Issues of MHD Control for Stellarator Burning Plasmas

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Workshop on Active Control of MHD Stability
Princeton, NJ
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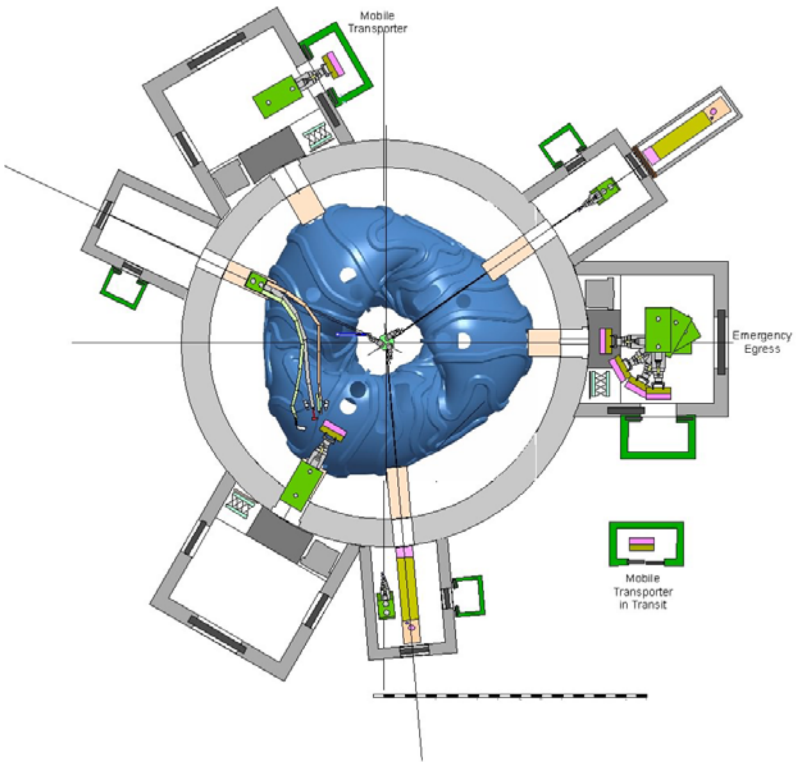
Outline

- Motivation and Context
- ARIES-CS
- Feedback needs



Motivation: Key advantages

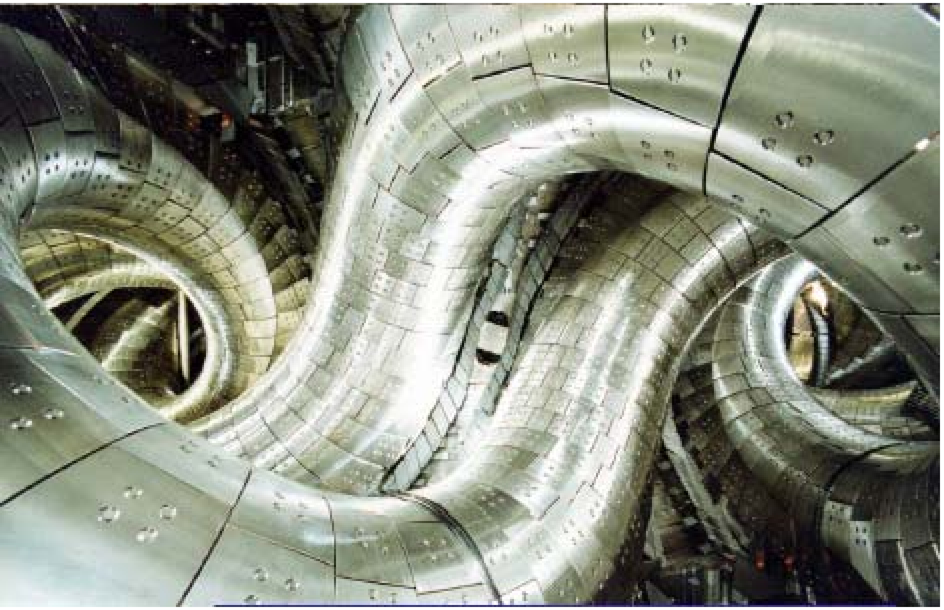
- Stellarators: Toroidal magnetic configurations, fully 3D shape

- Most of the rotational transform (iota = 1/q) due to 3D shape, not plasma current
  Can control rotational transform & shear from external coils
  No need for current drive to sustain configuration. Naturally compatible with steady state.

- Stellarators are typically disruption free
  Equilibrium is not lost due to changes in pressure or current.

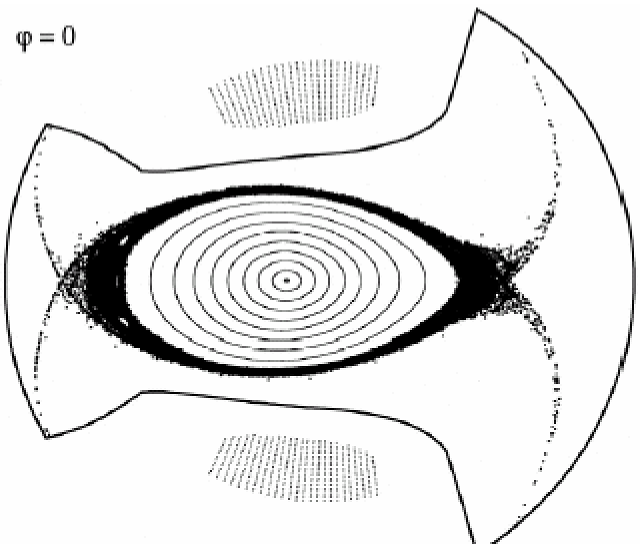
- Can use 3D plasma shaping to control physics properties (~ 40 shape parameters instead of ~4 for axisymmetric)
  More flexibility in configuration design

LHD: largest stellarator, record parameters



- R = 3.6 – 3.9 m
- minor radius <a> = 0.6m
- B   3T
- 12 MW NBI, 3MW ICH, 2MW ECH

- T_e, T_i up to 10 keV
- < > up to 4.5%
- pulse lengths up to 3268 s
- ${}_E$ up to 0.36 s



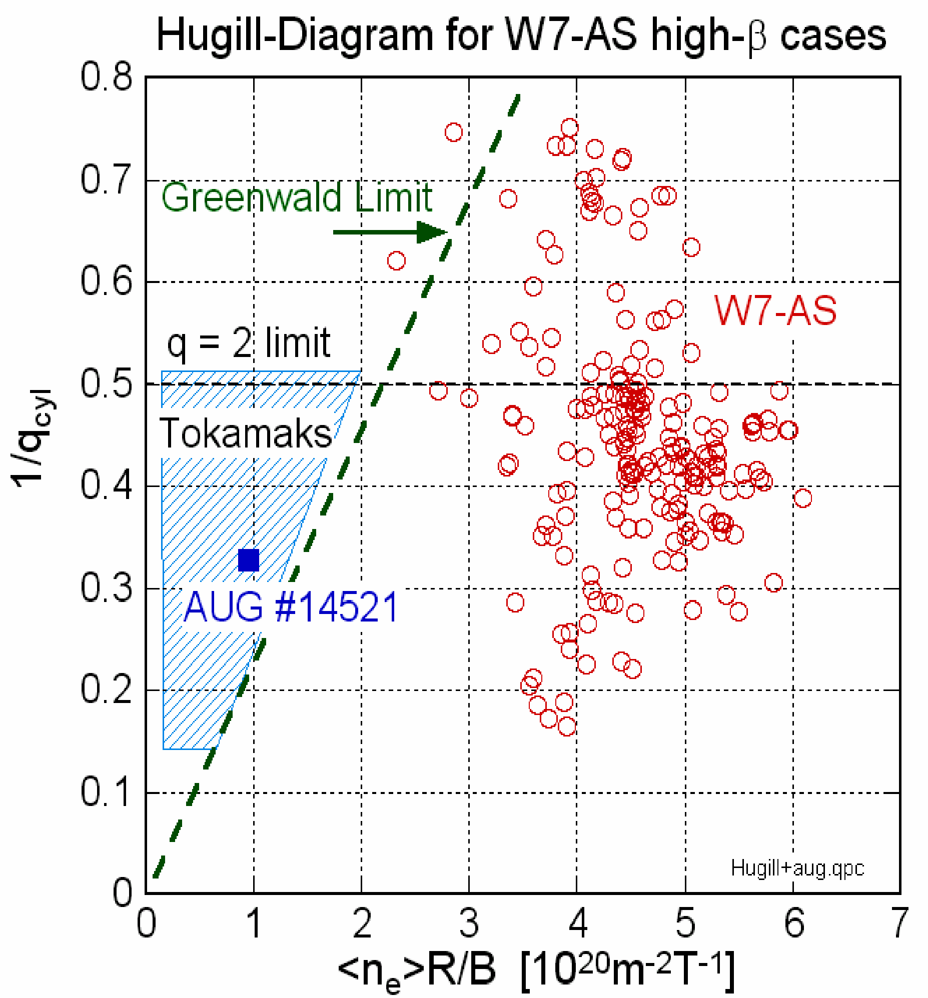
Worlds largest superconducting coil system

- 1 GJ of magnetic energy
- 850 ton cold mass at 4K

Stellarator Operating Limits
Very Different than Tokamaks

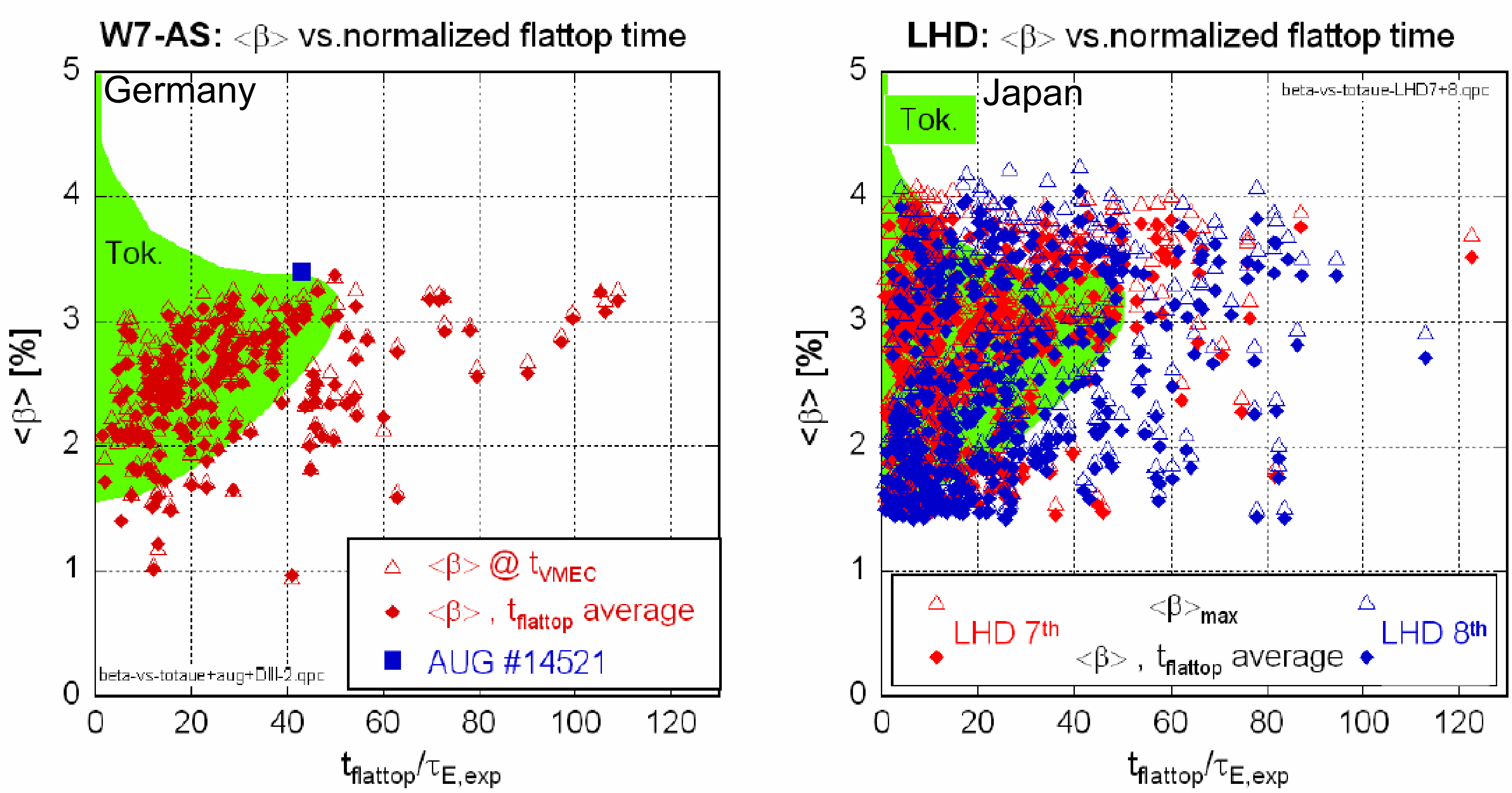
- Stellarators operate at much higher density than tokamaks
- Limit not due to MHD instabilities. Density limited by radiative recombination
- High- is reached with high density (favorable density scaling in W7-AS)

- High density favorable for burning plasma/power plant:
  - ✓ Reduces edge temperature, eases divertor solution
  - ✓ Reduces pressure and reduces -particle instability drive



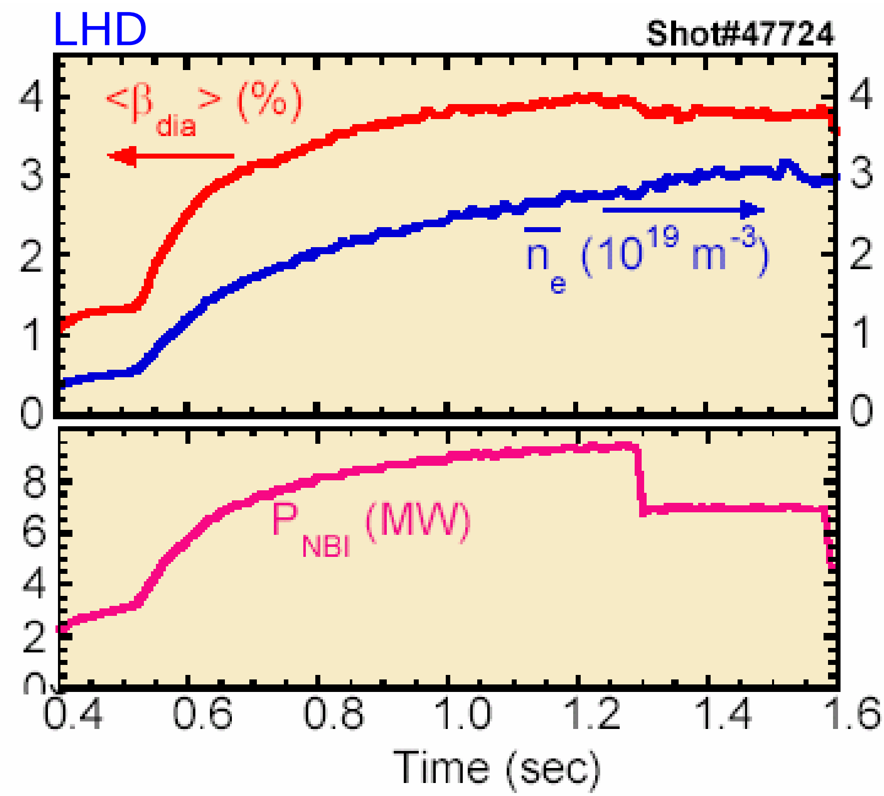
Greenwald density evaluated using equivalent toroidal current that produces experimental edge iota

Recent W7AS and LHD Experiments: Steady High- , Above Linear Limit



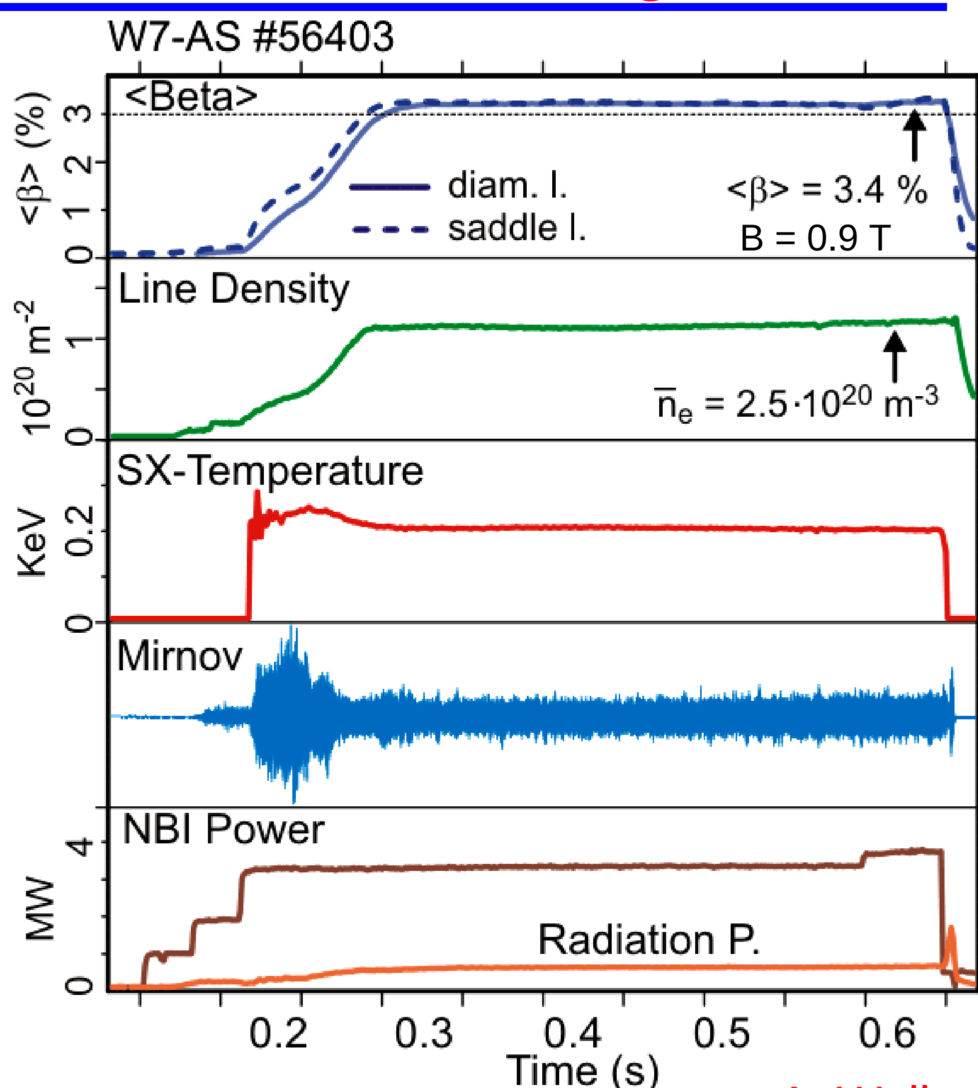
- In both cases, well above ideal MHD instability threshold < 2%
- Not limited by MHD activity. No disruptions observed.
-   -limit does not change with pulse lengths, unlike tokamaks

LHD & Wendelstein 7-AS: Quiescent high-



B = 0.45 T, R=3.6 m

- K. Watanabe



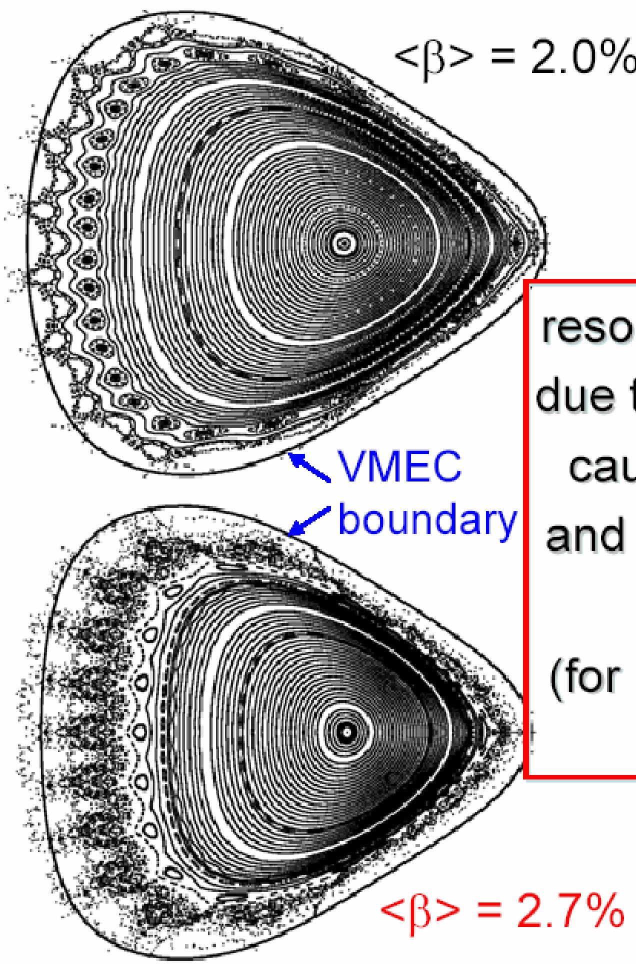
- A. Weller

- No disruptions
- In both experiments,   is not limited by observed MHD instabilities

β – Limit caused by Destruction of Outer Surfaces ?

- advanced 3D equilibrium codes show penetration of stochastic region -

PIES-Analysis for W7-AS



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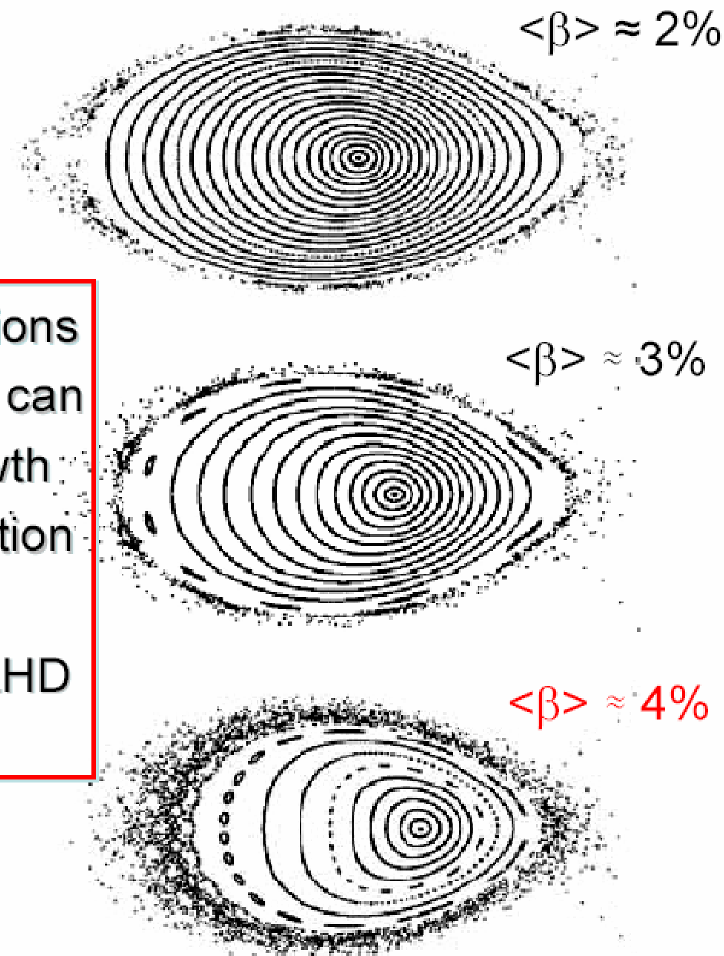
resonant perturbations due to PS-currents can cause island growth and field ergodization

(for <β> > 2 % in LHD and W7-AS)

HINT-Analysis for LHD

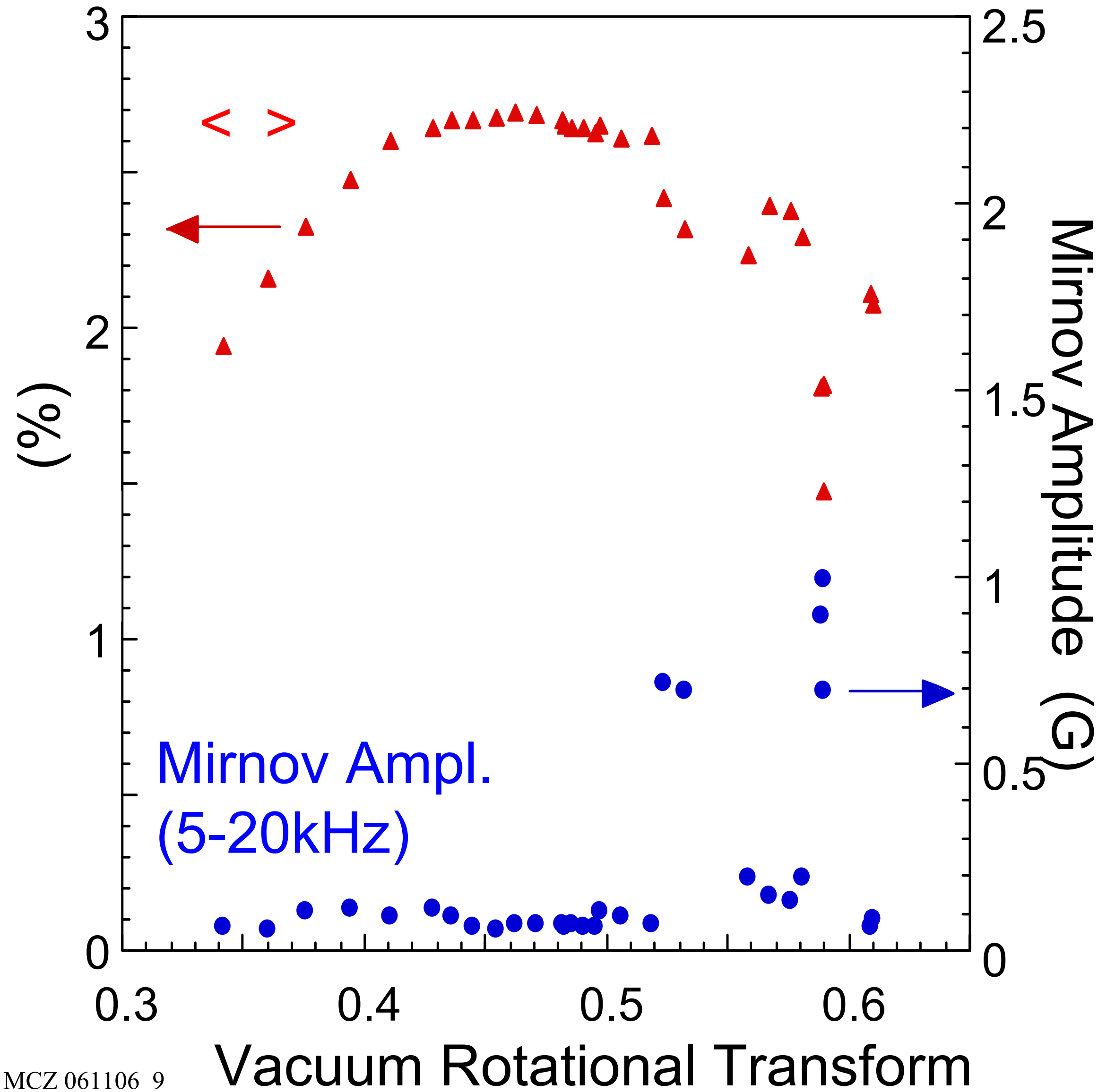


S. Sakaibara, Y. Suzuki

NCSX is designed to keep good flux surfaces at high

Can also be controlled by external trim coils

W7AS: MHD in Narrow Iota ranges



- Controlled iota scan, varying I_{TF} / I_M, fixed B, P_{NB}, flattop phase

- Strong MHD activity only in narrow ranges
  - –when total edge iota 0.5 or 0.6 (m/n=2/1 or 5/3)

MHD control: avoid low-order edge resonances
- by design
- by control of coil currents

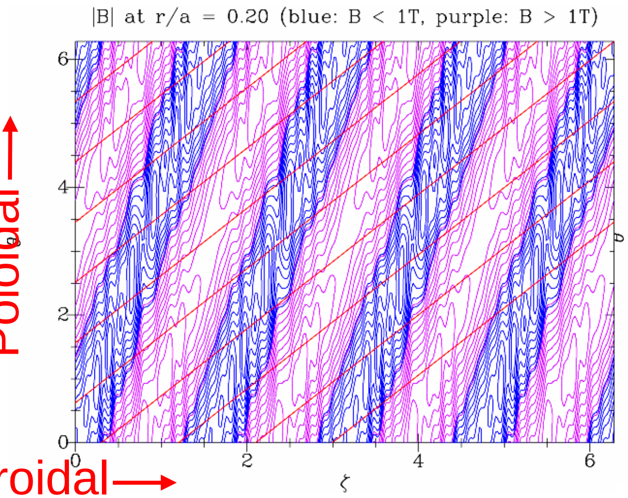
Stellarator Use Little Feedback Control

Typically, stellarators use simple control systems

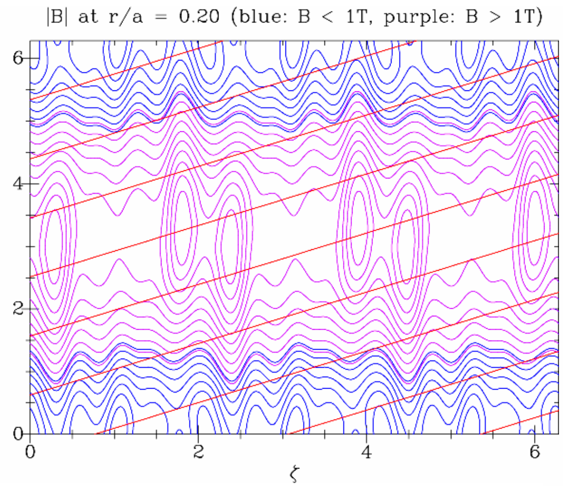
- Density, in some machines I_P
- Pre-programmed helical coil currents dominantly determine equilibrium
- More complicated feedback not crucial, since <u>no disruptions</u> and plasma does not show hysteresis (for low Ip)
- No analytic way to determine <R>, <a>, or boundary shape parameters
  No diagnostic to unambiguously measure flux surface locations
- If low m/n MHD control is needed, can be done via control of iota

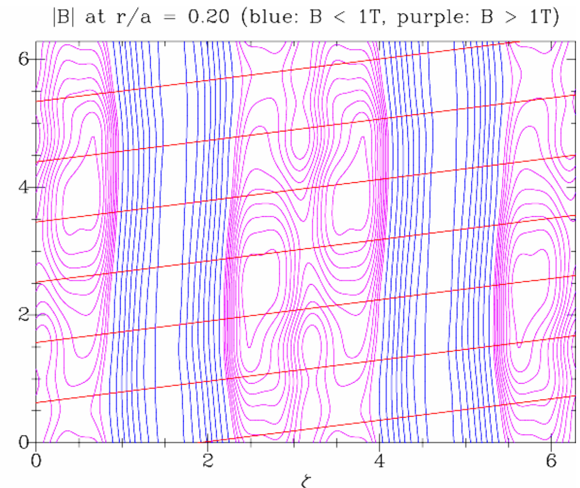
Quasi-symmetry to Optimize Orbit Confinement

Poloidal

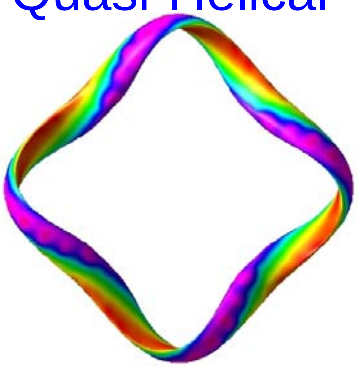


Toroidal



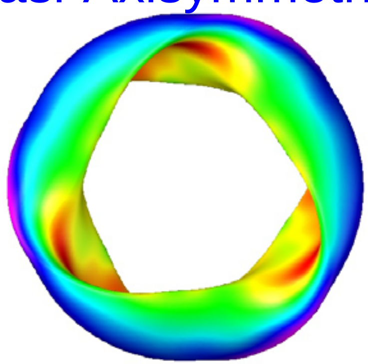


Quasi-Helical



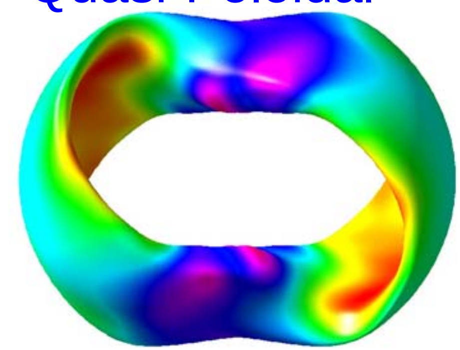
HSX, operating.

Quasi-Axisymmetric



NCSX, construction

Quasi-Poloidal



W7-X, construction
QPS, Proposed

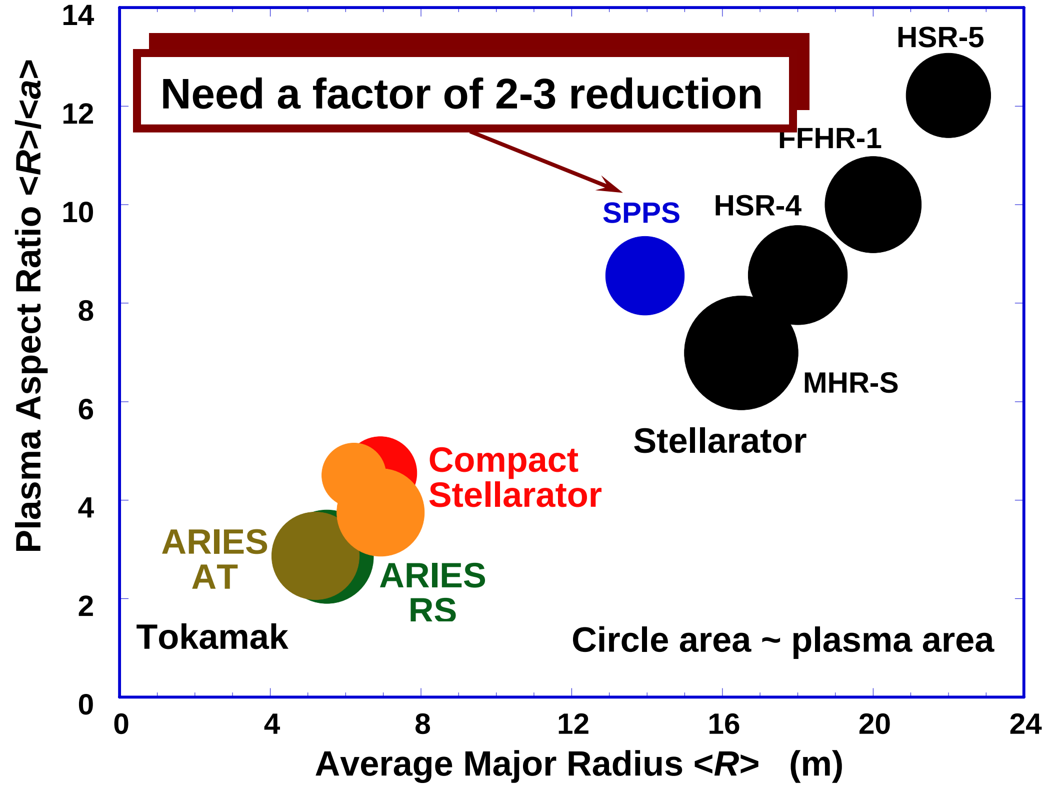
- Reduces ripple-driven cross field transport
- Compact designs found, with R/<a> < 4.5
- Can confine -particles in burning plasma

ARIES-CS Reactor Study Completing

- The ARIES Group has studied different reactors: RFP (TITAN), tokamaks (ARIES-I, -II, -IV, -AT and -RS), spherical torii, and stellarator (SPPS in the mid-1990's)
- 3-year study of a compact stellarator as a reactor (ARIES-CS) will complete at end of 2007.
- Tenth-of-a-kind power plants with aggressive physics and engineering assumptions

Goal: Stellarator Power Plants Similar in Size to Tokamak Power Plants

- Multipolar external field -> coils close to the plasma
- First wall/blanket/shield set a minimum plasma/coil distance (~2m)
- A minimum minor radius
- Large aspect ratio leads to large size.



- Approach:
  - ✓ **Physics:** Reduce aspect ratio while maintaining “good” stellarator properties.
  - ✓ **Engineering:** Reduce the required minimum coil-plasma distance.

Typical Plasma Configuration Optimization Criteria

Maximum residues of non quasi-axisymmetry in magnetic spectrum.

- ✓ neo-classical transport   anomalous transport:
  - overall allowable “noise” content < ~2%.
  - effective ripple in 1/   transport, $_{\text{eff}}$ < ~1%
- ✓ ripple transport and energetic particle loss
  - energy loss < ~5%

Equilibrium and equilibrium β limits

- ✓ Shafranov shift $\frac{\Delta}{<a>} \approx \frac{<\beta>\cdot A}{2\kappa\iota^{2}} < 1/2$
- ✓ large islands associated with low order rational surfaces
  - flux loss due to all isolated islands < 5%
- ✓ overlapping of islands due to high shears associated with the bootstrap current
- ✓ limit d /dr

Stability limits (linear, ideal MHD)

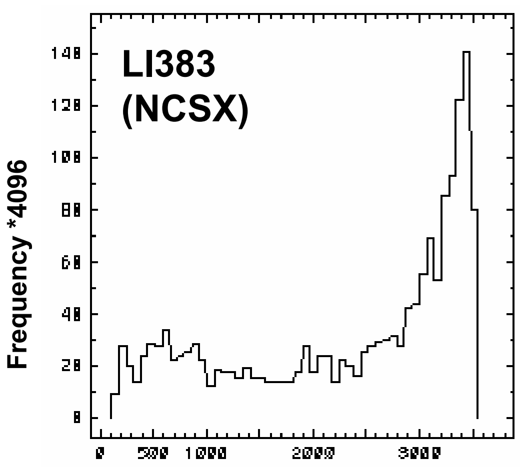
- ✓ vertical modes $\iota_{\text{ext}}/\iota \geq \frac{\kappa^{2}-\kappa}{\kappa^{2}+1}$
- ✓ interchange stability: V″~2-4%.
  - LHD, CHS stable while having a hill.
- ✓ ballooning modes: stable to infinite-n modes
  - LHD exceeds infinite-n results. High-n calculation typically gives higher   limits.
- ✓ kink modes: stable to n=1 and 2 modes without a conducting wall
  - W7AS results showed mode (2,1) saturation and plasma remained quiescent.
- ✓ tearing modes: d /dr > 0 for neoclassical stabilization

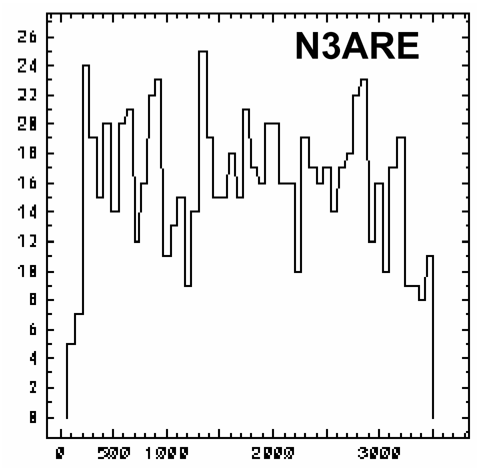
➢ Each criteria is assigned a <u>threshold</u> and a <u>weight</u> in the optimization process.

Key Development: Optimzation of NCSX-like Configuration for α-Confinement

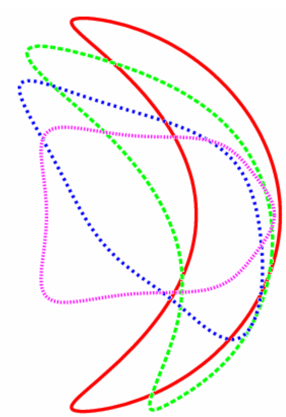
A bias is introduced in the magnetic spectrum in favor of B(0,1) and B(1,1)

✓ A substantial reduction in loss (to ~ 3.4%) is achieved.





N3ARE



Baseline Configuration

✓ The external kinks and infinite-n ballooning modes are marginally stable at 4% with no nearby conducting wall.

✓ Rotational transform and shape is similar to NCSX, so the same quality of equilibrium flux surface is expected.

✓ Can test physics in NCSX using shaping flexibility.

ARIES-CS Reactor Core

Reference parameters for baseline:

R = 7.75 m

a = 1.72 m

n = 3.6 x 10^{20} m^{-3}

T = 5.73 keV

B_{axis} = 5.7 T

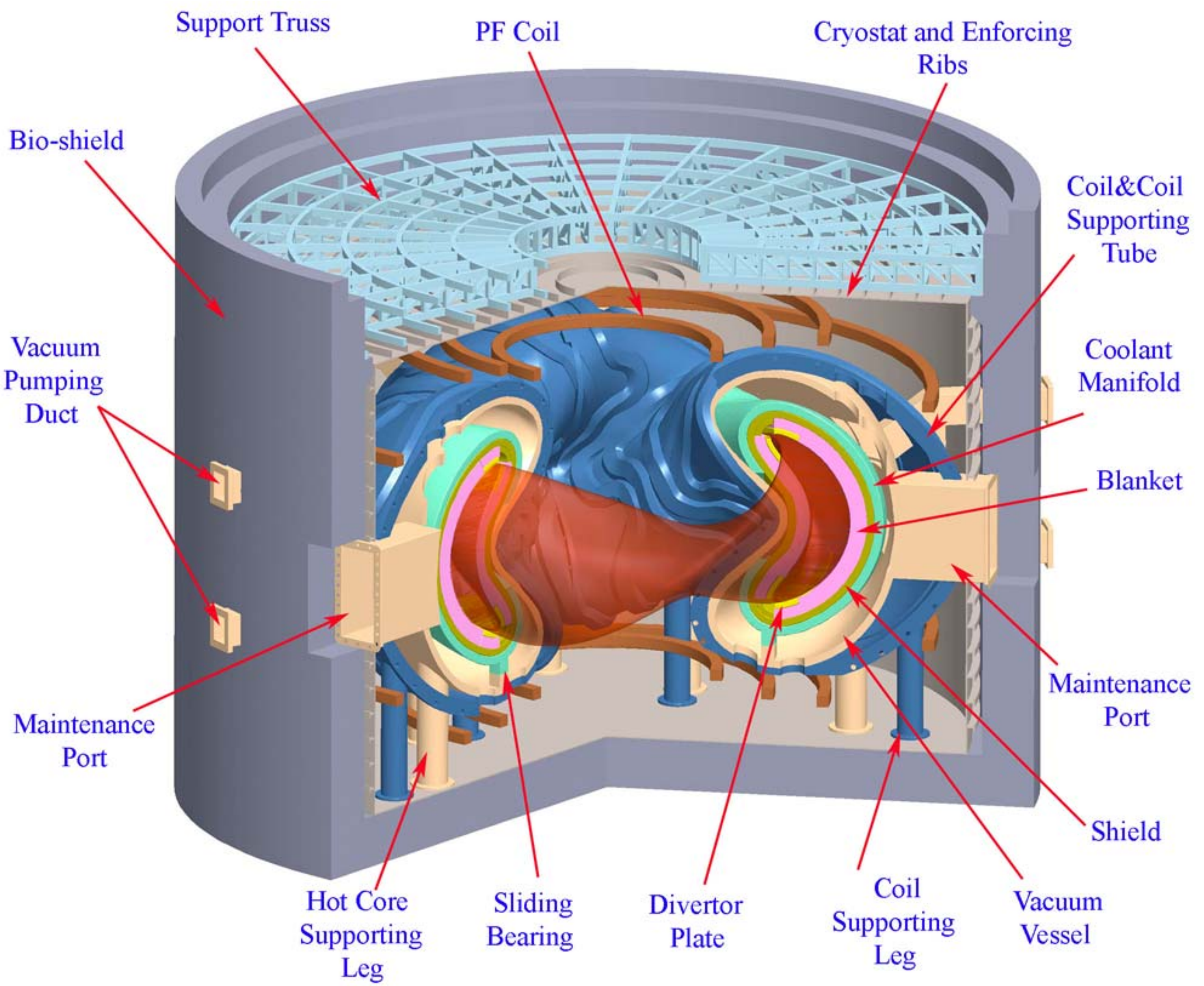
β = 5%

H(ISS95) = 1.5

I_{plasma} = 3.5 MA (bootstrap)

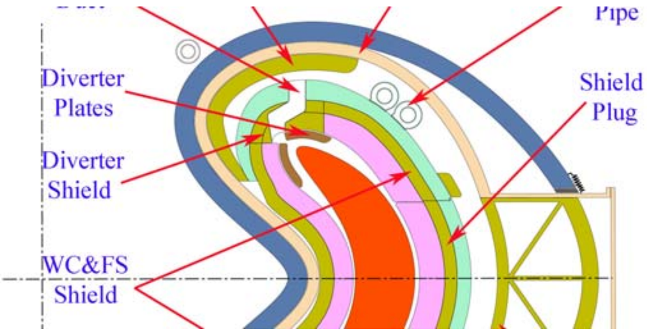
P(fusion) = 2.364 GW

P(electric) = 1 GW

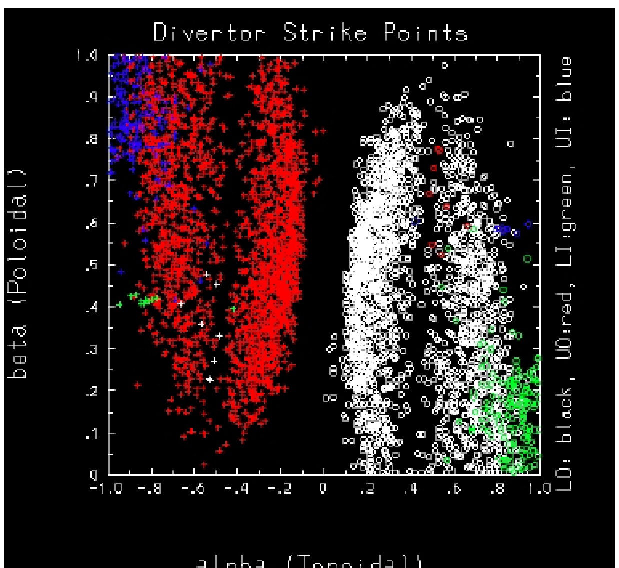


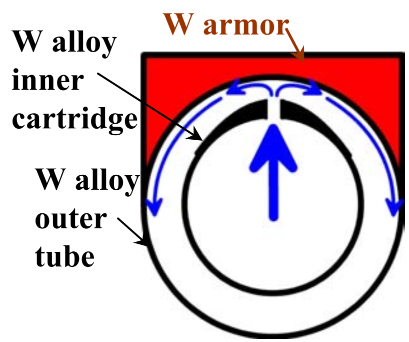
A highly radiative core is needed for divertor operation

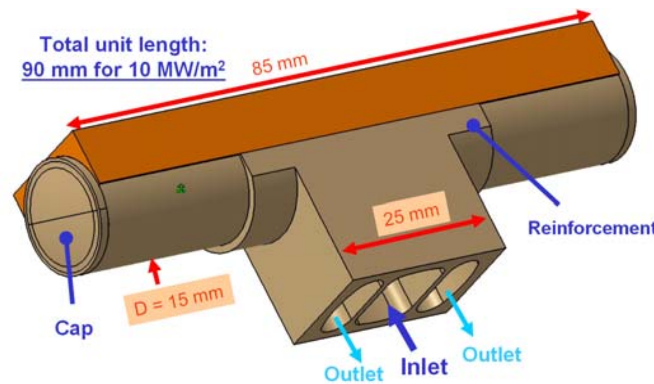
- ➢ Heat/particle flux on divertor was computed by following field lines outside LCMS.
  - ✓ Because of 3-D nature of magnetic topology, location & shaping of divertor plates require considerable iterative analysis.



Top and bottom plate location with toroidal coverage from -25° to 25°.

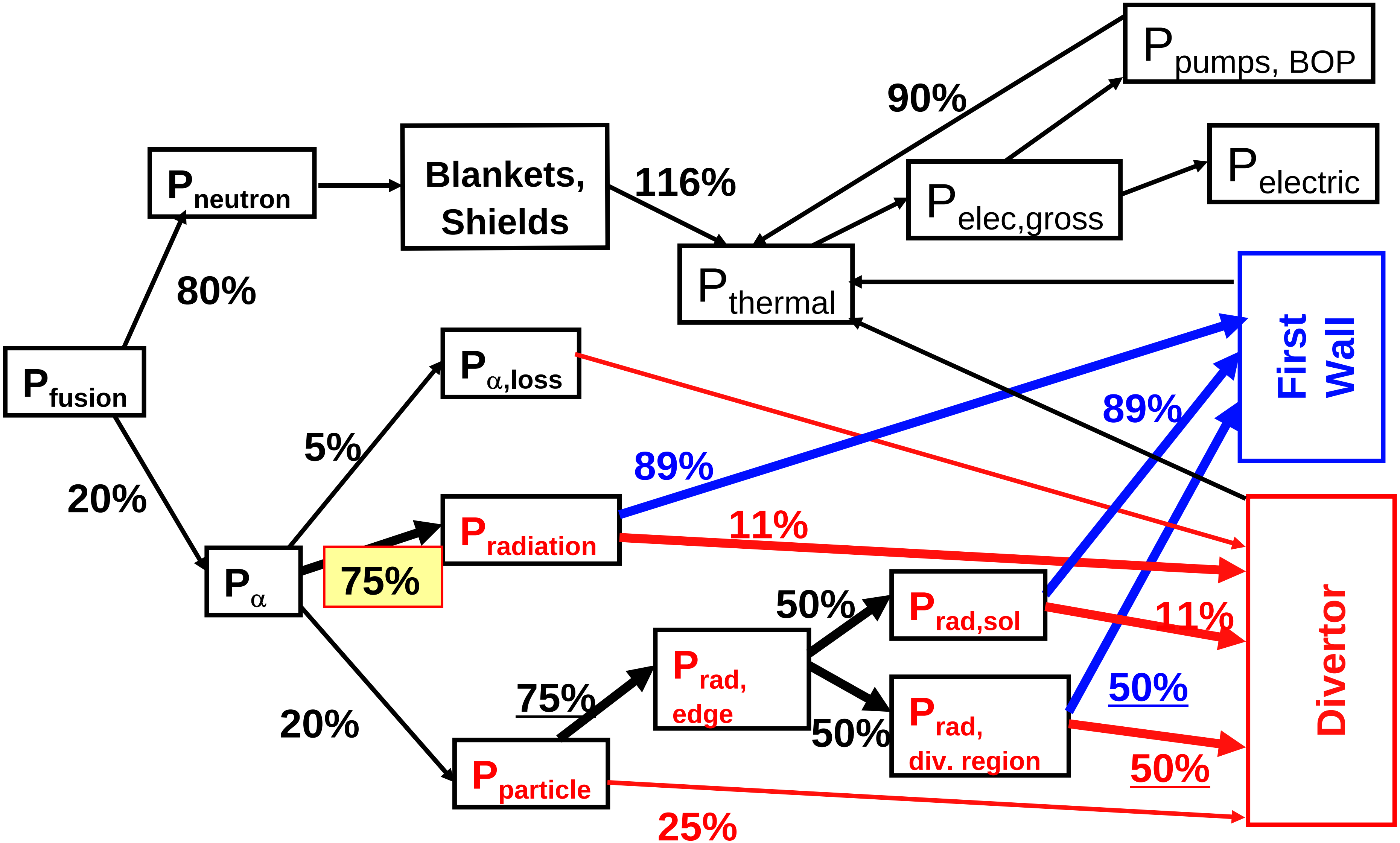






- ➢ Divertor module is based on W Cap design (FZK) extended to mid-size (~ 10 cm) with a capability of 10 MW/m^2

Divertor Solution Uses 75% Core Radiation



Stellarator Burning Plasma Needs Little Feedback Control

Typically, stellarators use simple control systems

For Stellarator Burning Plasma, expect to control:

- Burn control via density feedback
- Divertor heatflux via radiated power (impurities for edge radiation)
- Divertor strike location via plasma shape and location

- If need low m/n MHD control (e.g. Kinks), will do via control of edge iota

Need relatively simple diagnostics, few blanket penetrations.

magnetics
density
Prad
Divertor heat deposition pattern

We expect burning plasma control to be much simpler in a stellarator compared to a tokamak

MHD Equilibrium Control Must Be Developed for Stellarators

Need to ensure power goes to divertor, independent of plasma

- When commissioning, or starting up burning plasma, need controlled rise in fusion power to limit thermal stresses evolving
- May take days -> weeks

Want to preserve flux surface quality at control of <R>, iota, trim coils

May also want iota for MHD stability control

Need to develop control of 3D equilibrium

- Want to control <R>, divertor strike positions, gaps to PFCs, resonant islands
- No analytic way to determine <R>, <a> or boundary shape parameters

Realtime equilibrium control being investigated at LHD, W7-X, NCSX

- Candidate strategies: functional parameterization, neural-net, 3D external flux fit (e.g. filament code).
  Generalize to include shape and topology control.

What about RWMs and NTMs??   -limits? Profiles?

In W7AS and LHD, β limit is well above linear ideal stability threshold

- low-n modes are observed, but saturate without disrupting or preventing access to higher β.
- not clear that RWMs are a concern.
- ARIES & NCSX: linear stability threshold in already high enough to be compelling

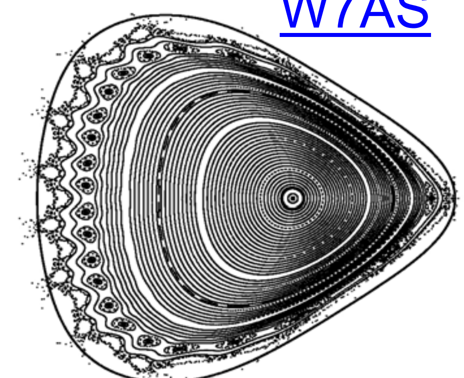
ARIES & NCSX: NTMs stabilized by reversed shear.

W7AS and LHD   -limit <u>may</u> be due to flux-surface breakup at high   .

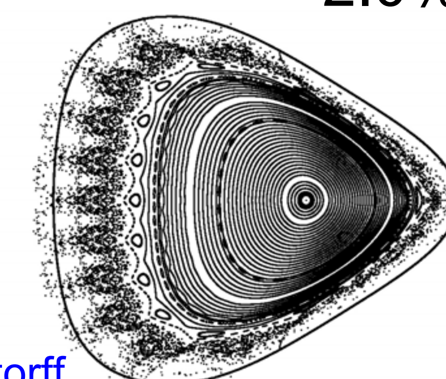
- ARIES & NCSX designed for good flux surfaces at high-
- Can potentially go further using trim coils

NCSX MHD equilibrium and stability appear relatively insensitive to profile effects.

<u>W7AS</u>



= 2.0%



= 2.7%

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A. Reiman

Summary

- Stellarator characteristics solve many challenges of MFE
  - ✓ Steady state at high-beta without need for current drive
  - ✓ No disruptions => eases PFC choices
  - ✓ High density => easier plasma solutions for divertor, Alfvenic stability
  - ✓ No need for feedback to control instabilities

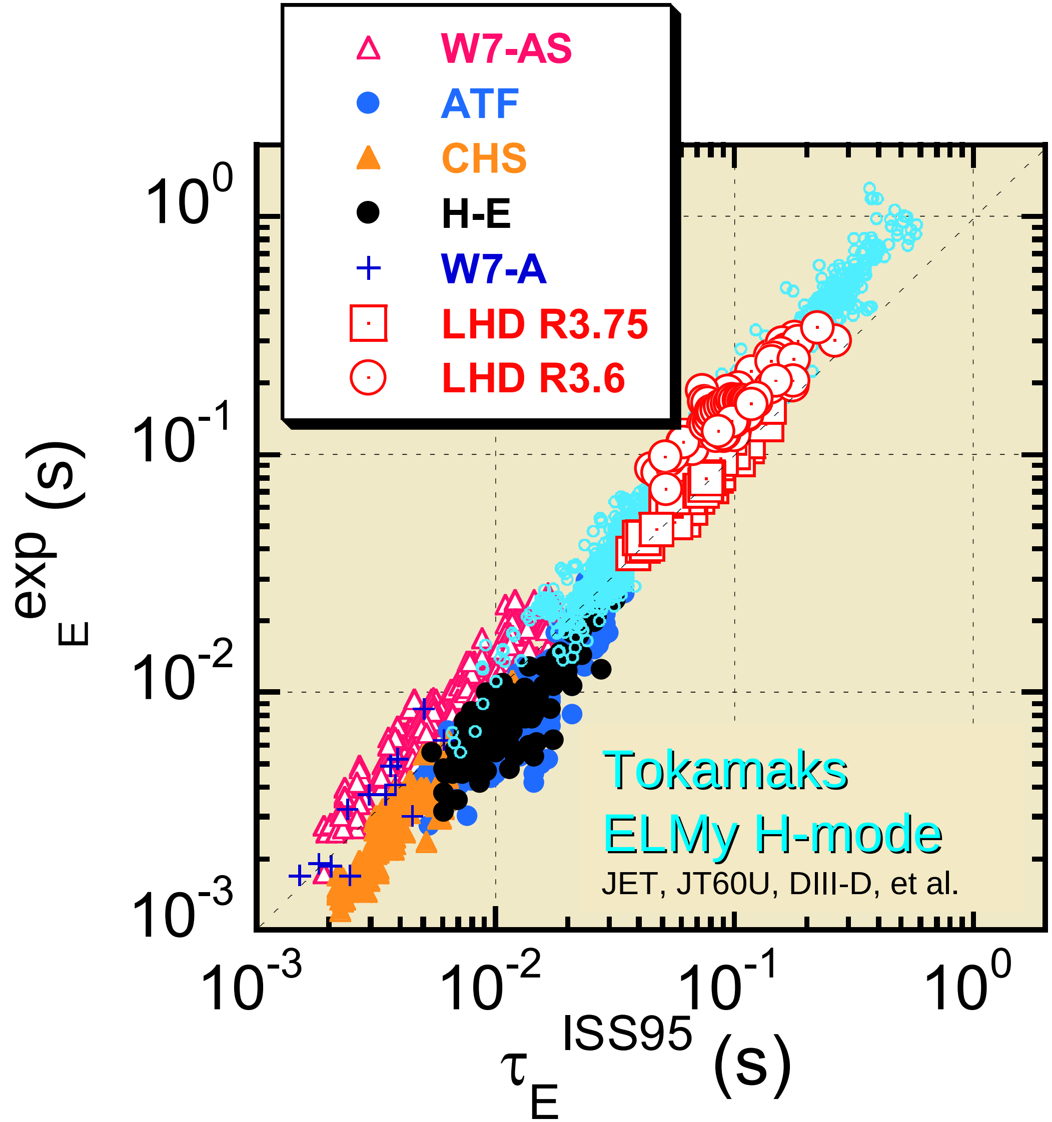
  These characteristics already available in existing experiments.

- Need for MHD control substantially reduced, relative to tokamaks
  - Primarily equilibrium control, for divertor control
  - Flux surface quality.
- Other control requirements in stellarator burning plasmas will likely be simple.

- ARIES-CS: compact stellarators project to attractive reactors
- Need to test and understand these characteristics for low A, quasi-symmetric configurations : NCSX

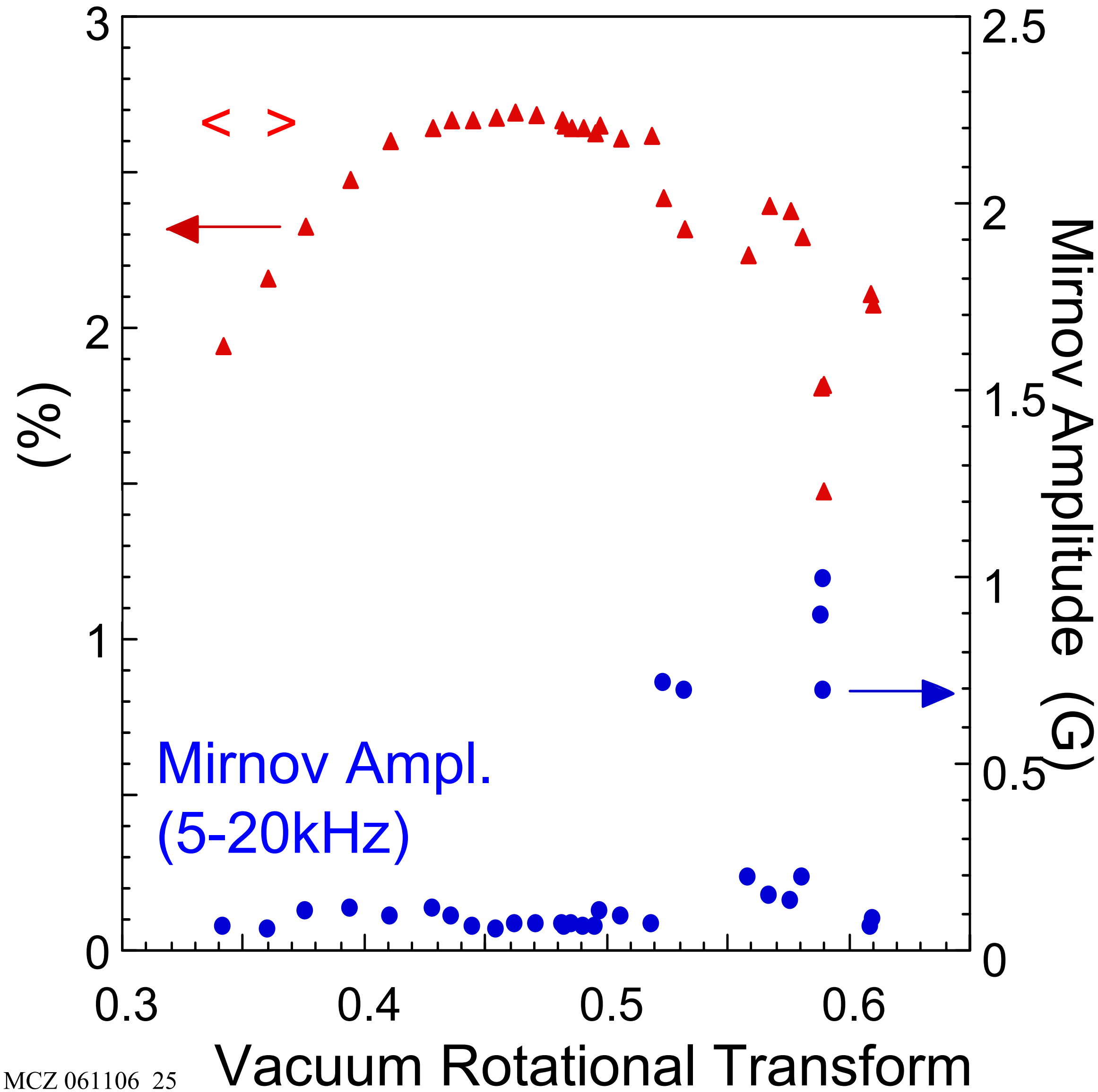
Stellarator Confinement Similar to Tokamak ELMy H-modes

- ISS-95 confinement scaling from existing experiments
- More optimized experiments achieve 1.5 – 2.5 x ISS-95
- More recent ISS-04 scaling suggests an $_{\text{eff}}^{-0.4}$ scaling, where $_{\text{eff}}$ characterizes residual ripple



$a^{2.2}R^{0.65}\ B^{0.84}$

W7AS: MHD in Narrow Iota ranges

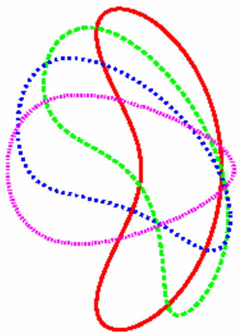
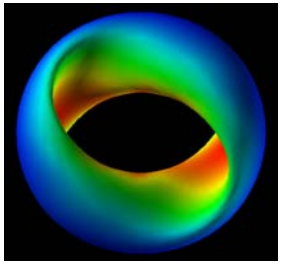
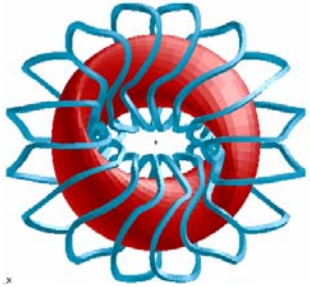
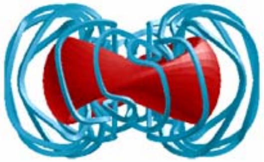


- Controlled iota scan, varying I_{TF} / I_M, fixed B, P_{NB}, flattop phase
- Strong MHD activity only in narrow ranges
  - –when total edge iota 0.5 or 0.6 (m/n=2/1 or 5/3)

Two New Classes of QA Configurations

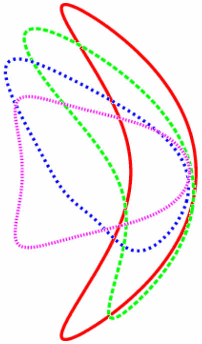
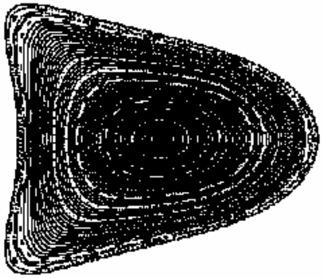
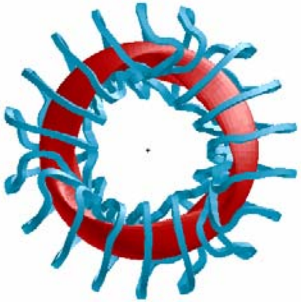
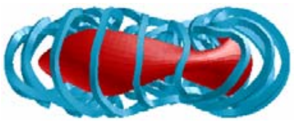
II. MHH2

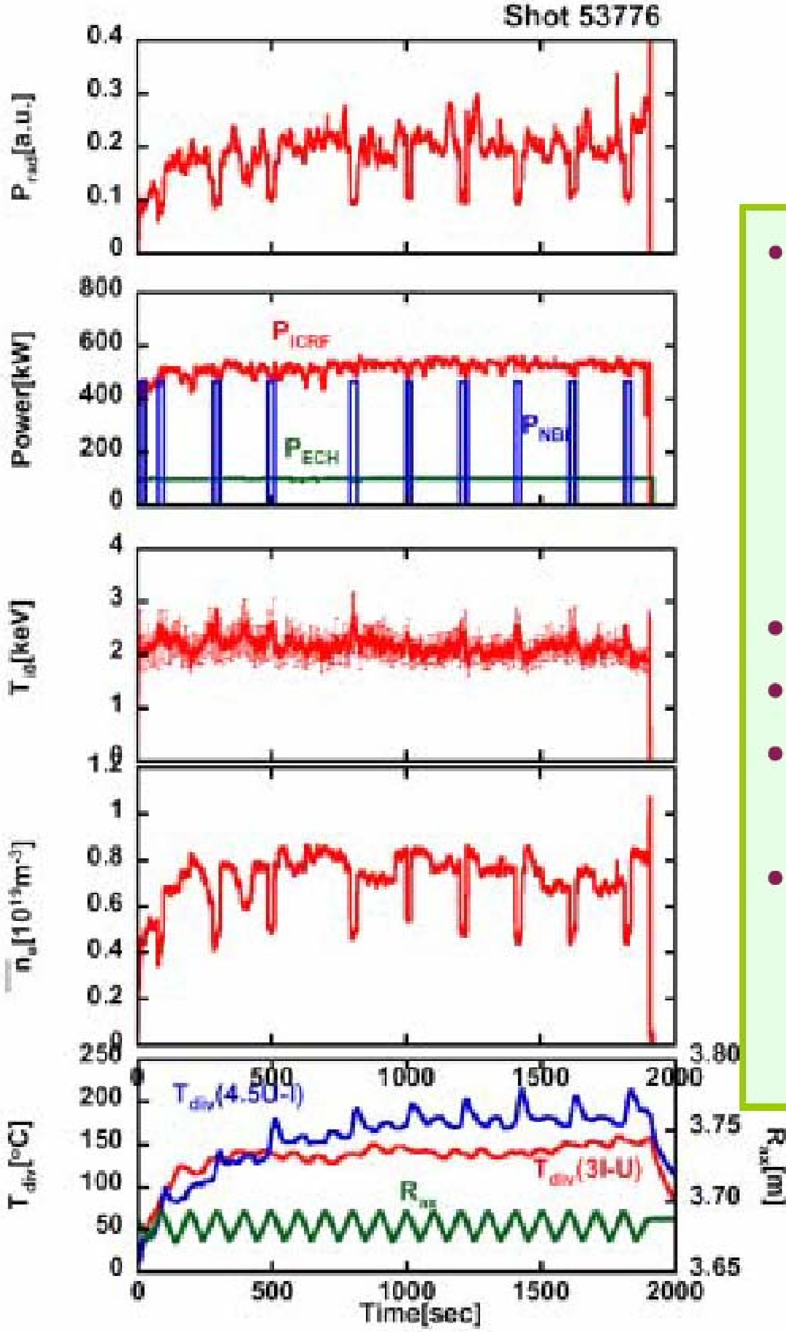
- ✓Low plasma aspect ratio ($A_p \sim 2.5$) in 2 field period.
- ✓Excellent QA, low effective ripple (<0.8%), low   energy loss (5%) .

III. SNS

- ✓$A_p \sim 6.0$ in 3 field period. Good QA, low -eff (< 0.4%),   loss 8% .
- ✓Low shear rotational transform at high , avoiding low order resonances.

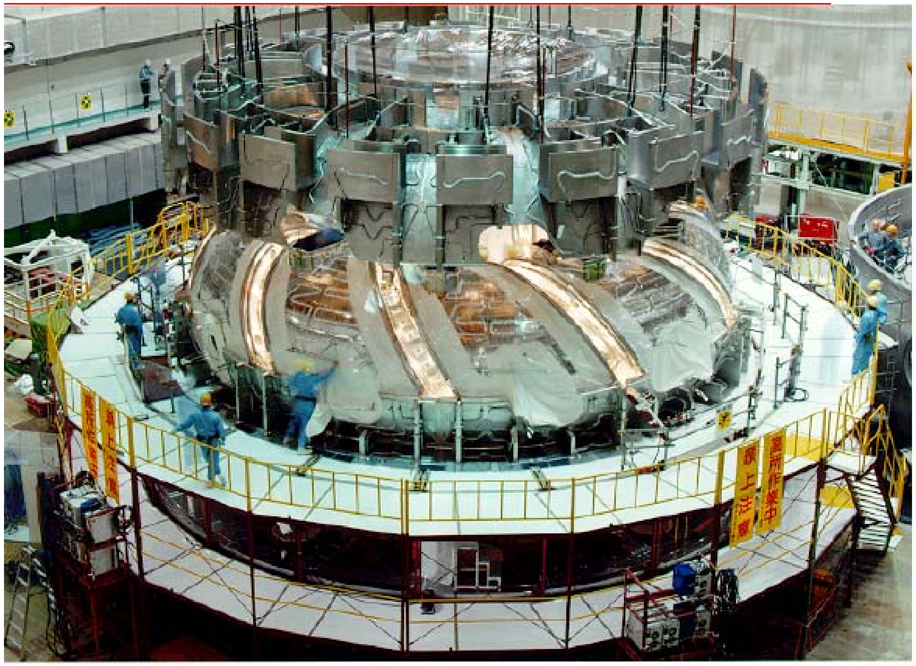


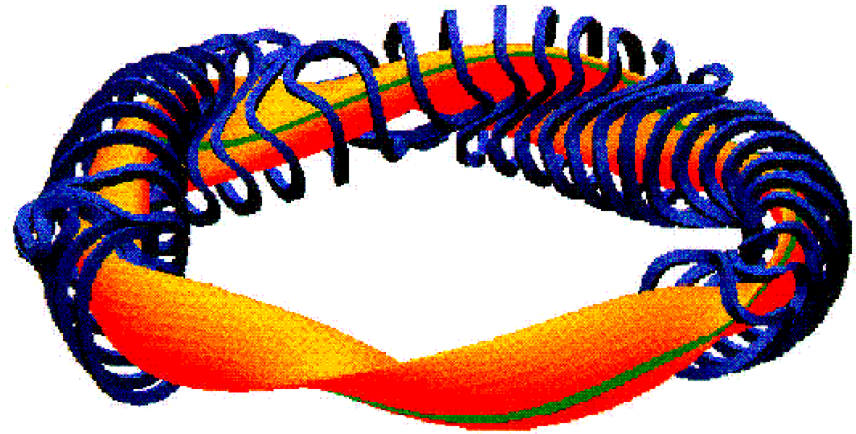
Successful 31 min. long discharge

- Combination of three heating schemes
  Average power is 680kW
  Steady state injection of ICRF(520 KW) and ECH(100 kW)
  25s pulse of NBI at intervals : 60 kW
  (averaged for one duty cycle)
- Ion temperature 2.0keV
- Electron temperature 1.3-1.7keV
- Line averaged electron density $7\text{-}8\times10^{18}\ m^{-3}$
  Density drops during NBI pulses
- Sweep of magnetic axis (one round of 3cm for 3min. 18 rounds between R_{ax}= 3.67-3.7m)
  ➔ maintain the temperature of divertor plates close to antenna at moderate level.

(B = 2.75T at R=3.6m, #53776, Helium)

The World Stellarator Program is Substantial





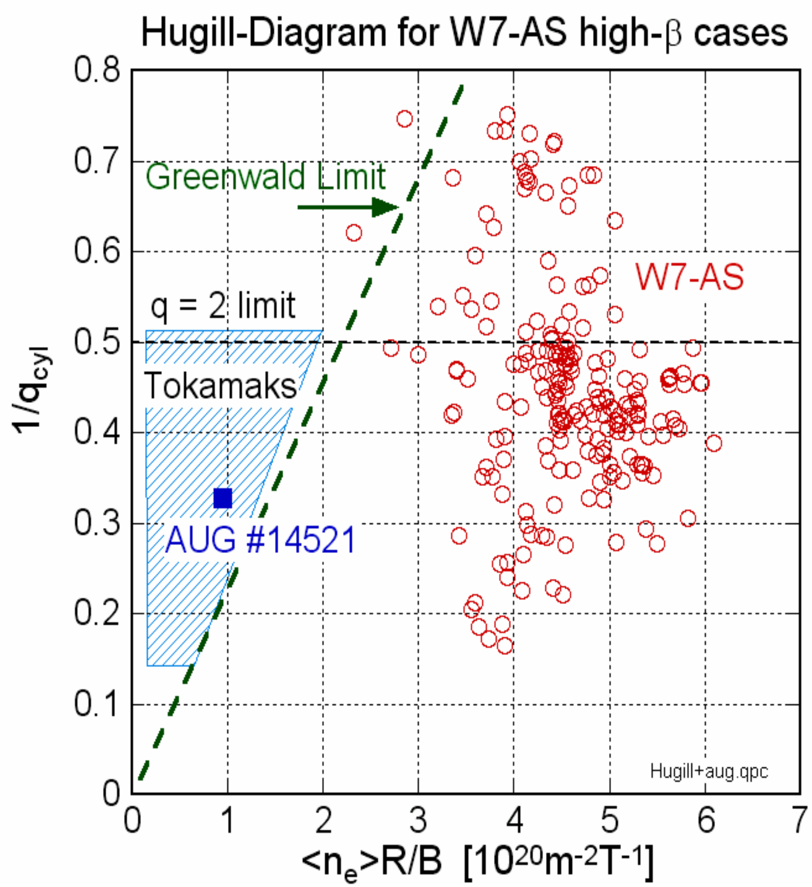
Large Helical Device (Japan)
Enhanced confinement, high ;
A = 6-7, R=3.9 m, B=3 4T

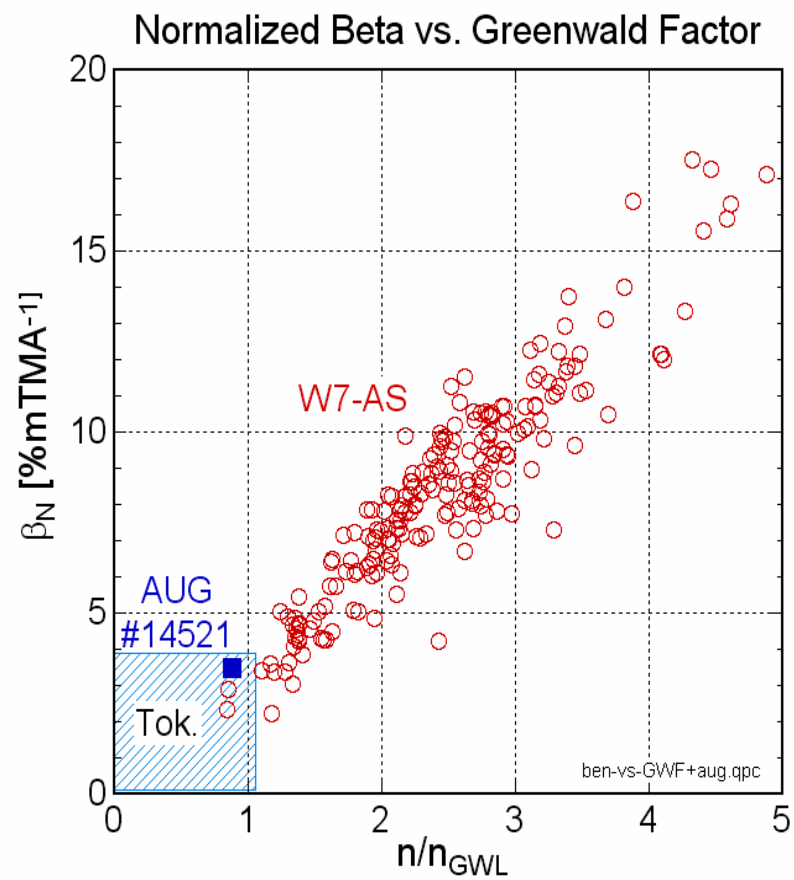
Wendelstein 7-X (Germany) (2010)
non-symmetric optimized design:
no current, A = 11, R=5.4 m, B=3T

- New large international experiments use superconducting coils for steady-state
- Medium-scale experiments (W7-AS, CHS), and
- Exploratory helical-axis experiments in Australia Japan, Spain, US.

Large aspect ratios; physics-optimized designs without symmetry, no current.

W7-AS Operating Range much larger than Tokamaks





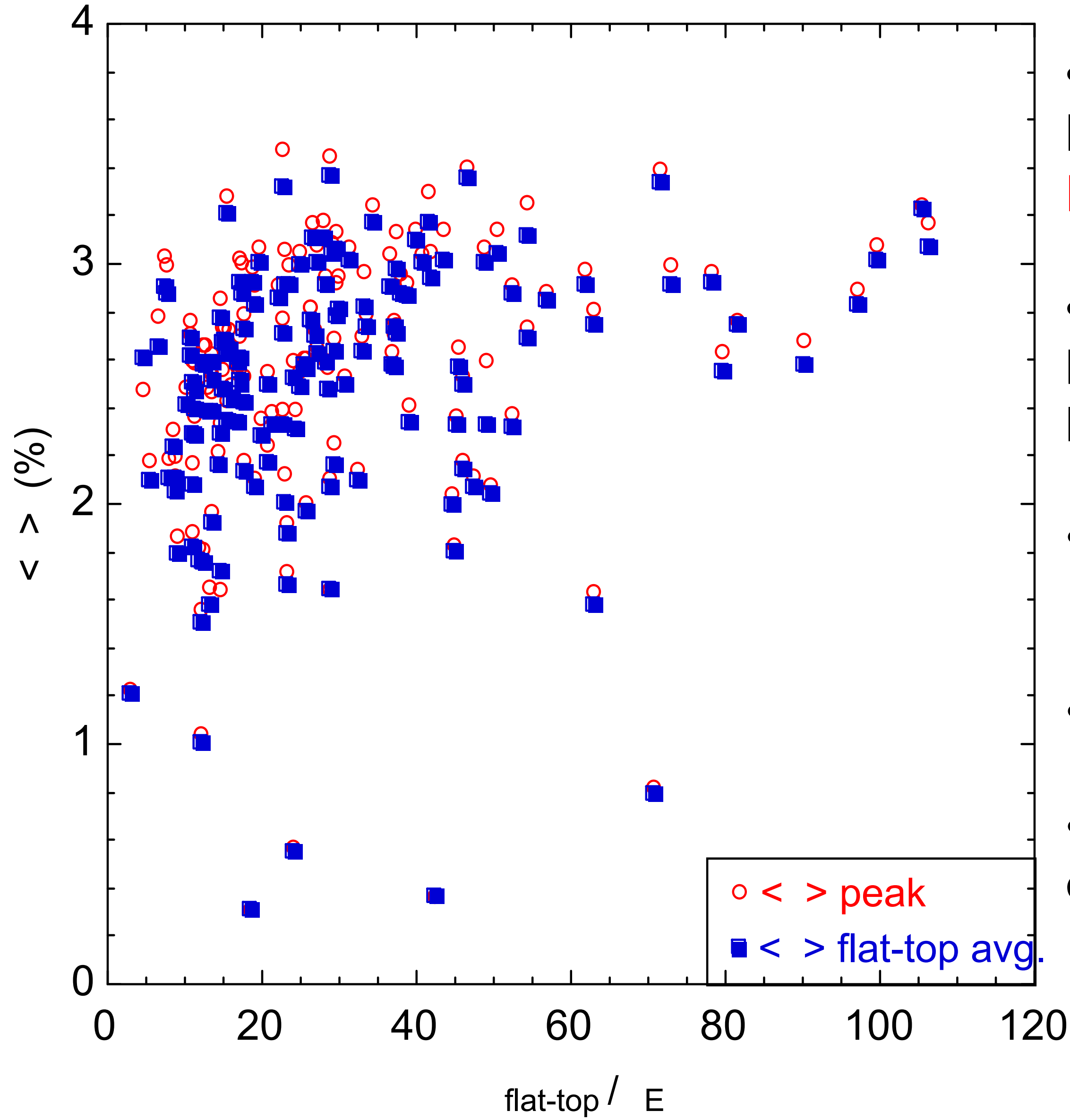
Using equivalent toroidal current that produces same edge iota

Limits are not due to MHD instabilities

high- is reached with high density (favourable density scaling in W7-AS)

All **W7-AS** high- data points beyond operational limits of tokamaks

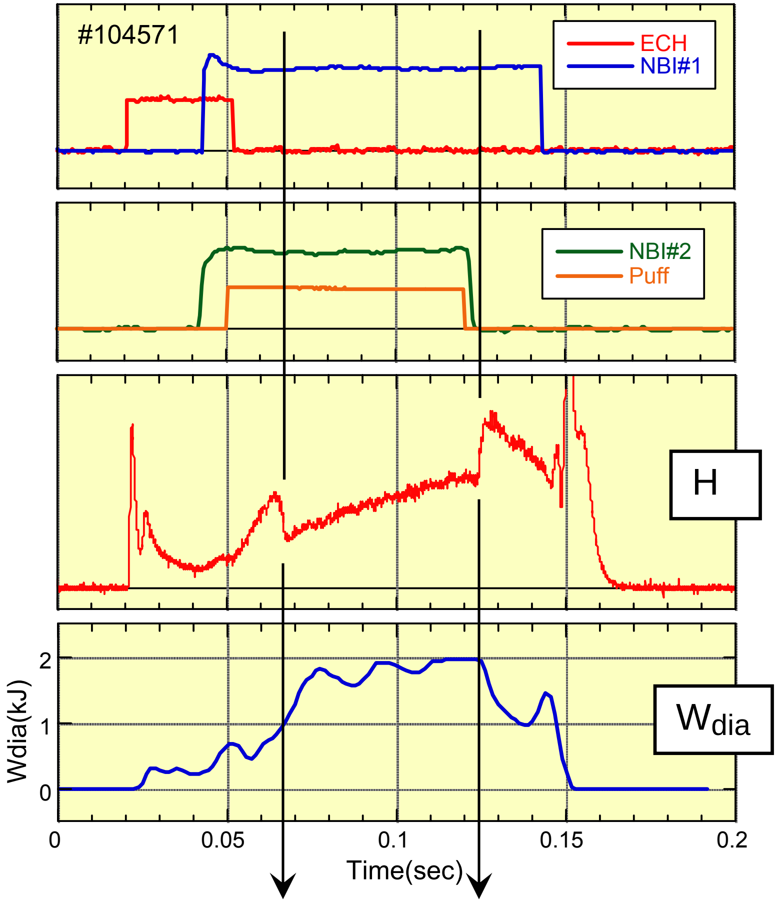
W7AS: > 3.2% maintained for > 100 E



- Tokamaks typically see beta-limit dropping for longer pulses Not observed on W7-AS.
- High- maintained as long as heating maintained, up to power handling limit of PFCs.
- -peak -flat-top-avg very stationary plasmas
- No disruptions
- **Duration and β not limited by onset of observable MHD**

Stellarator H-modes and Edge Barriers similar to tokamaks

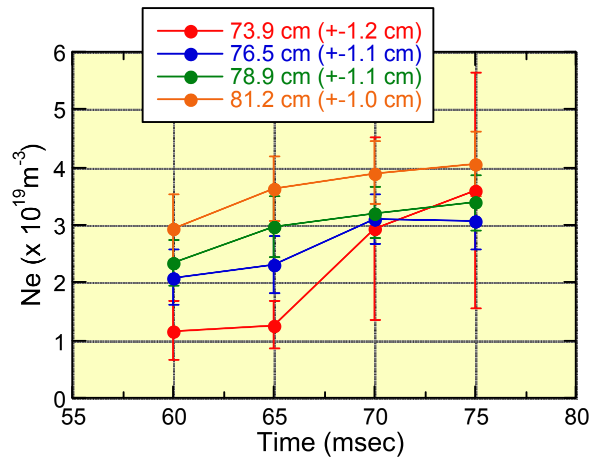
CHS



Transition to ETB

Back Transition

Thomson measurement shows edge density increases at transition



Two NBIs, B = 0.95 T

R_{ax} = 92.1 cm

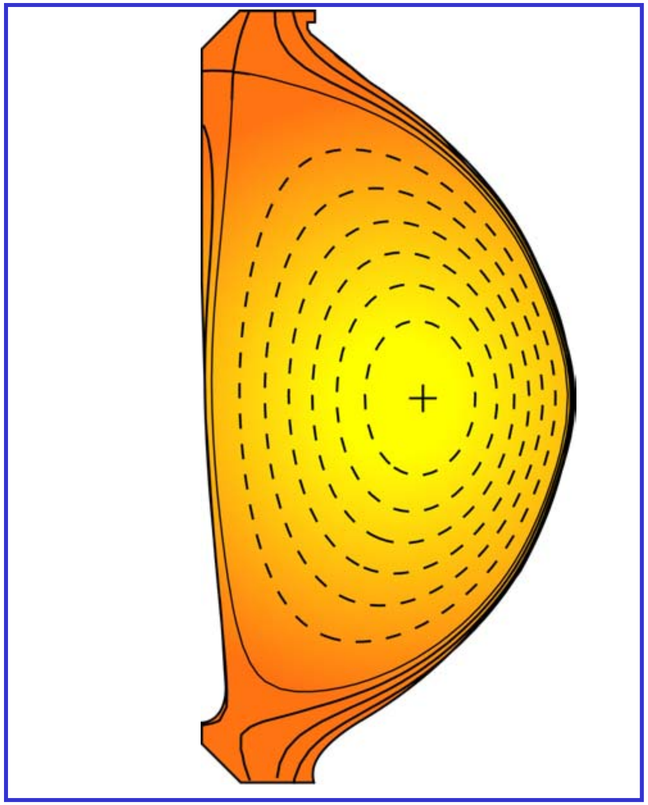
- S. Okamura

- Drop in Ha, broadening of density & pressure, increase in confinement
- Also observed on W7-AS, LHD, Heliotron-J
- Any ELM-like events appear small

Fusion Plasma Challenges for Reactors

e.g. NAS Burning Plasma Report

- Macroscopic Stability
  - Maximize plasma pressure
  - No disruptions
- Transport & Microturbulence
  - Adequate energy confinement
  - 3D: suppression of ripple-transport
- Wave-particle Interactions
  - Successful alpha heating
  - 3D: alpha orbit confinement
- Plasma-material Interactions
  - First wall survivability, exhaust
- Configuration Sustainment



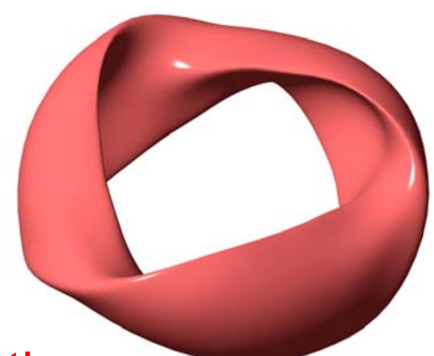
NCSX Motivation: Build Upon and Combine Advances of Stellarators and Tokamaks

Tokamaks:

- Confirmation of ideal MHD equilibrium & stability theory; neoclassical transport theory; neoclassical tearing-mode theory
- Importance of flows (including self-generated) for turbulence stabilization
- Reversed shear to reduce turbulence, increase stability
- Effectiveness of plasma shaping to control plasma physics
- Compact   cost-effective

Stellarators:

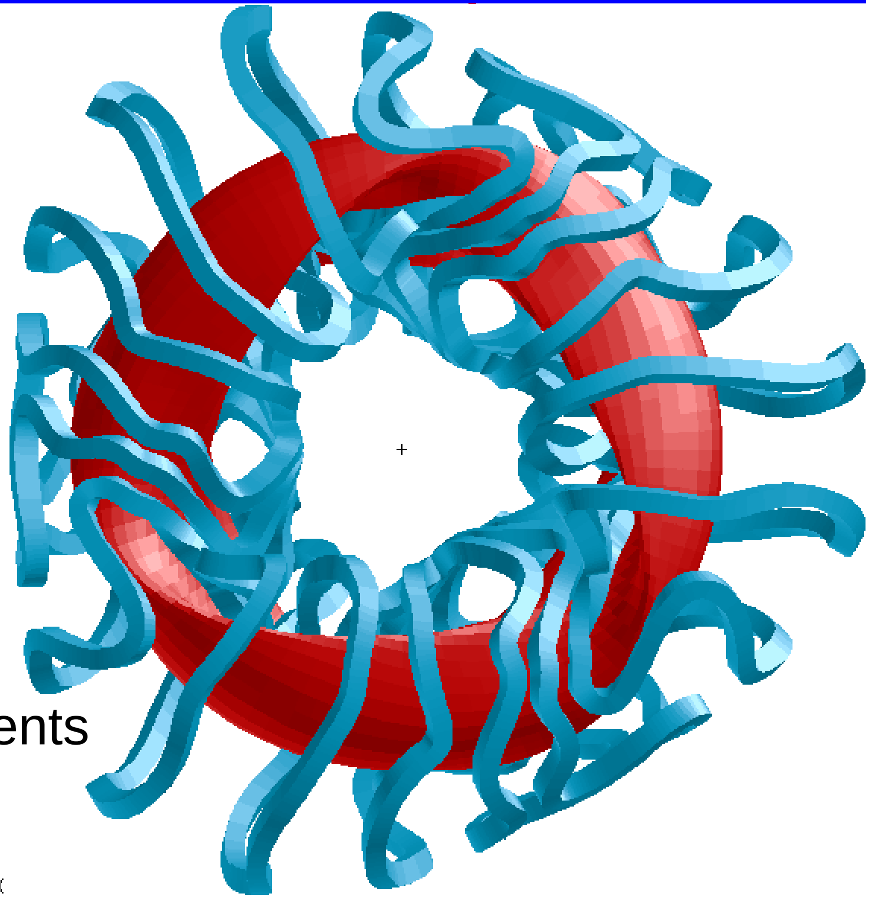
- Externally-generated helical fields
  - Plasma current not required. No current drive. Steady-state easy.
  - Robust stability
  - Generally, disruption-free
- Numerical design of 3D shape to obtain desired physics properties



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

NCSX Designed for Attractive Properties

- 3 periods, R/ a =4.4, ~1.8 , ~1
  ¾ of rotational transform from coils,
  ‘reversed shear’ across whole plasma
- Quasi-axisymmetric
- Passively stable at =4.1% to kink,
  ballooning, vertical, Mercier,
  neoclassical-tearing modes
- Stable for > 6% by adjusting coil currents
- Passive disruption stability: equilibrium
  maintained even with total loss of or I_P
- Flexible coils: by adjusting currents can control
  stability, transport; shape: iota, shear



NCSX Research Mission

Acquire the physics data needed to assess the attractiveness of compact stellarators; advance understanding of 3D fusion science.

(FESAC-99 Goal)

Understand…

- Pressure limits and limiting mechanisms in a low-A optimized stellarator
- Effect of 3D magnetic fields on disruptions
- Reduction of neoclassical transport by quasi-axisymmetric design.
- Confinement scaling; reduction of turbulent transport by flow shear control.
- Equilibrium islands and tearing-mode stabilization by design of magnetic shear.
- Compatibility between power and particle exhaust methods and good core performance in a compact stellarator.
- Energetic-ion stability in compact stellarators

Demonstrate…

- Conditions for high normalized pressure disruption-free operation
- High pressure, good confinement, compatible with steady state

Quasi-Axisymmetric: Very Low effective ripple

|B| approximately constant in toroidal direction. Transport similar to tokamaks

Very low effective helical ripple (deviation from perfect symmetry)

$_{eff}$ ~ 1.4% at edge

< 0.1% in core

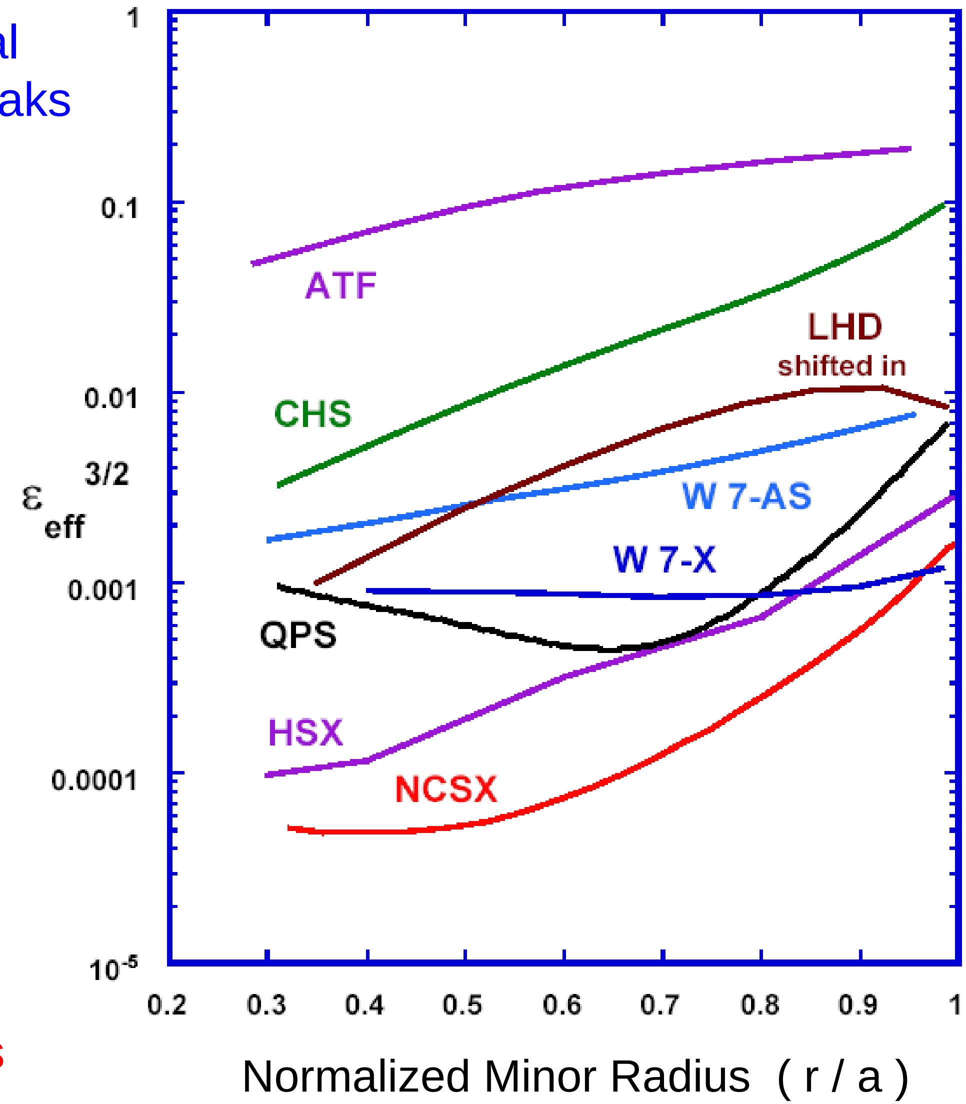
Gives low flow-damping

allow manipulation of flows for flow-shear stabilization, control of E_r

Can vary ripple to study:

- Effects of flow damping
- Interaction of 3D filed with fast ion confinement

Understand 3D effects in tokamaks and ITER

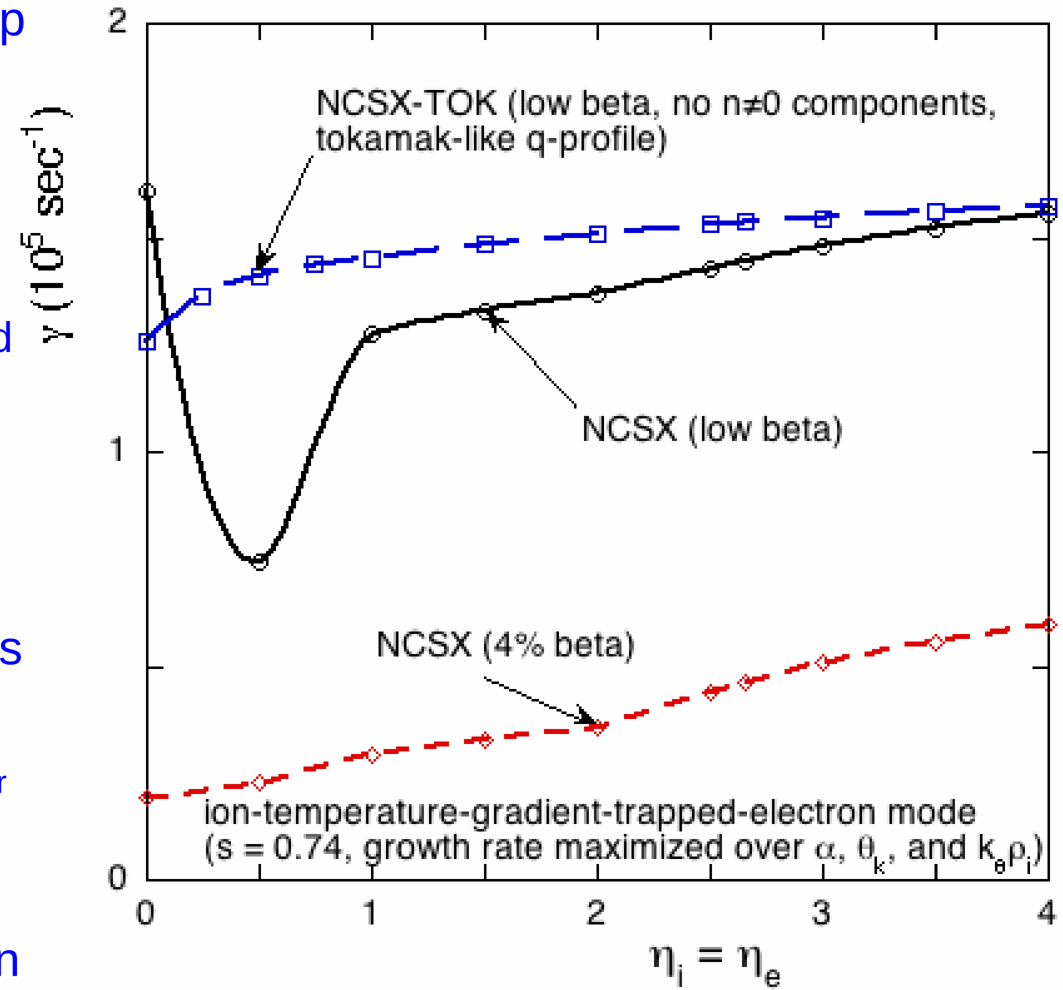


Turbulence Growth Decreases for Higher p
Similar to Reversed Shear Tokamak

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
- Similar to reversed shear tokamak



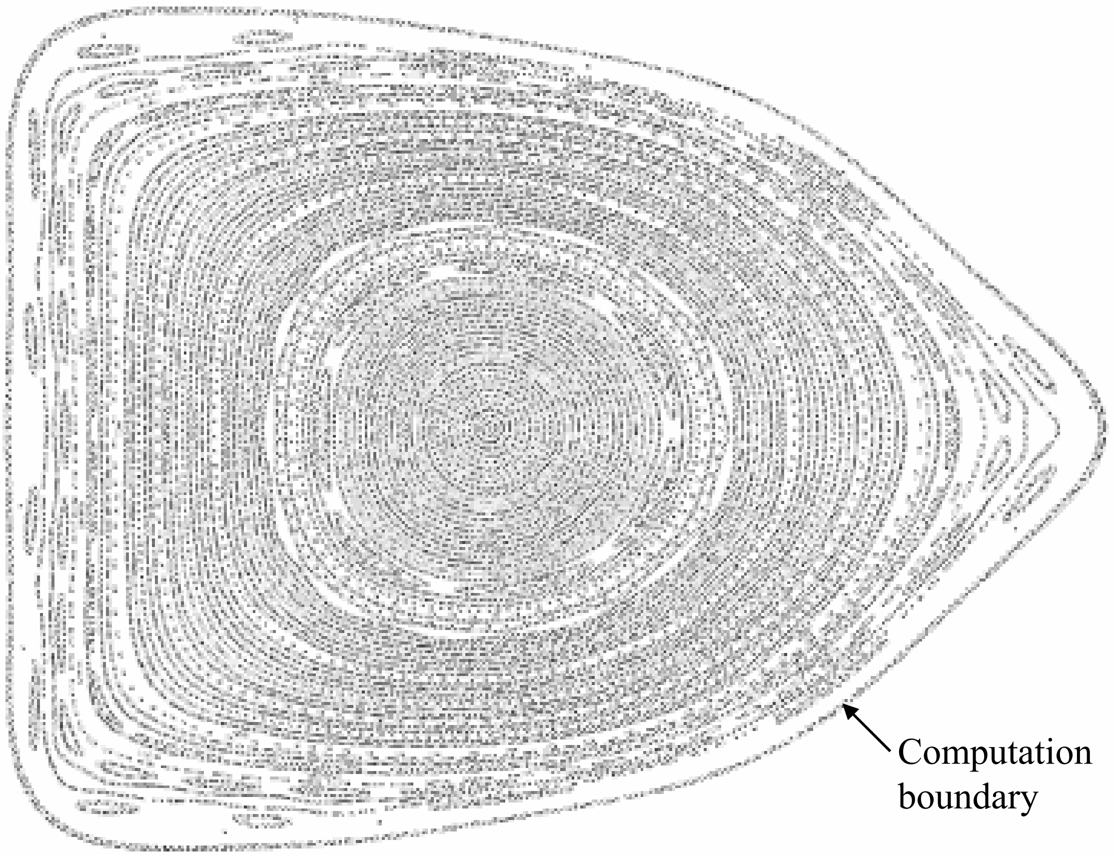
G. Rewoldt

Very low effective helical ripple gives low flow-damping allows efficient flow-shear stabilization, control of E_r

Persistent self-generate flows estimated to be similar or larger than tokamak

(using Sugama & Watanabe, 2005)

Coils Designed to Produce Good Flux Surfaces at High-



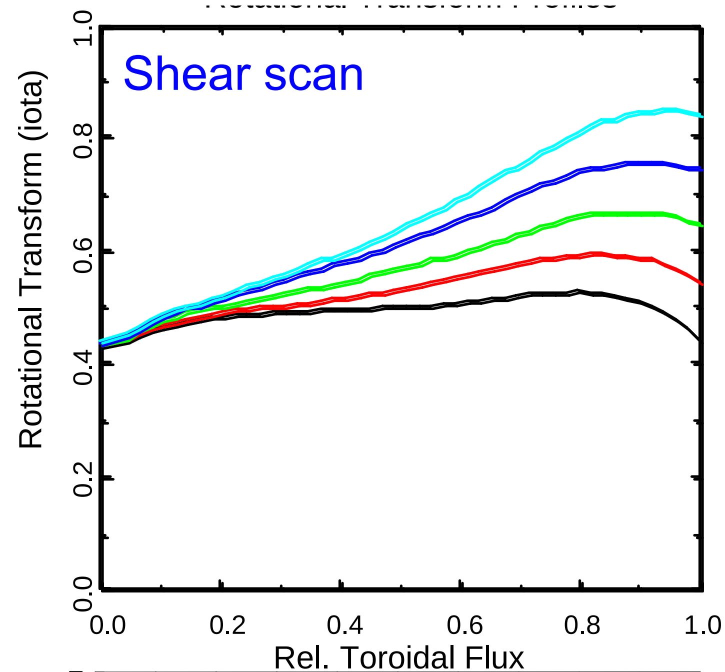
Poincare: PIES, free boundary without pressure flattening

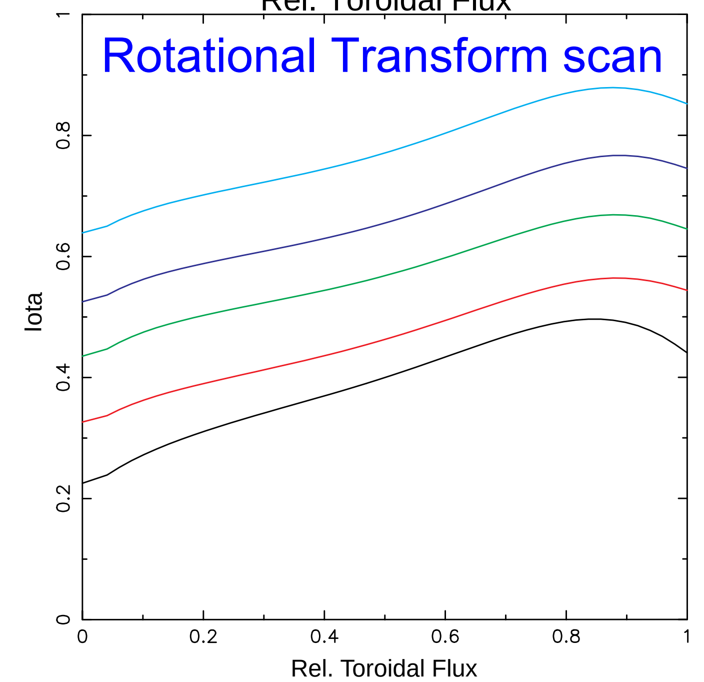
< 3% flux loss, including effects of neoclassical healing and || vs. transport.

- Explicit numerical design to eliminate resonant field perturbations
- ‘Reversed shear’ configuration neoclassical healing of equilibrium islands and stabilization of tearing modes (already observed in LHD)
- Using stellarator analysis tools in collaborations to understand effect of 3D perturbations on tokamaks, preparing for ITER

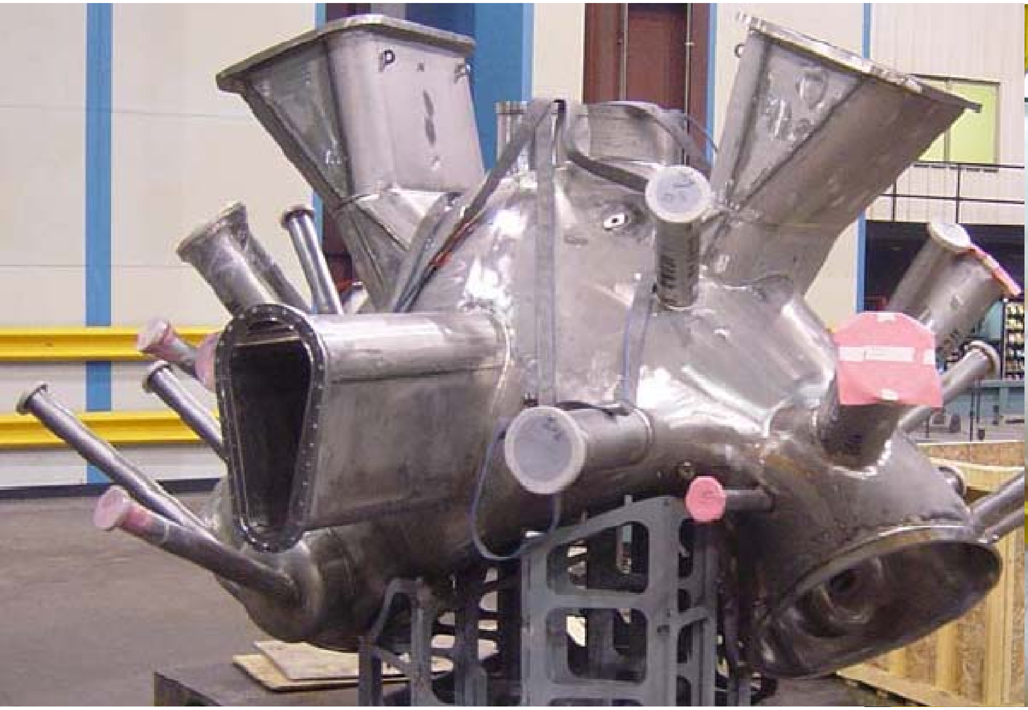
NCSX Coils Designed for Flexibility

- Modular Coils + Toroidal Solenoid + Poloidal Coils, for shaping control & flexibility
- Useful for testing understanding of 3D effects in theory & determining role of iota-profile
- E.G., can use coils to vary
  - effective ripple by factor > 10.
  - Avg. magnetic shear by factor > 5
  - Edge rotational transform by factor of 2
- Reduce kink-instability threshold down to 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



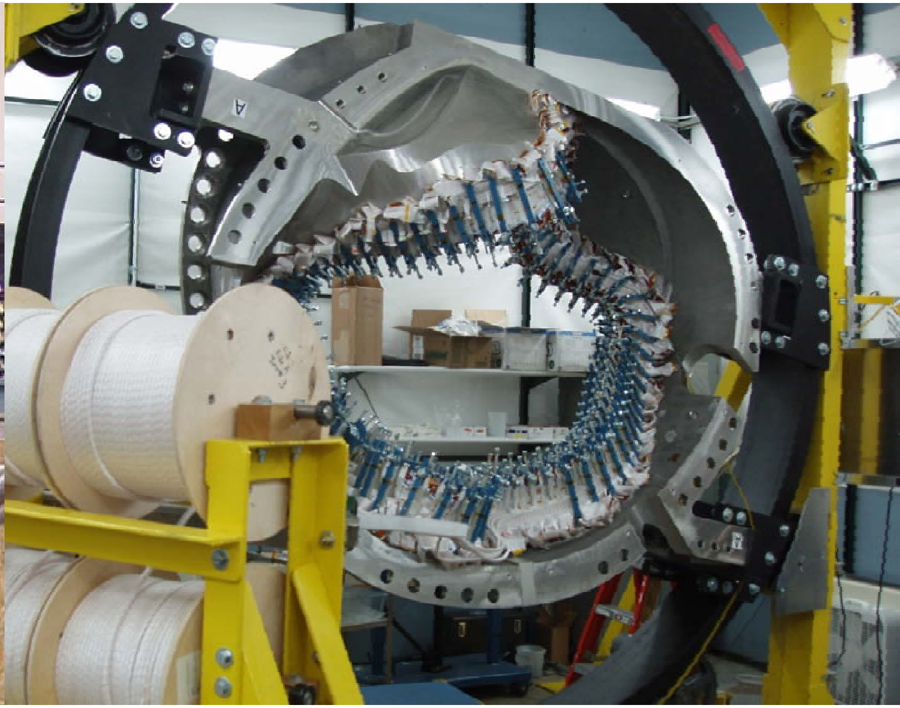


NCSX Construction is Well Underway



Vacuum Vessel sector fabrication

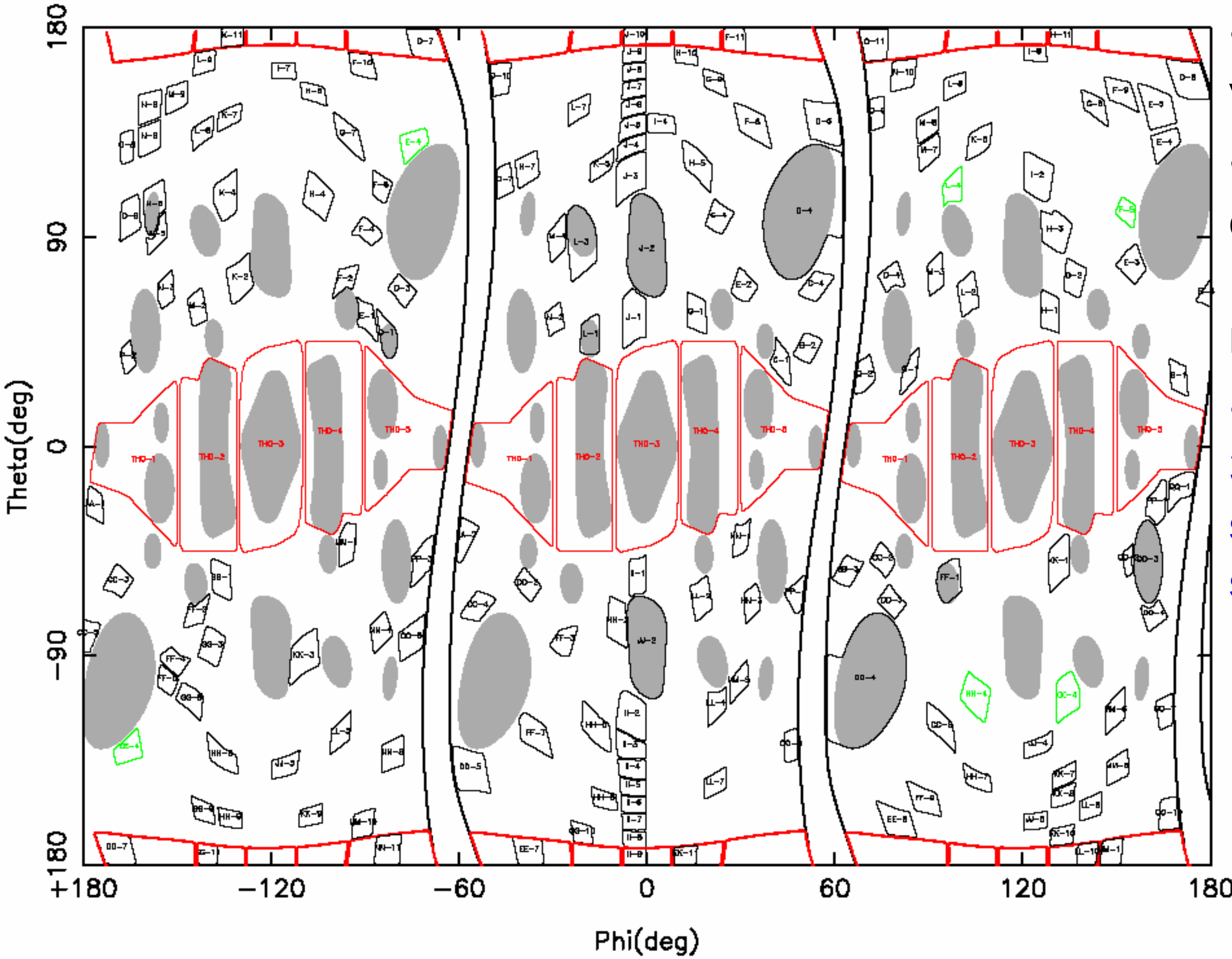
- All plates formed
- Two periods fabricated (of 3)
- One leak chased.



Modular Coil Winding Form during winding

- Thirteen winding forms cast (of 18)
- Three being machined
- Two being wound

Ex-Vessel Magnetic Diagnostics Designed for Reconstuction



- **saddle coils mounted on vessel**
- **~2500 free-boundary equililbria analyzed to identify critical regions for measurement**
- **Array distributed across 3 periods + extra coils to sense symmetric and non-symmetric components**

N. Pomphrey
E. Lazarus

Several strategies being developed for equilibrium reconstuction:

- V3FIT – reconstruction code based on VMEC (cannot represent islands)
- PIES – 3D equilibrium with islands
- 3D external flux fit (e.g. filament code), to find boundary shape and characteristics

NCSX Feedback Control under Development

Typically, stellarators use simple control systems

- Ip, density
- Pre-programmed helical coil currents
- More complicated feedback not crucial, since no disruptions and plasma does not show hysteresis (for low Ip)
- No analytic way to determine <R>, <a> or boundary shape parameters

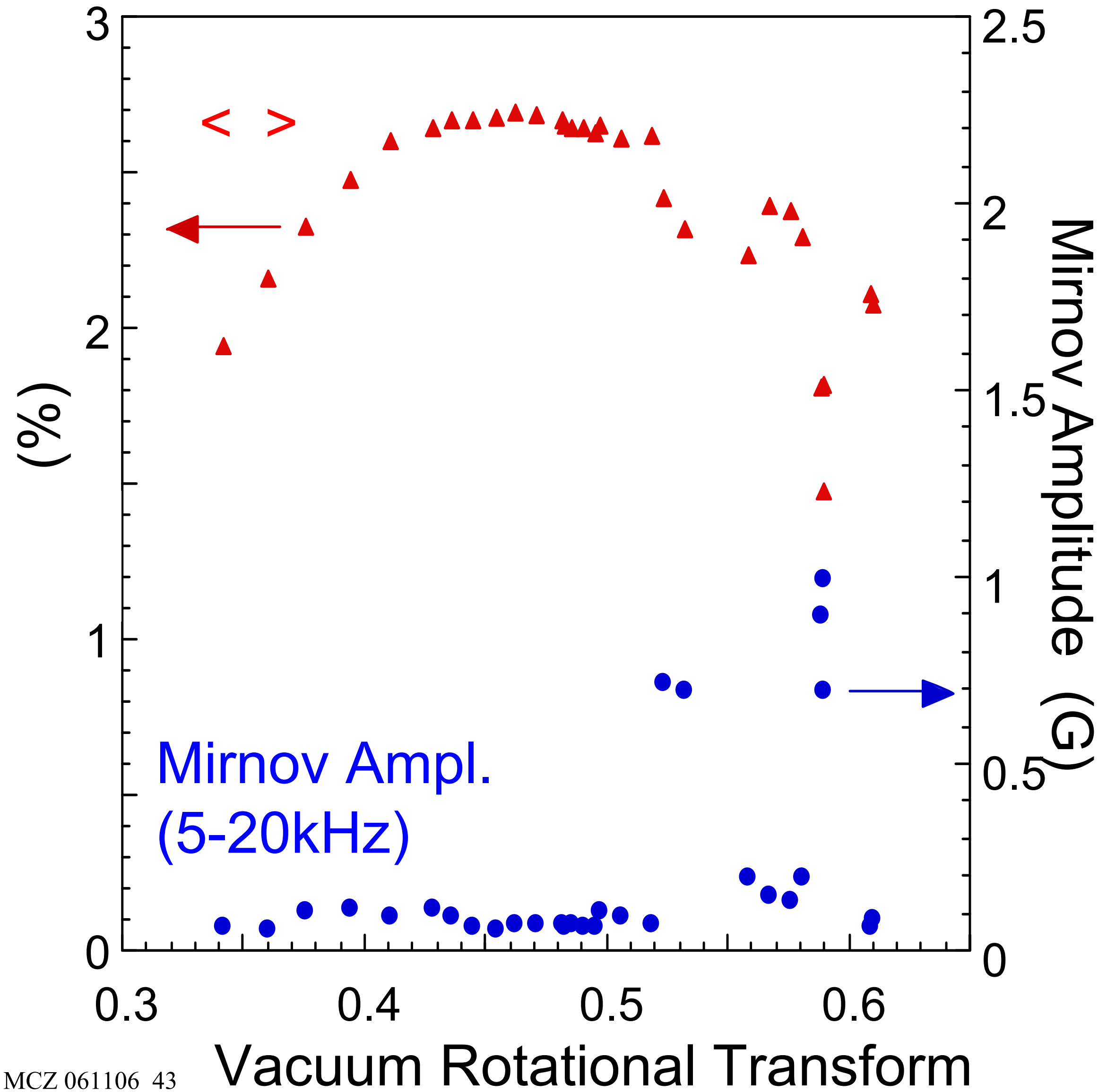
NCSX: want feedback methods to

- From W7AS results: want to control rotational transform to avoid low-order edge resonances (e.g. iota(a) = ½).
- For high-   experiments, want to control <R>
- Eventually: divertor deposition, islands, …

- Plan: use extensive magnetic diagnostics for realtime boundary calc.
- Candidate strategies: functional parameterization, neural-net, 3D external flux fit (e.g. filament code). Generalize to include shape and topology control.

- Initially, will start with simple system & develop capability

W7AS: MHD in Narrow Iota ranges



- Controlled iota scan, varying I_{TF} / I_{M}, fixed B, P_{NB}, flattop phase
- Strong MHD activity only in narrow ranges
  - –when total edge iota 0.5 or 0.6 (m/n=2/1 or 5/3)

What about RWMs and NTMs?? -limits?

In W7AS and LHD, β limit is well above linear ideal stability threshold

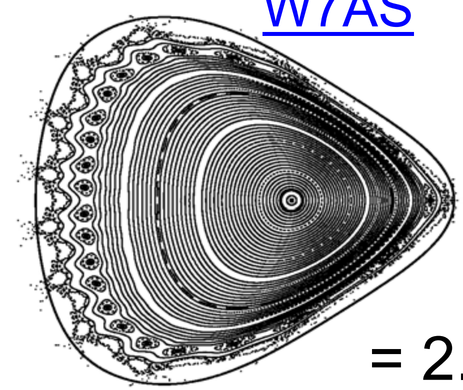
- low-n modes are observed, but saturate without disrupting or preventing access to higher β.
- not clear that RWMs are a concern.
- NCSX: linear stability threshold in already high enough to be compelling (without CD or RWM feedback)

NCSX: NTMs stabilized by reversed shear.

W7AS and LHD -limit <u>may</u> be due to flux-surface breakup at high .

- NCSX designed for good flux surfaces at high-
- Can potentially go further using trim coils

<u>W7AS</u>



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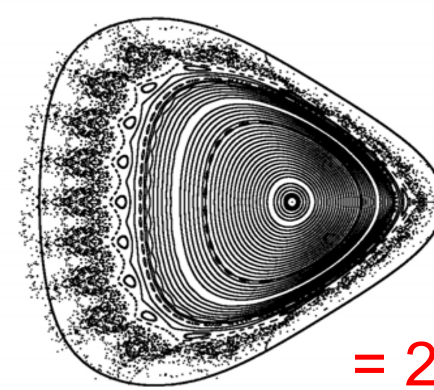


= 2.7%

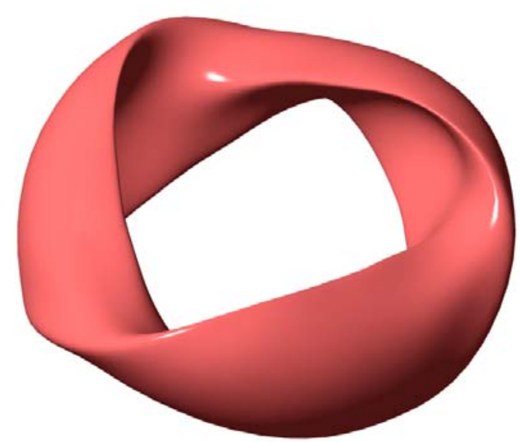
M.C. Zarnstorff
A. Reiman

Summary

- Stellarator characteristics solve current major challenges of MFE
  - ✓ Steady state at high-beta without need for current drive
  - ✓ No disruptions => eases PFC choices
  - ✓ High density => easier plasma solutions for divertor
  - ✓ No need for feedback to control instabilities
  - ✓ Projects to ignition
- Quasi-symmetry offers solutions to confinement
  - Quasi-axisymmetry should connect to tokamak confinement experience
- Compact designs developed. Project to competitive size reactors.
- Scaling of confinement to reactor regime is uncertain.
- New experiments arriving, optimized for orbit confinement,   , A
  - First results from HSX: Quasi-symmetry matters! Flow damping reduced.
  - W7-X and NCSX under construction. Optimized for   and confinement.
  - QPS: proposed for construction. Quasi-poloidal at low aspect ratio.

Conclusions

- NCSX is an exciting opportunity for unique fusion-science research.
  - Stabilize high- instabilities with 3D shaping with > 4%
  - High no-wall -limit without driven currents
  - Tokamak-like transport using quasi-axisymmetry.
  - Low ripple-transport, low rotation damping
  - Flexible coil system
- NCSX magnetics being designed for reconstruction and control of equilibrium
  - Several methods being developed.
- NCSX will have a trim coil array for
  - Control of resonant perturbations -> islands
  - Fine control of 3D shaping
  - Control of divertor strike point
- NCSX construction underway!
  - First plasma: FY09



NCSX Trim Coil Design

Trim coils have been very effective on existing experiments:

- W7AS and LHD, small saddle trim coils are used to control resonant fields to control islands
- On W7AS, trim coil was used to increase the maximum beta by ~50%, probably by controlling the edge magnetic stochasticity

NCSX external trim coils being designed for

- Control resonant field perturbations from assembly errors and plasma currents
- Give fine control on 3D plasma shape, to control physics
- Divertor strike-point control

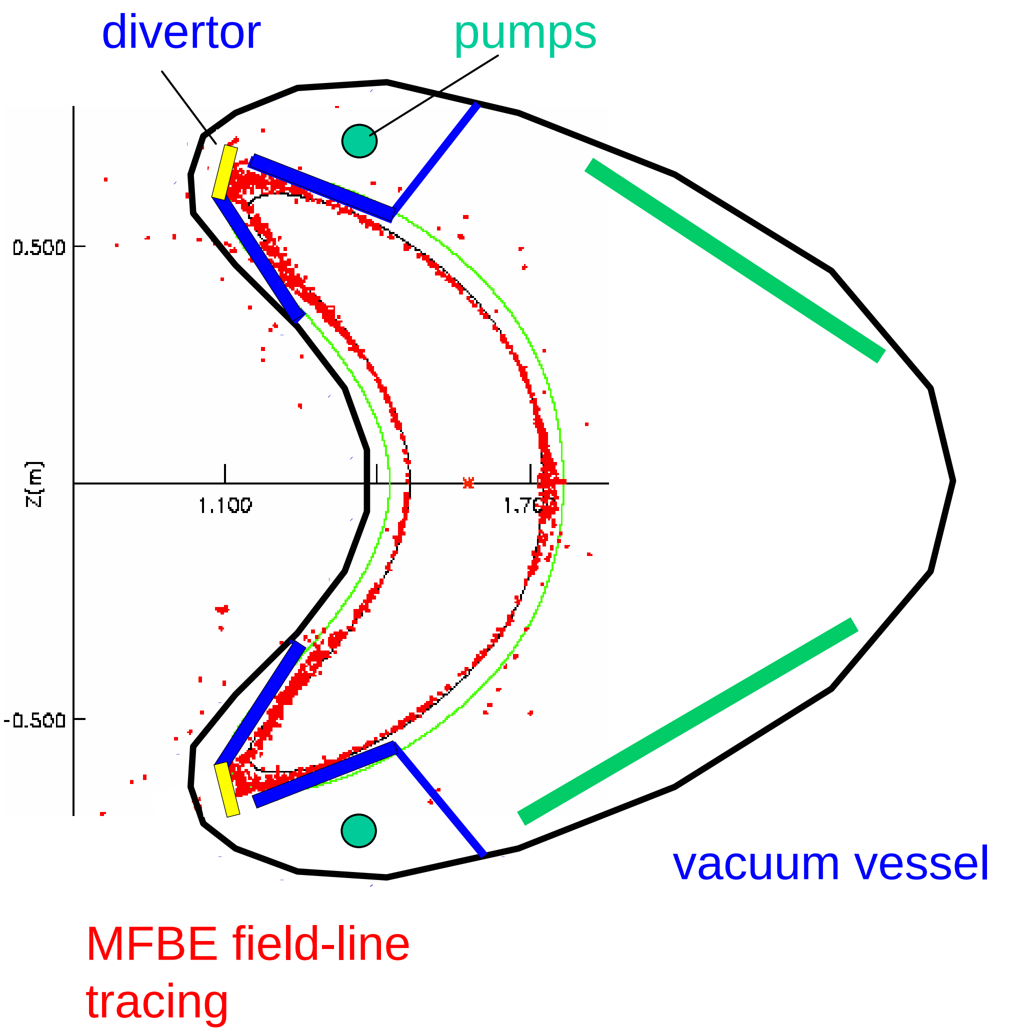
Candidate trim-coil arrays of saddle coils, mounted outside modular coil shell being analyzed.

Control strategy to be developed..

Divertors in Bean-tips

- Strong flux-expansion always observed in bean-shaped cross-section. Allows isolation of PFC interaction.

- Similar to expanded boundary shaped-tokamak configurations



Quasi-Symmetry: Orbit Confinement in 3D

3D shape of standard stellarators no conserved canonical momenta

orbits can have resonant perturbations, become stochastic lost

B is bumpy every direction rotation is strongly damped

'Quasi-symmetry'

(Boozer,1983) Orbits & collisional transport depends on variation of |B| within flux surface, not the vector components of B !

(Nührenberg) If |B| is symmetric in flux coordinates, get confined orbits like tokamak

neoclassical transport very similar to tokamaks,

undamped rotation

- Recently tested in HSX, a small experiment at Univ. Wisconsin
  - quasi-Helical-Symmetry reduces electron transport, flow damping
  - Too small to study high pressure or ion transport