code:
  - Electron-drive stabilized by reversed shear
  - Ion-drive strongly reduced with increasing β
  - Similar to reversed shear tokamak

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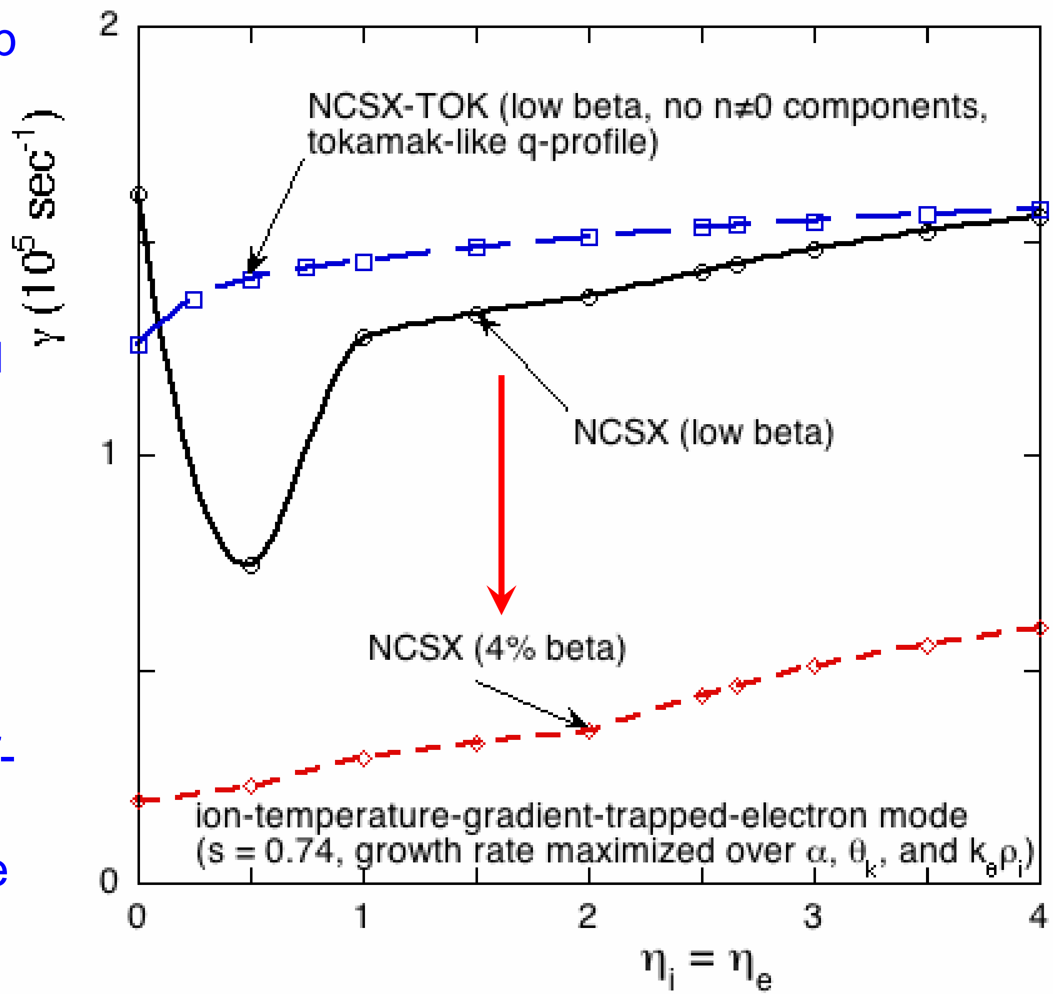
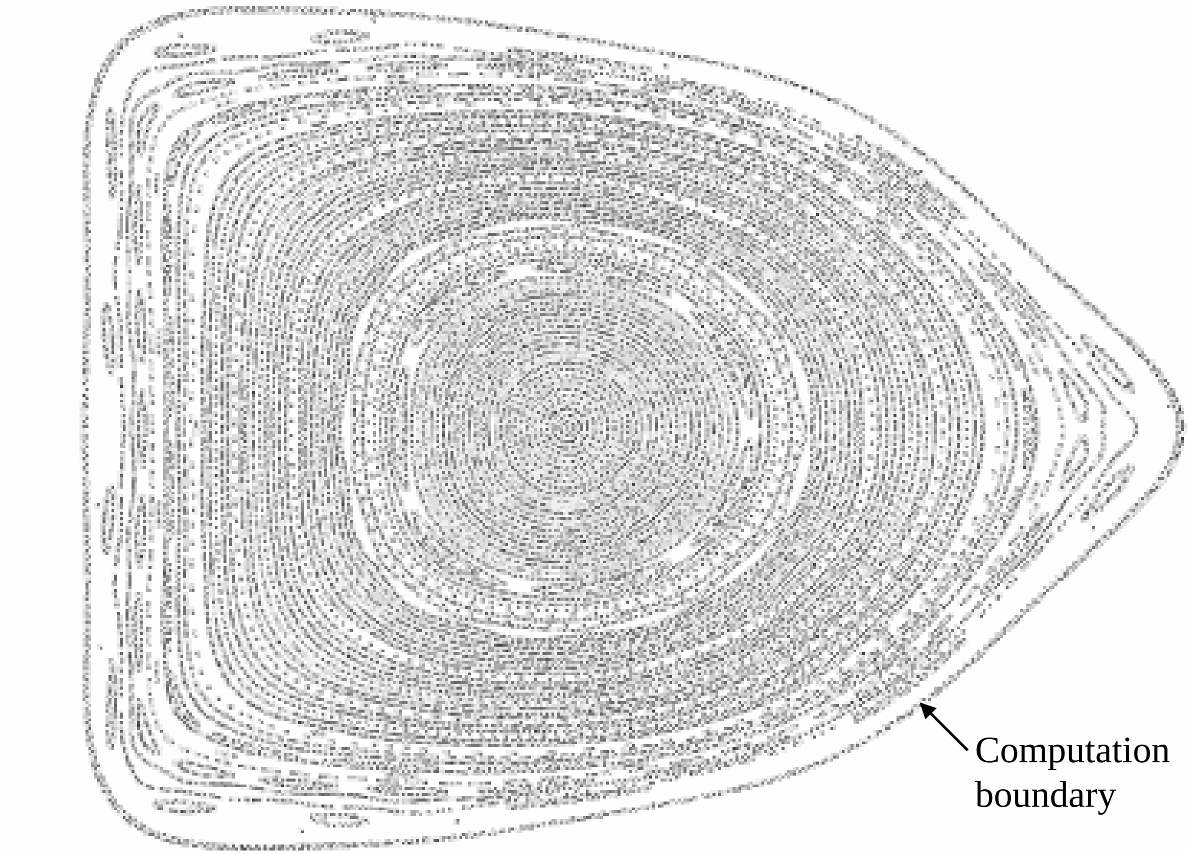
- In tokamaks with reversed shear, low growth rate at finite β allows self-stabilization driven flows. Produces neoclassical ion-thermal and particle transport!



- G. Rewoldt

NCSX Designed to Produce Good Flux Surfaces at High-β



Poincare: PIES, free boundary without pressure flattening

< 3% flux loss, including effects of reversed shear and || vs. ⊥ transport.

A. Reiman
S. Hudson
D. Monticello

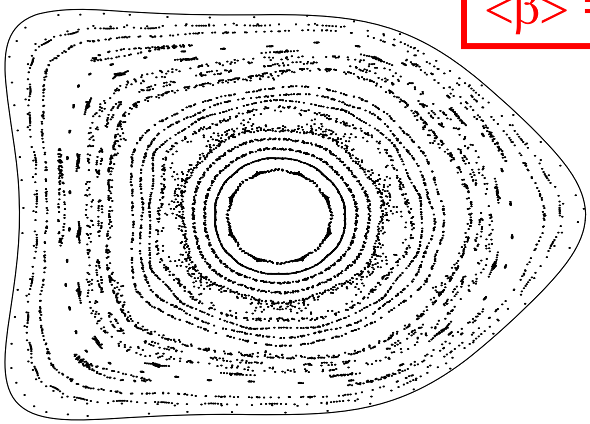
- Explicit numerical design to eliminate resonant field perturbations
- ‘Reversed shear’ configuration ⇒ theoretically, pressure-driven plasma currents heal equilibrium islands (not included in PIES calculation)

Initial Non-Linear Kinetic-MHD Calculations Indicate Possible Higher Pressure-Limit for NCSX

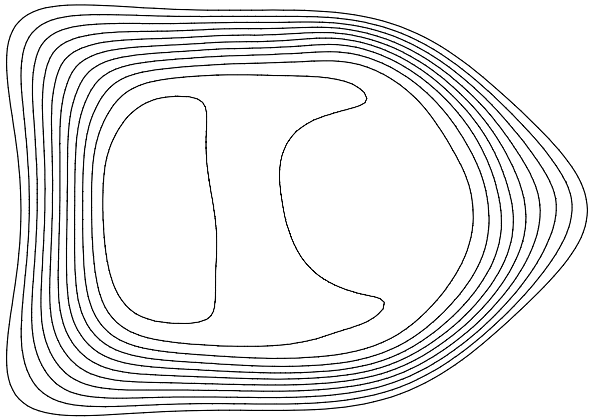
NCSX

$\langle\beta\rangle = 7\%$

Magnetic Flux Surfaces



ExB Flow Surfaces

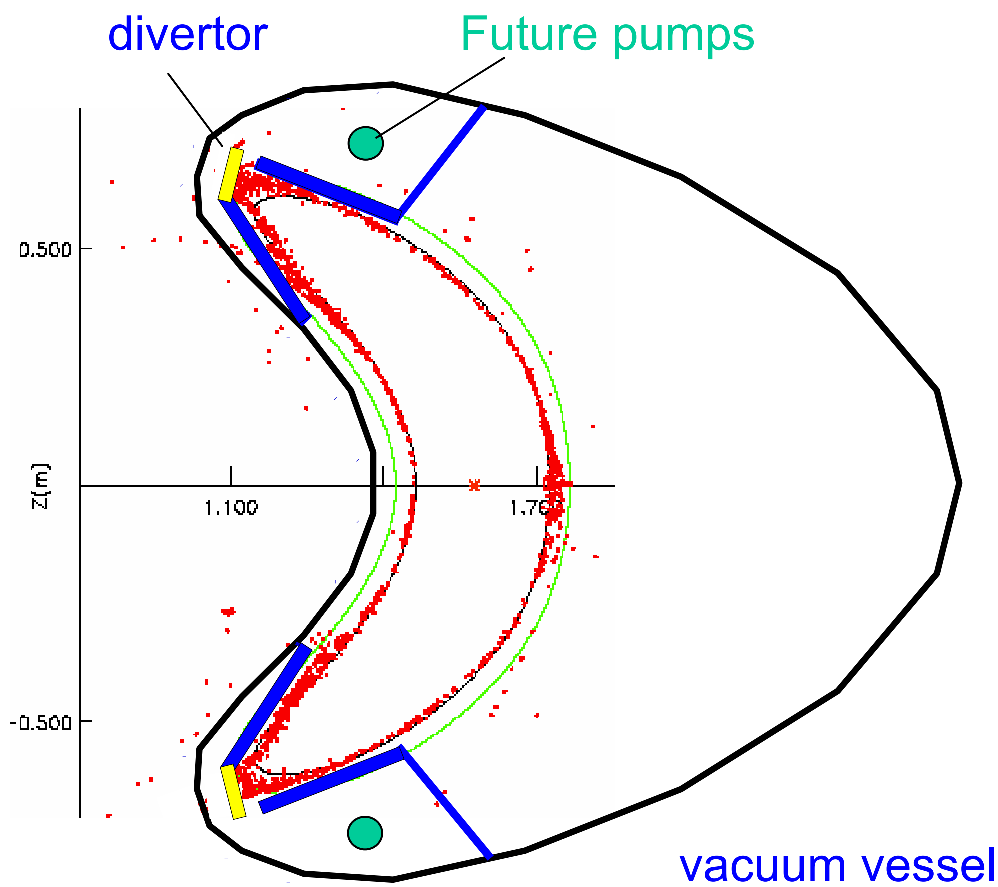


L. Sugiyama
H. Strauss

- Preliminary M3D calculations. Fixed boundary.
- Finite gyro-radius and ω_* stabilize equilibrium: good flux surfaces and no instabilities.
- Does not include neoclassical effects yet. Should further increase stabilization.
- What will the pressure limit be for NCSX?

Intrinsic Divertors in Bean-tips

- Divertors already operated successfully in LHD and W7-AS. Controlled exhaust and impurities.
- Strong flux-expansion always observed in NCSX bean-shaped cross-section. Allows isolation of PFC interaction.
- Similar to expanded boundary shaped-tokamak configurations
- SOL connection length can be ~100m. Long enough to ensure low temperature divertor plasma.
- EMC3 collaboration starting for divertor analysis/design



Inconel vacuum vessel
350 C bake

Edge Group:
R. Maingi
T. Kaiser

ARIES-CS: a Competitive, Attractive Reactor

Reference parameters
for baseline:

NCSX-like config.

$\langle R \rangle$ = 7.75 m

$\langle a \rangle$ = 1.72 m

$\langle n \rangle = 4.0 \times 10^{20}\ m^{-3}$

$\langle T \rangle$ = 6.6 keV T(0)=12 keV

$\langle B \rangle_{axis}$ = 5.7 T

$\langle \beta \rangle$ = 6.4%

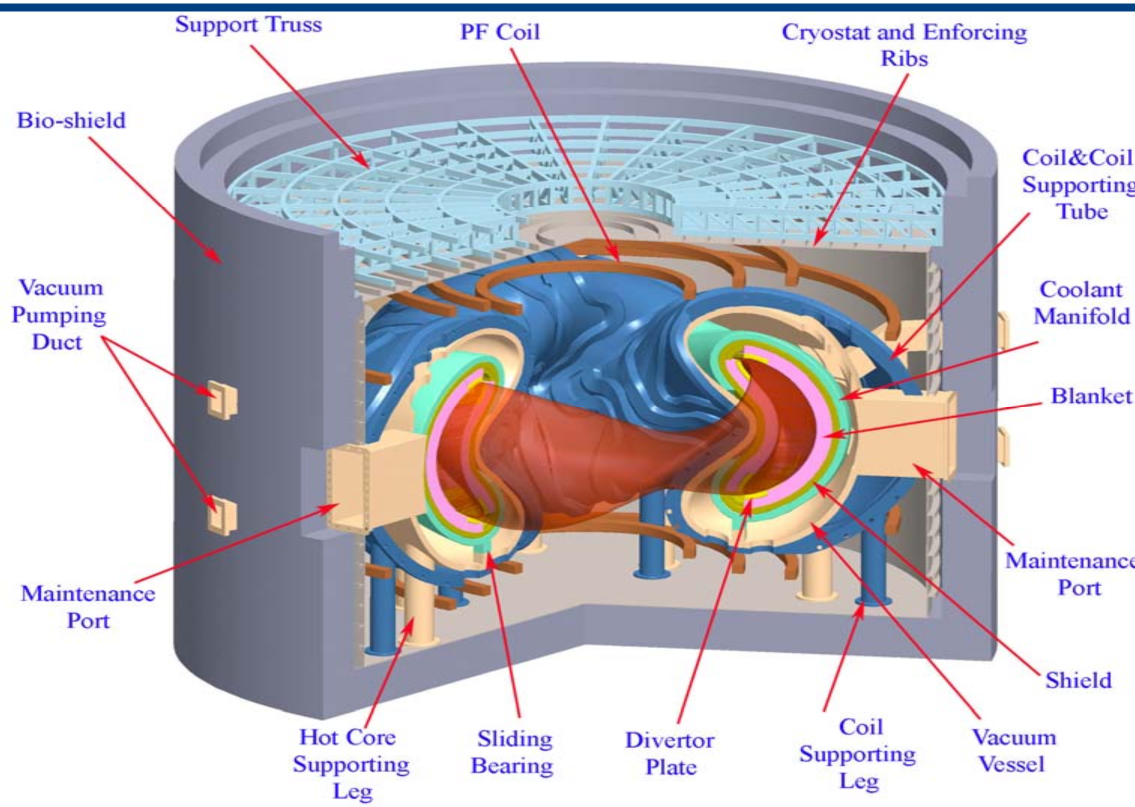
H(ISS95) = 2.0

H(ISS04) = 1.1

I_{plasma} = 3.5 MA
(bootstrap)

P(fusion) = 2.364 GW

P(electric) = 1 GW



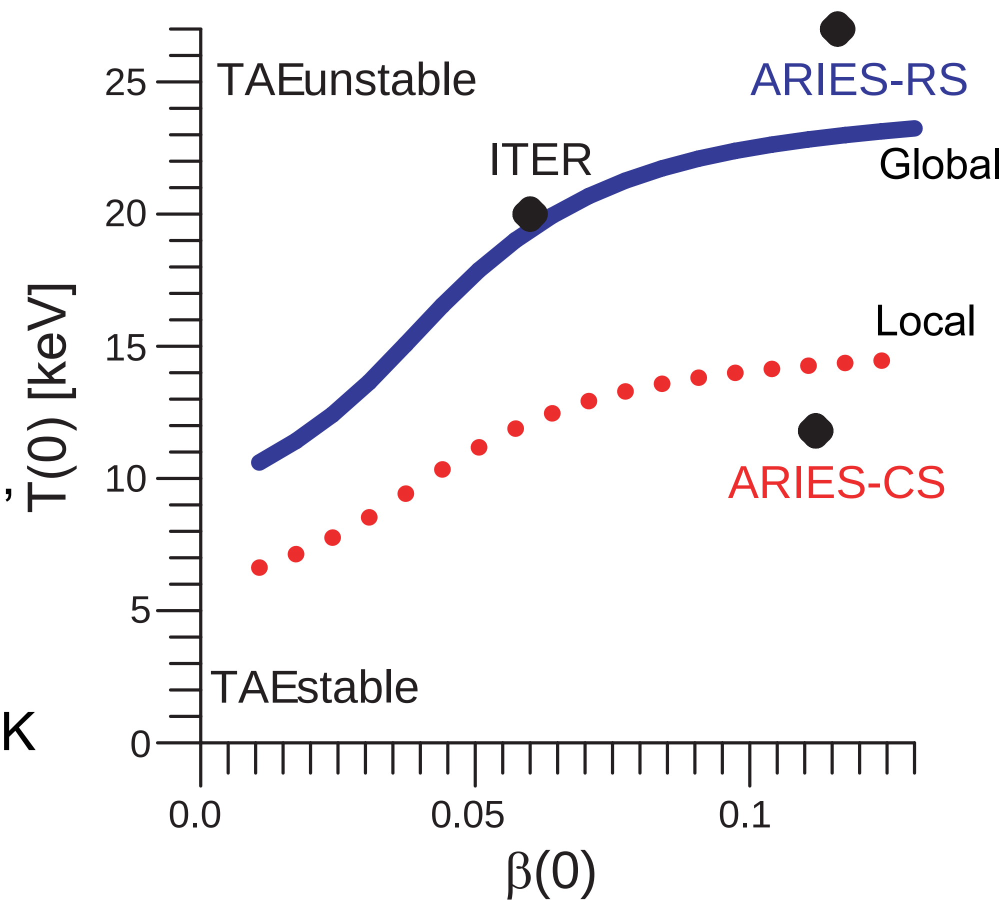
Aries-	-I	-RS	-CS	-AT	-CS
Blanket			LiPb/FS	LiPb/SiC	LiPb/SiC
COE(92)	99.7	75.8	61.3	47.5	48.

Based on NCSX design

Aries-CS Team

ARIES-CS Should be Stable to α-driven TAE

- Higher density in stellarators
  - Decreases T
    $\Rightarrow$ decreases β_α
  - Increases Landau damping on ions
  - Increases damping on trapped electrons



- Analytic local stability analysis axisymmetric. N. Gorelenkov et al, NF **43**, 594 (2003). Curves for 3 configurations ~ identical.

- Global curve normalized to Nova-K global analysis of ITER. Includes FLR and mode-structure effects.

- Aries-AT has T(0)~31 kev. More unstable.

N. Gorelenkov
G.-Y. Fu

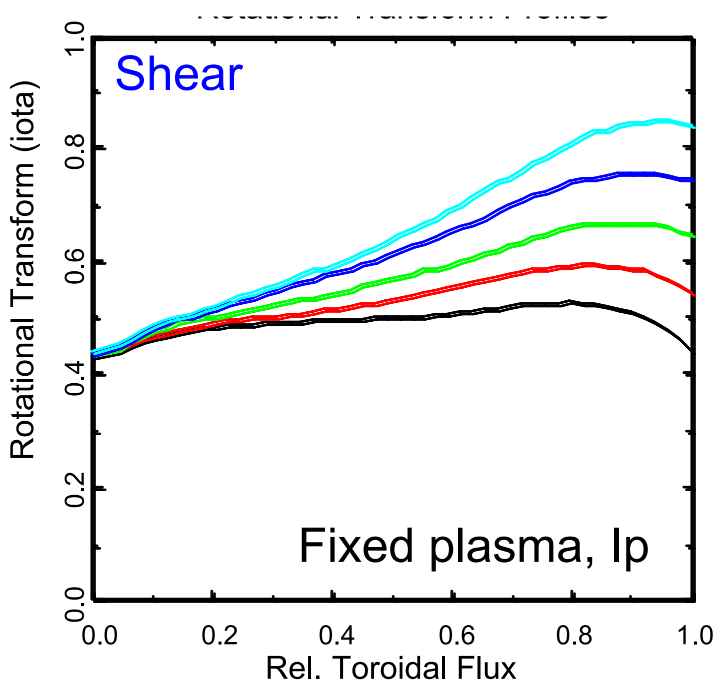
NCSX Coils Designed for Flexibility

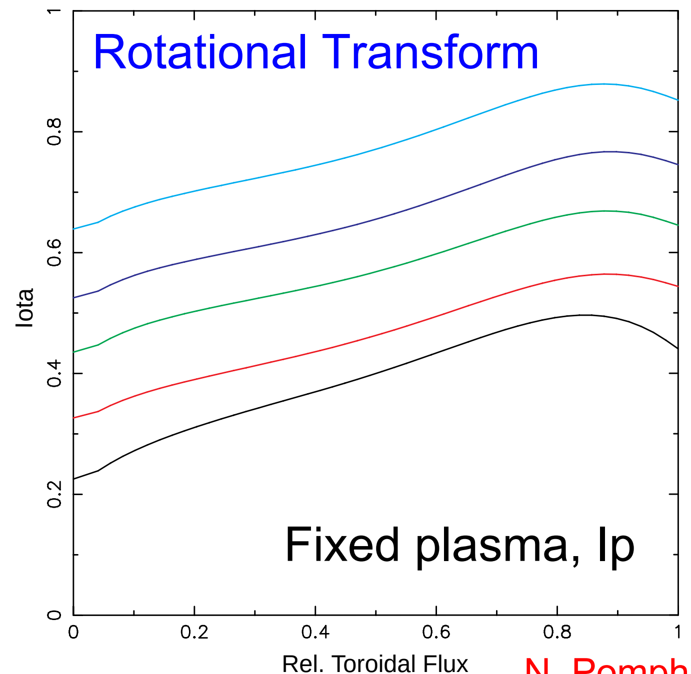
- Modular Coils + B_T Solenoid + Poloidal coils + Trim coil array.
  Coils for shaping control & flexibility

- E.G., can use coils to vary
  - effective ripple by factor ~ 30.
  - Avg. magnetic shear by factor > 5
  - Edge rotational transform by factor of ~4 (vac) or factor of ~2 (high bootstrap current)

- Reduce kink-instability threshold down to β=1% by modifying plasma shape
  - either at fixed shear or fixed edge-iota !

- These types of experiments will be key for developing and validating our understanding of the role of 3D QA shaping.





N. Pomphrey

NCSX : A Proof-of-Principal Facility

NCSX designed for 12MW of heating power

- NBI (0.5s), ECH (2s), and ICRF (3s), to heating ions and electrons
- NBI needed for high-beta access (historically) and gives torque for rotation studies
- ECH via collaboration with IPP/Greifswald and GA

Magnetic field strength up to 2T

- Enables experiments separating ρ^* scaling of transport from ν^* and β scalings.
- Gives overlap with tokamak database in dimensionless parameters

  Crucial for connecting to tokamak transport understanding,

  and forming joint understanding with ITER (particularly ρ^* scaling)

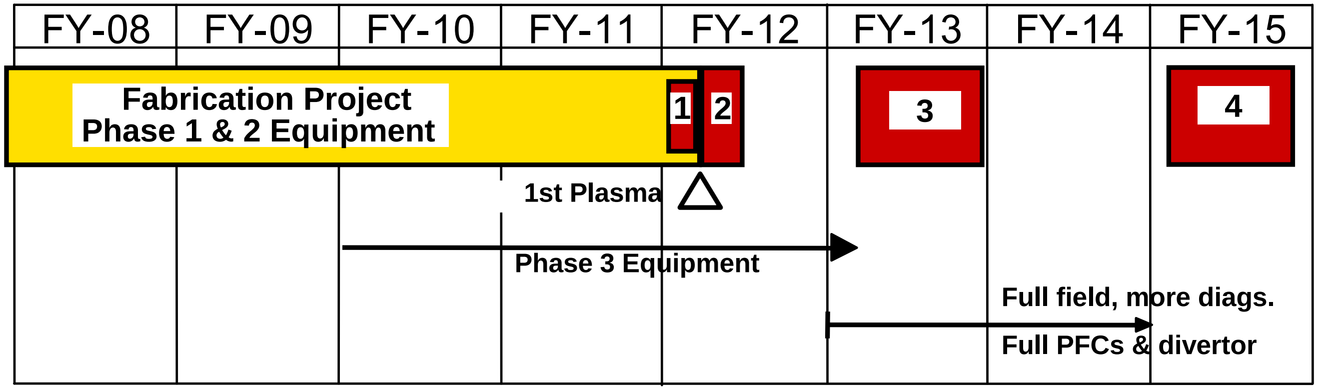
Comprehensive Diagnostics

- Time evolving profiles, fluctuations, 3D magnetics, fast ions, edge,…

Divertor

Provide integrated understanding of 3D QA physics at high β, low ν^*

NCSX Research Timeline



1. Stellarator Acceptance Testing & First Plasma (Fabrication Proj.)
2. Magnetic configuration studies
   – electron-beam mapping studies
3. Initial Heating Experiment
   – 3MW NBI. ECH?
   – B $\geq$ 1.2T
   – Partial PFC coverage
   – Initial diagnostics, magnetics, profiles (n_e, T_e, T_i, v_ϕ, P_{rad}) & SOL
4. High beta Experiments
   – 6MW heating, NBI & ECH
   – B = 2T; divertor & full PFC coverage
   – Improved diagnostics

Approximately 1/3 of research will be carried out by collaborators

FY13 and FY15 Campaigns will Investigate Critical Issues

FY13

- Effect of QA shaping and effective ripple on confinement and rotation damping
- Resilience to disruptions from MHD instabilities, density limits
- Initial comparisons between observed & calculated MHD stability thresholds

FY15

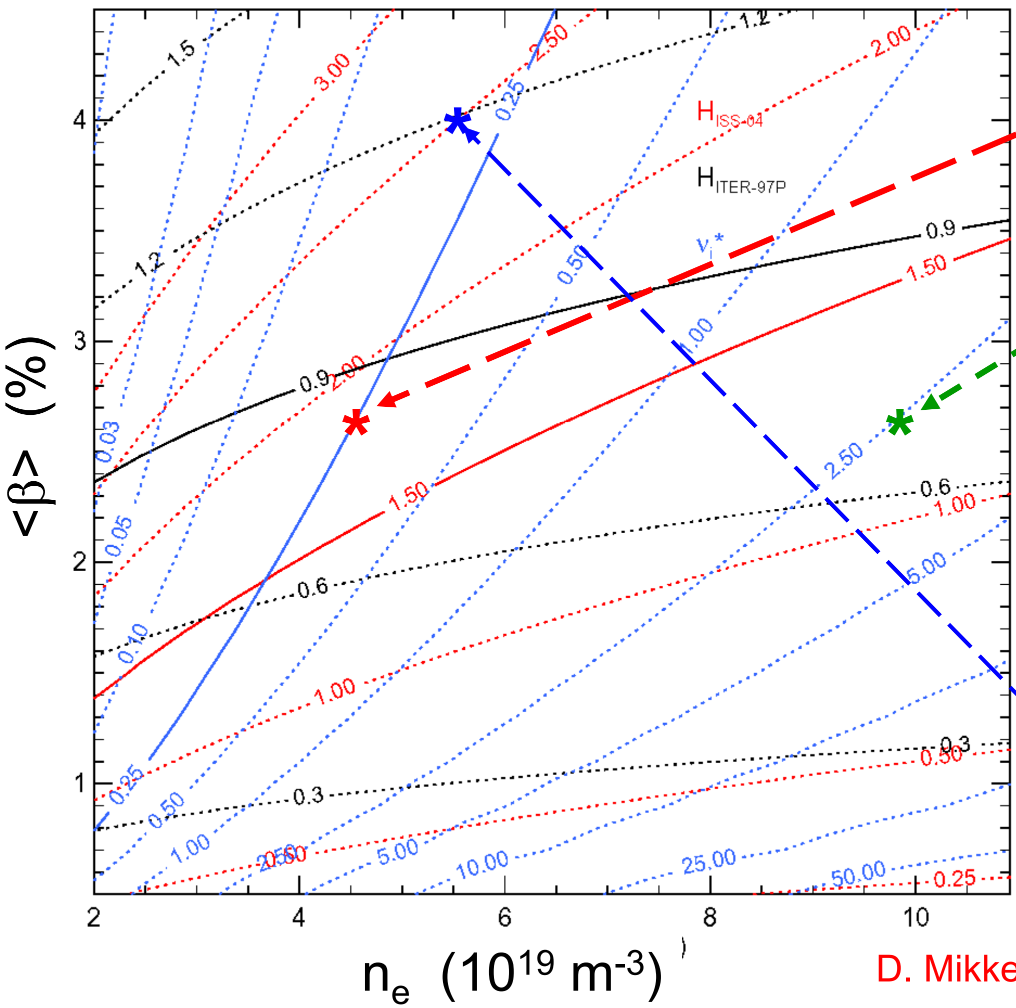
- β-limits and limiting mechanisms
- Safe operating area against disruptions
- Local transport properties; impurity transport
- Fast ion transport due to effective ripple. Alfvenic-mode stability.
- Initial divertor effectiveness; scrape-off layer characteristics.

Prioritized plans have been developed for each campaign.

Reviewed at PAC meetings, Research Forum.

Wide Range of β and ν^* Accessible in FY13 For initial studies

Contours of H_{ISS04}, $H_{ITER\text{-}97P}$, and min ν_{*i}



D. Mikkelsen

B = 1.2 T, 3MW NBI

β=2.7%, ν_{*I} =0.25 with

H_{ISS04}=1.8 $H_{ITER\text{-}97P}$=0.8

β=2.7%, ν_{*I} =2.5 with

H_{ISS04}=1.2 $H_{ITER\text{-}97P}$=0.7

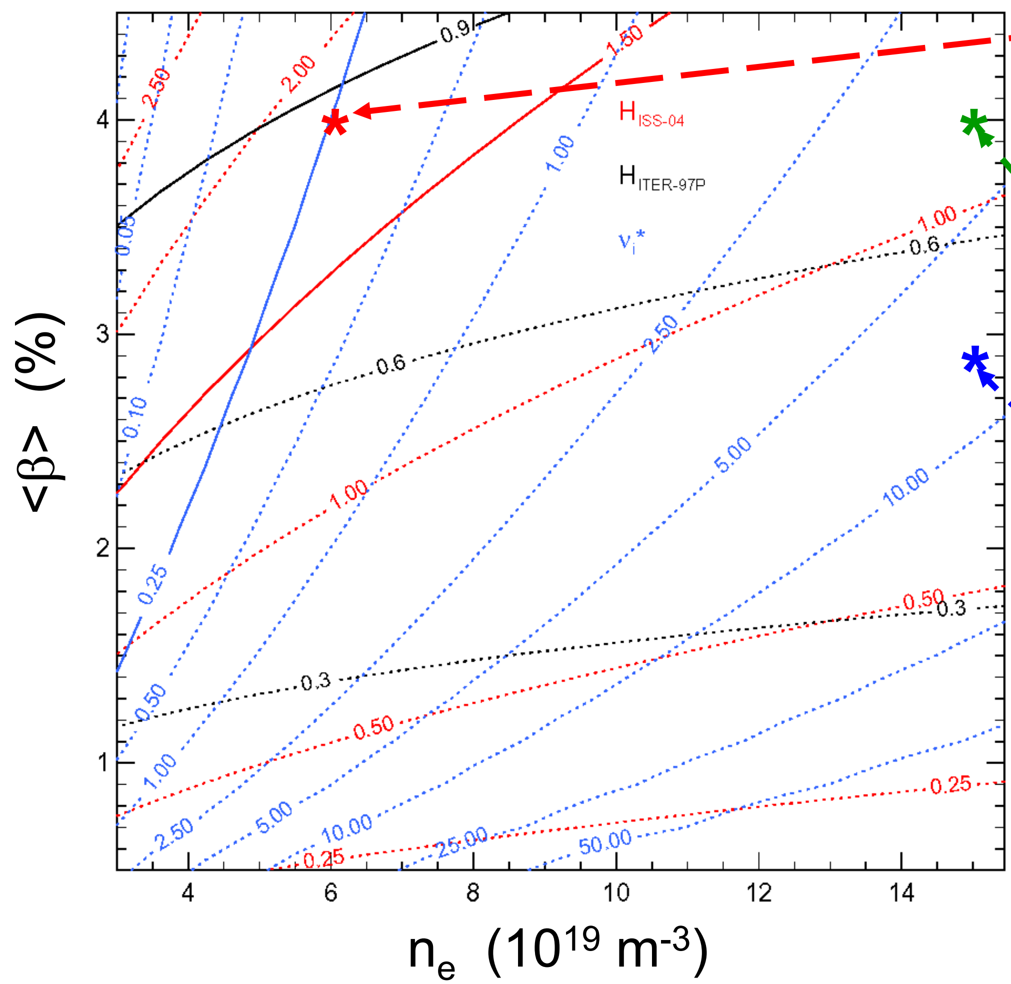
If get **$H_{ITER\text{-}97P}$**~1.2, can access **β=4%** with 3MW

(e.g. ITPA H-mode threshold is ~0.5 MW for NCSX 1.2T)

Stellarator and tokamak scaling laws differ for NCSX, due to strong shaping

High-β, low ν* Plasmas Accessible in FY15

Contours of H_{ISS04}, $H_{ITER-97P}$, and min ν_{*i}



B = 1.2 T, 6MW

* **β=4%,** ν_{*i} =0.25 requires
H_{ISS04}=1.8 $H_{ITER-97P}$=0.9

* **β=4%** at Sudo-density
H_{ISS04}=1.1, **$H_{ITER-97P}$=0.75**

* **H_{ISS04}=0.5** gives **β=2.9%**
at high collisionality

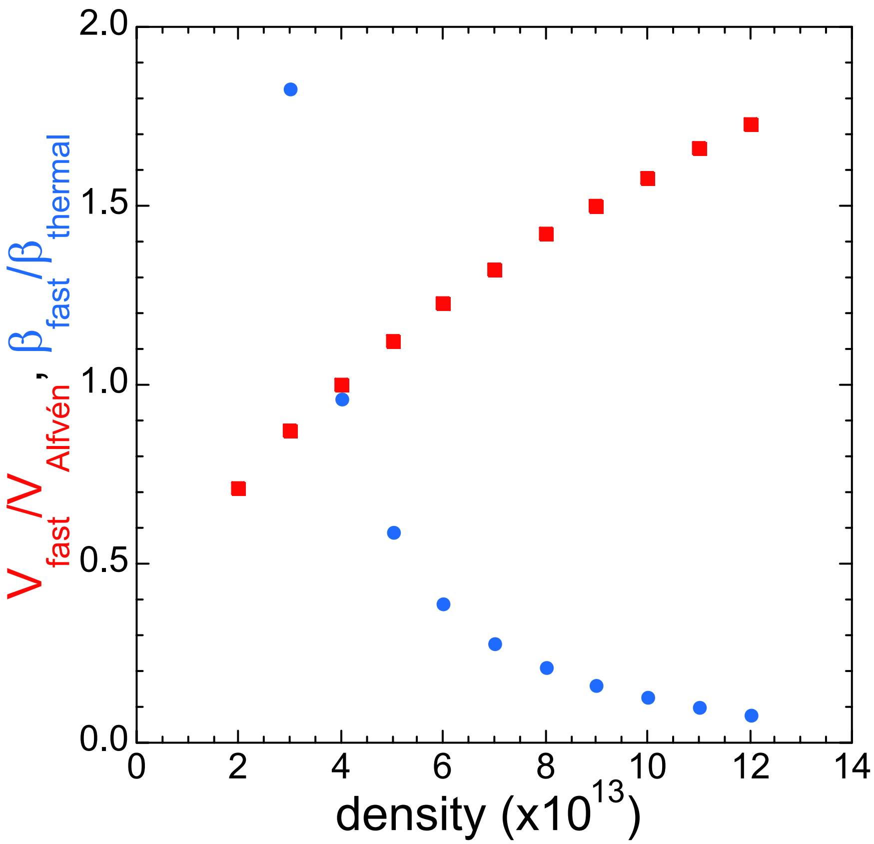
$H_{ITER-97P}$=1.2 gives access to β> 6% with 6MW

M

Fast-Ion Alfvenic Stability Experiments will Start in FY13 with 3MW

- Density scan controls β_{fast} and $V_{Alfvén}$.
- Window around 4–6 x $10^{13}/cm^3$ for Alfvén mode studies:
  - beam ions should be super-Alfvénic.
  - $\beta_{fast}/\beta_{thermal}$ > 30% .
- E.g. early TFTR beam driven TAE studies were at ≈ 3 x $10^{13}/cm^3$, 110 keV deuterium beams, $B_{tor} \approx$ 1T $V_{fast}/V_A \approx \beta_{fast}/\beta_{thermal} \approx 1$
- Will measure ripple transport of fast ion, starting in FY15.

1.2 T, H_{ISS04}=1
3 MW of 45 kV hydrogen beams



E. Fredrickson

Broad Diagnostic Set Available in FY13

Initial diagnostic upgrades

- In-vessel + ex-vessel magnetic diagnostics
- Thomson scattering (n_e, T_e profiles)
- Imaging x-ray crystal spectrometer (v_ϕ, T_i profiles)
- UV spectrometer
- PFC-mounted probes
- Filtered 1D and 2D cameras. Filterscopes.
- IR cameras
- SXR camera
- Interferometer
- Bolometer array

Black: shared w/ NSTX

Collaborations on diagnostics are expected and being discussed.
Diagnostics upgrading will continue throughout the Research Program.

Candidate diagnostic upgrades for FY15 run

- MSE
- Confined and lost fast ion diagnostics
- Heavy ion beam probe (via NIFS collaboration?)
- Soft x-ray tomography
- Reciprocating Langmuir probe
- CHERS
- Additional Thomson Scattering spatial channels
- Reflectometer or other fluctuation diagnostic

NCSX will have clear conclusions by 2015

<u>After the FY15 run</u>

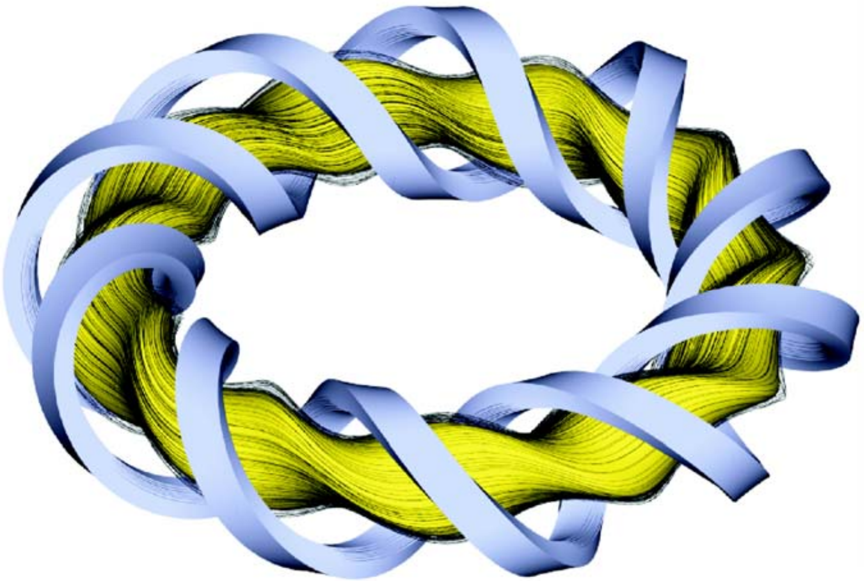
Some definitive conclusions on attractiveness of 3D QA shaping

- Do disruptions occur? ELMs?
- Is the β-limit determined by MHD instabilities?
- Confinement: tokamak-like or stellarator-like? Effect of ε_{eff}
- Is energetic ion confinement as expected?
- Can we spread the divertor footprint?

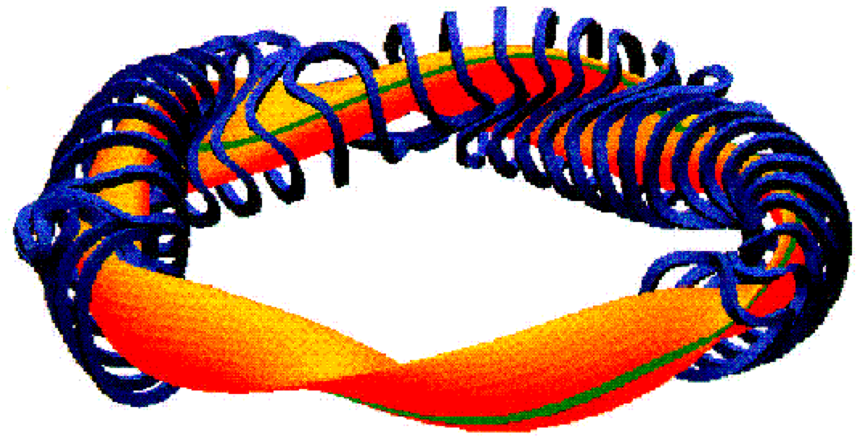
<u>In the 2015 ~ 2023 time-range</u>

- Detailed physics studies
- Refine and complete scientific understanding of 3D QA shaping
- Document basis for optimization of follow-on devices

NCSX Complements the Large International Superconducting Stellarators



Large Helical Device (Japan)
Non-symmetric
A = 6-7, R=3.9 m,   B=3T



Wendelstein 7-X (Germany)
QP optimized design
A = 11, R=5.4 m,   B=3T

- Focused on steady state, including power handling. LHD has achieved 54-minute pulses.
- Optimized for other properties than quasi-symmetry $\Rightarrow$ flows strongly damped
- Not compact. Extrapolate to larger fusion systems than favored in U.S.
- Neither can directly build on or inform tokamak understanding.

Implications of the Loss of NCSX

<u>If NCSX is not completed</u>

- There will be no experiment in the world studying 3D QA shaping
- The option to use 3D QA shaping in combination with ITER results to solve the DEMO issues will be lost
  - $\Rightarrow$ *Significantly increased risk for 1st DEMO success.*
- There will be no PoP scale experiment studying quasi-symmetric or compact stellarators, in the US or the world.
- Likely, the 1st DEMO would not consider 3D shaping to eliminate disruptions, or produce steady-state high-Q high-power plasmas.

Summary: NCSX offers unique contributions

2(a): What critical, unique contributions does NCSX potentially offer for addressing the issues?

- Only experiment studying 3D Quasi-Axisymmetric shaping, to build upon ITER and tokamak understanding
- Uniquely, will study 3D QA plasma shaping in an integrated context: at high-β, low collisionality, and over a range of ρ^* (gyro-radius), including a divertor edge
  - Ion, electron, and impurity transport and confinement
  - Operational limits and mechanisms: β and density
  - Divertor effectiveness and control of impurity influx
  - Energetic ion confinement and stability
  - Effects of error fields
- NCSX offers unique contributions to solve the plasma physics challenges for DEMO, lowering the risk of DEMO, and build on ITER and tokamak results
  - No disruptions. Non-disruptive limits.
  - Higher pressure limits without feedback stabilization.
  - Sustainment without current drive.
- NCSX has flexibility to understand the degree of 3D shaping needed in the future.

NCSX will provide timely answers

2(b): Given the proposed plans for operation, what would be the timetable for resolving relevant issues identified in (1)?

- NCSX will resolve critical issues in FY13 and FY15:
  - Do disruptions occur? Can ELMs be eliminated?
  - Is the β-limit determined by MHD instabilities?
  - Is confinement in a strongly shaped QA-torus tokamak-like or stellarator-like? What is the effect of very low ε_{eff} ?
  - Is energetic ion confinement as expected?
  - Can we spread the divertor footprint?

- _In the 2015 ~ 2023 time frame, NCSX will provide
  - Detailed experimental studies of all the physics issues
  - Refine and complete scientific understanding of 3D QA shaping
  - Provide the basis for optimization of follow-on devices

NCSX fills a critical void

2(c): What are the differences between NCSX and other stellarators? What is the significance of these differences? Does NCSX fill a critical void in the development of the stellarator concept?

- NCSX is the only experiment studying 3D Quasi-Axisymmetric shaping, to build upon the tokamak and ITER confinement understanding
- International stellarator experiments are studying other 3D optimization strategies, which do not allow low flow-damping and building on tokamak confinement understanding.
- NCSX is the only PoP scale quasi-symmetric or compact stellarator experiment, studying high-β plasmas with low collisionality, over a range of ρ^* (gyro-radius), including a divertor edge.
- NCSX fills a critical void in the development of fusion energy: studying 3D QA shaping to solve DEMO's plasma physics challenges, in a way that builds on ITER and tokamaks
  - No disruptions. No external current drive. Higher pressure and density.
- NCSX will reduce the risks of DEMO.