Stellarator Options For MFE Development

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1st Workshop on MFE Development Strategy in China

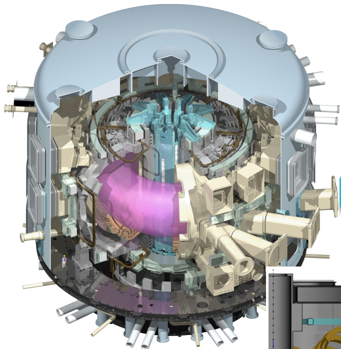
5 January 2012

Outline

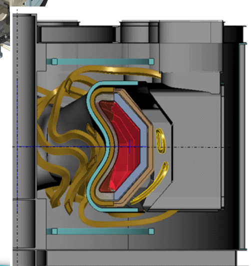
- Introduction and motivation
- Goals and requirements for Fusion Energy Pilot Plant
- Stellarators: opportunity to simplify fusion physics
- Stellarator pilot plants
- Conclusions

Charting the Roadmap to Fusion Energy: Options for a Next Step

ITER

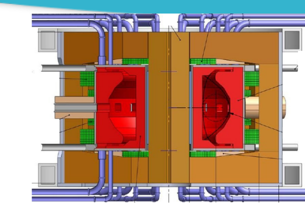


Demo

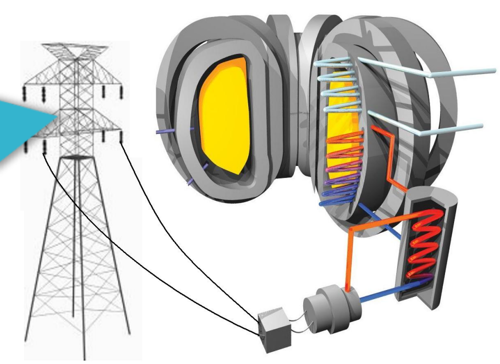


Pilot plant

Fusion Nuclear Facility



Component test facility (CTF)



Supporting Physics and Technology

- High performance, steady state
- Materials R&D
- Plasma Material Interface

How best to address R&D needs & risks before DEMO?
Evaluating: AT, ST, Stellarators

What must we learn beyond ITER?

ITER: 500 MW for 400s, gain > 10 pulsed
gain > 5 steady

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

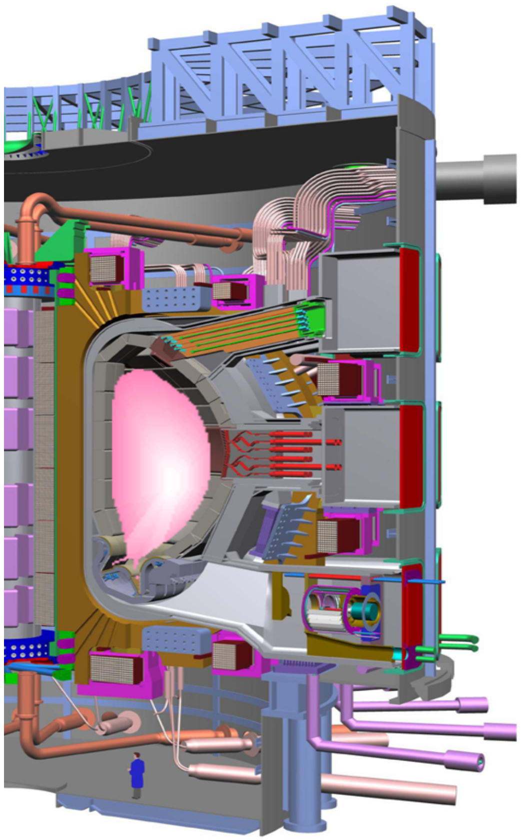
This requires plasmas with:

- Higher β, by at least factor of ~2.2
- Less external current drive, more efficient CD
  No inductive current
- Essentially no disruptions or ELMs

- Stable confinement of α-particles
- Low-temperature plasma exhaust

- And: technological advances for T-breeding & burning environment

Issues: How to sustain plasma confinement?
How to keep it stable & disruption free?



ITER (~ 2021/2027)

M

Pilot plant goals, requirements

Integrate key science and technology capabilities of a fusion power plant in a reduced-scale R&D facility.

- Target needed capabilities:
  - Net electricity production
    - Efficient operation, low recirculating power
  - Fusion nuclear component testing
    - Steady-state operation
    - Tritium self-sufficiency
    - Neutron wall loading ≥ $1 MW/m^2$
  - Maintenance scheme applicable to power plant
    - Demonstrate methods for fast replacement of in-vessel components
    - Eliminate disruptions

Key pilot metric is overall electrical efficiency: Q_{eng}

$$Q_{eng} = \frac{\textit{Electricity produced}}{\textit{Electricity consumed}} = \frac{\eta_{th}(M_n P_n + P_\alpha + P_{aux} + P_{pump})}{\frac{P_{aux}}{\eta_{aux}} + P_{pump} + P_{sub} + P_{coils} + P_{control}}$$

$$Q_{eng} = \frac{\eta_{th}\eta_{aux}Q(4M_n + 1 + 5/Q + 5P_{pump}/P_{fus})}{5(1+\eta_{aux}QP_{extra}/P_{fus})}$$

η_{th}	=	**thermal conversion efficiency**
η_{aux}	=	**injected power wall plug efficiency**
Q	=	**fusion power / auxiliary power**
M_n	=	neutron energy multiplier
P_n	=	neutron power from fusion
P_α	=	alpha power from fusion
P_{aux}	=	injected power (heat + CD + control)
P_{pump}	=	coolant pumping power
P_{sub}	=	subsystems power
P_{coils}	=	power lost in coils (Cu)
$P_{control}$	=	power used in plasma or plant control that is not included in P_{inj}
P_{extra}	=	$P_{pump} + P_{sub} + P_{coils} + P_{control}$

Blanket and auxiliary heating and current-drive efficiency + fusion gain largely determine electrical efficiency Q_{eng}

Pumping, sub-systems power assumed to be proportional to $P_{thermal}$ – needs further research

Wall Plug Efficiency & Current Drive Efficiency Are Critical

D.Stork 2009

- DEMO assumptions:
  $\eta_{WP} \bullet \gamma_{CD} = 0.24 - 0.27$
- Negative NBI $\eta_{WP} \bullet \gamma_{CD} \sim 0.12 - 0.14$
- ECCD $\eta_{WP} \bullet \gamma_{CD} \sim 0.08$
- ICRF $\eta_{WP} \bullet \gamma_{CD} \sim [0.18 - 0.24] \bullet f_{coupled}$
  (where $f_{coupled}$ = fraction of generator power coupled at edge of plasma ~ 0.4 max H-mode – note no experiment has ever coupled >12MW ICRF power into an H-mode) ~0.07 – 0.095 for H-mode
- Lower Hybrid CD $\eta_{WP} \bullet \gamma_{CD} \sim [0.15 - 0.18] \bullet f_{coupled}$
  (LH klystrons are ~ 50% efficient – again $f_{coupled}$ is fraction of generator power coupled by grill to plasma – note, no experiment has ever coupled more than 4MW LH power into an H-mode)

- $\eta_{WP} = P_{CD}/P_{electrical}$
- **Most DEMO studies very optimistic. Realistic values imply much higher recirculating power**
- **NBI, ICRF, LH require large launchers; impact tritium breeding**

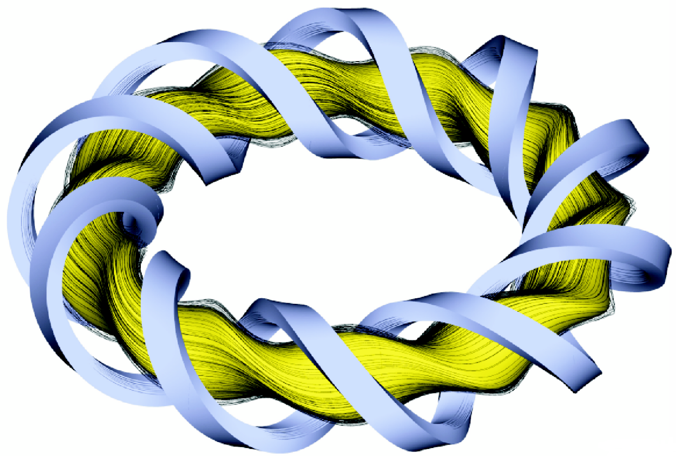
Stellarators:
Equilibrium from Helical Shaping

- Stellarators
  - 3D shape provides poloidal field
  - No driven current in plasma

    Very low recirculating power
  - *No disruptions*: equilibrium

    maintained with or without plasma
  - *Simple steady state*:

    LHD ~ 1 hour pulses



Large Helical Device (Japan)

A = 6-7, R=3.9 m, B=3T

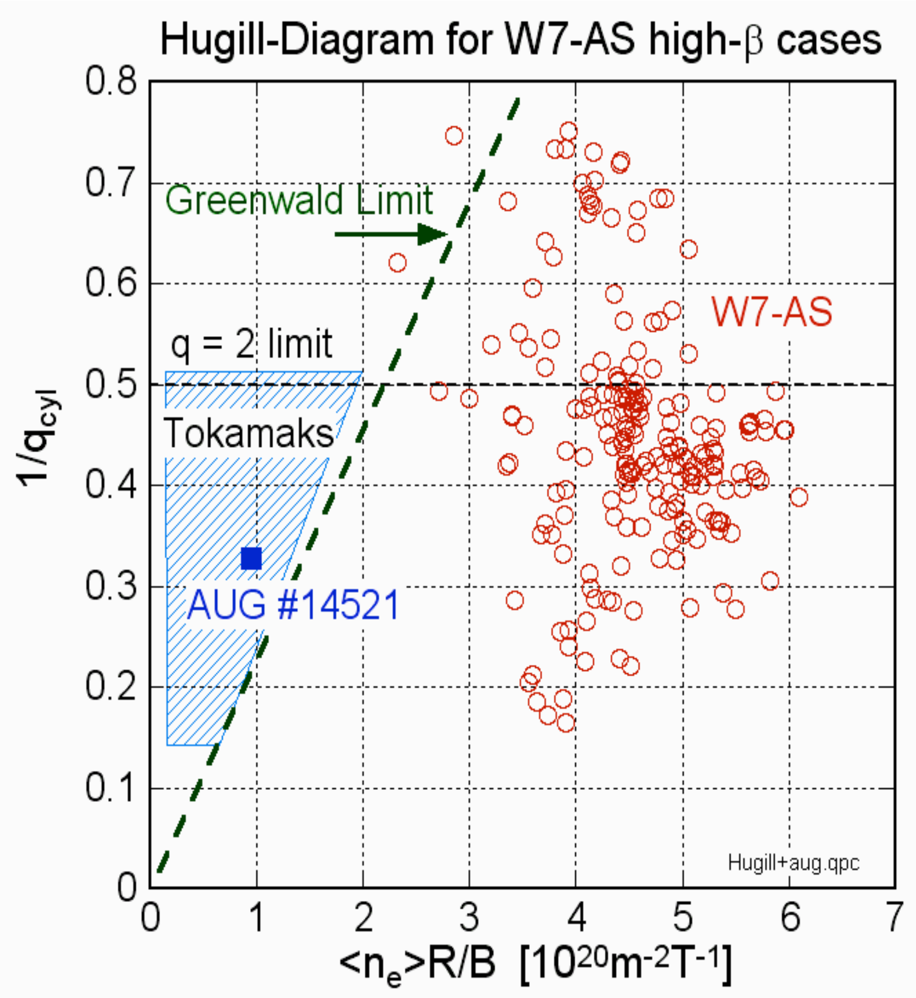
Superconducting

- ~ 40 shaping parameters controllable with 3D

  only ~ 4 shaping parameters available if axisymmetric

  ⇒ more ability to control plasma thru magnetic shape.

Stellarator Operating Range is much larger than for Tokamaks

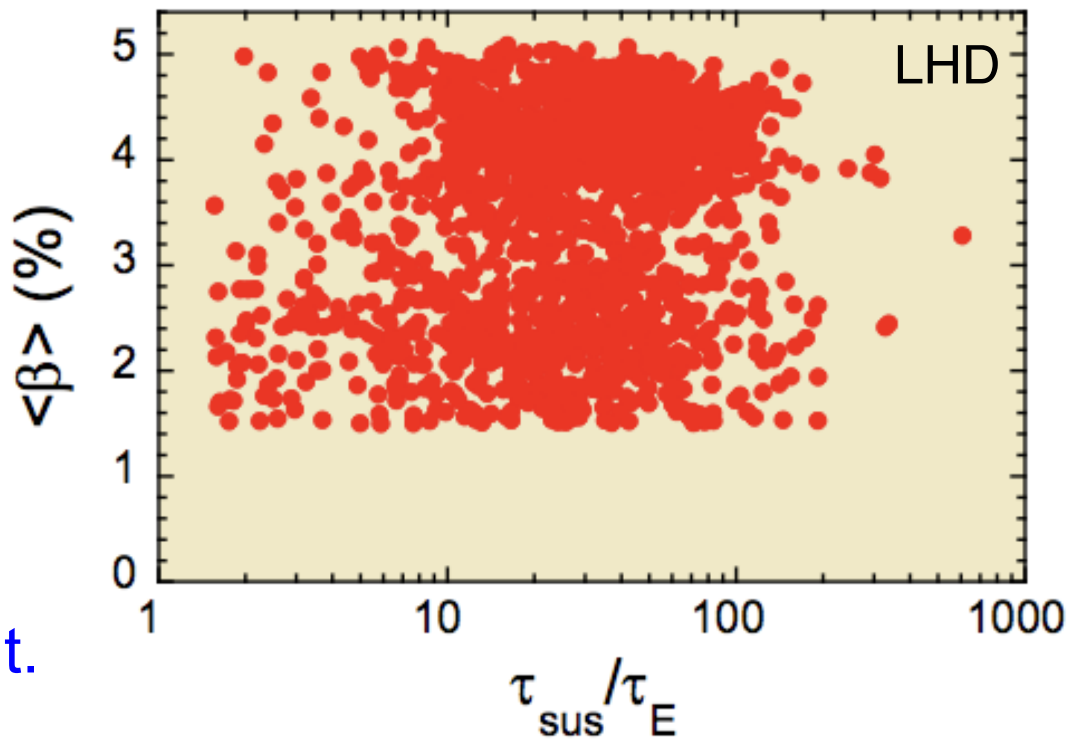


- Density limit ~5 X equivalent Greenwald density limit (from tokamaks).
- LHD n_{e0} = 1.2 x 10^{21} m^{-3} at B = 2.5 T
  p(0) = 1.5 atm
- Can operate with q>2, even q>1
- No disruptions.
  Limits are not due to MHD instabilities.
- High density favorable:
  - Lower plasma edge temperature,
    Eases edge design
  - Reduces energetic particle instability drive

High β Steady State, without Disruptions

- β =5.4% (LHD)
  and β=3.4% (W 7-AS)
  without *any* disruptions.
  Quiescent steady-state.

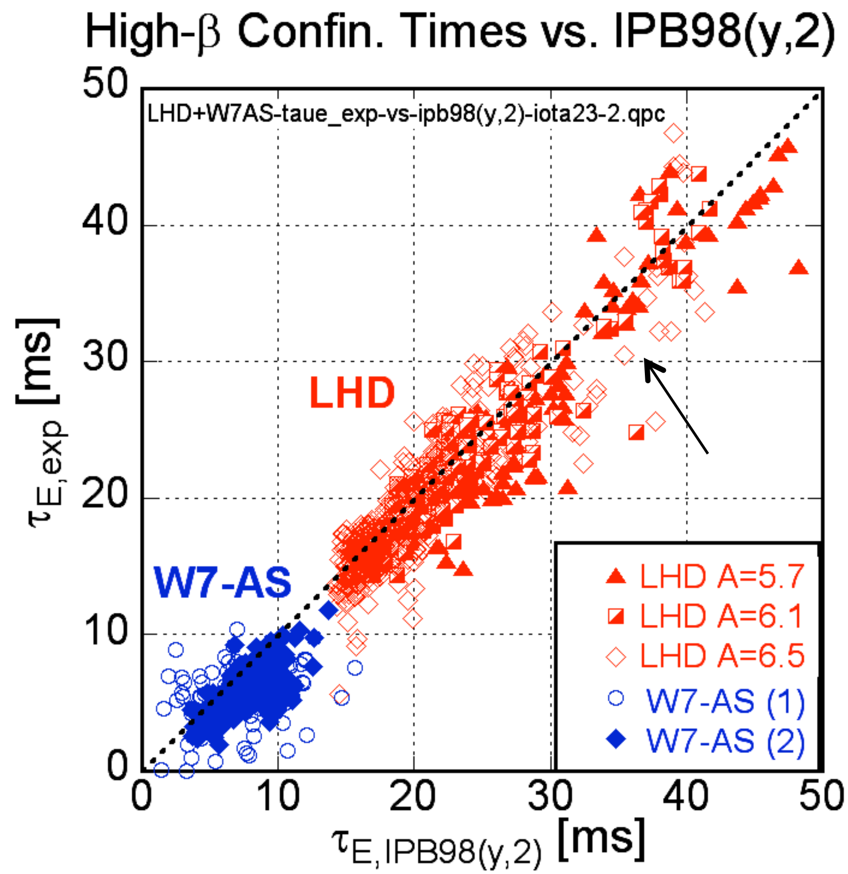
- Soft limit is observed, due
  to saturation in confinement.



- Highest β ~ twice ideal stability threshold. In W7AS: no MHD activity.   In LHD: saturated MHD observed.
- β-limit appears to be due to equilibrium limits (flux surface breakup).   Can be improved by design (W7X).

Stellarator Energy Confinement Similar to Tokamaks

- Stellarator τ_E data similar to tokamak ELMy H-mode
- ISS-04 confinement scaling derived from Stellarator L-mode data base. Gyro-Bohm like.
- ITER Tokamak Confinement DB in reasonable agreement with ISS-04 scaling
- T_i = 7 keV without impurity accumulation (LHD)



Stellarators Provide Solutions To Fusion Challenges

Steady-state toroidal plasmas with

- ✓ No disruptions. Equilibrium maintained by external coils
- ✓ Quiescent steady state at high-beta with confinement similar to tokamaks.
- ✓ Not limited by macroscopic instabilities. No need to control profiles. No need for feedback or rotation to control instabilities. Greatly simplify plasma control and related diagnostics
- ✓ Very high density limit ⇒ easier plasma solutions for divertor
  reduced fast-ion instability drive
- ✓ No current drive ⇒ intrinsically high Q, higher reliability

Greatly simplifies many aspects of Pilot Plant and DEMO designs.

Need to achieve these properties simultaneously, compatibe with high-power divertor.

Hybrids: 3D Ohmic Tokamaks (1970s)

Hybrids with encircling helical windings

CLEO (UK), JIPP-IB (Japan),

L-2 (USSR), W7-A (Germany)

+ No disruptions if iota-coils > 0.14 (q-vac < 7)

+ very low-q(a) operation possible:

L-2: q(a)=1, CLEO: q(a)=1.4, W7-A: q(a)=2

Recent Hybrid Experiment

CTH (Auburn Univ.)

+ No disruptions if iota-coils > 0.1 (q-vac < 10)

How to get good orbit confinement??

3D Configurations: Need to Optimize for Good Confinement

3D: No symmetry $\Rightarrow$ no conserved canonical momenta $\Rightarrow$ lost orbits

$\Rightarrow$ rotation is strongly damped

- 3D transport due to magnetic ripple
  - Collisionless ion thermal transport $\chi_i \propto \varepsilon_{eff}^{3/2} T_i^{7/2}$ (excl. E_r effects)

- ‘Quasi-symmetry’
  - (Boozer,1983) Orbits & neoclassical transport depend on variation of |B| within flux surface, not the vector components of B !
  - If |B| is symmetric in flux coordinates, get confined orbits like tokamak
  - Can be perfected on one surface in toridal system; degrades mildly

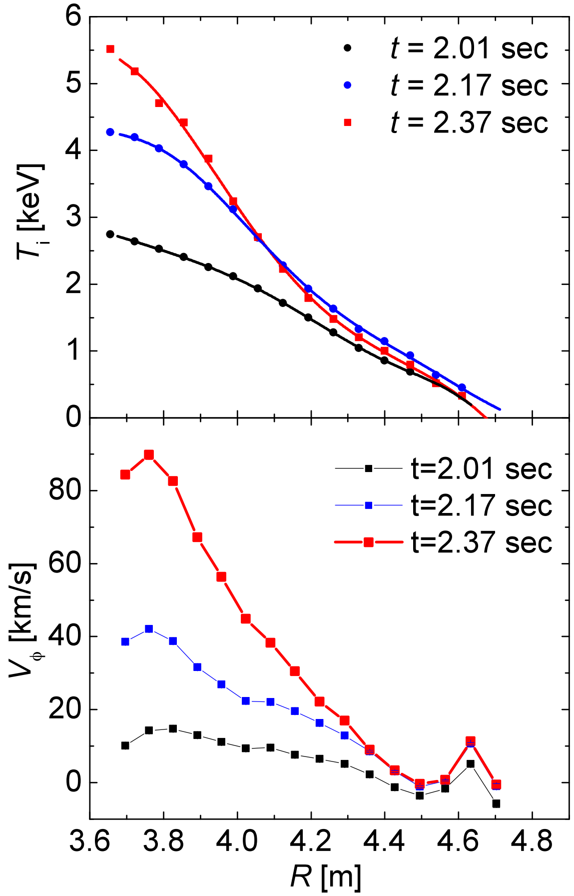
  $\Rightarrow$ Neoclassical transport very similar to tokamaks (theoretically), undamped rotation

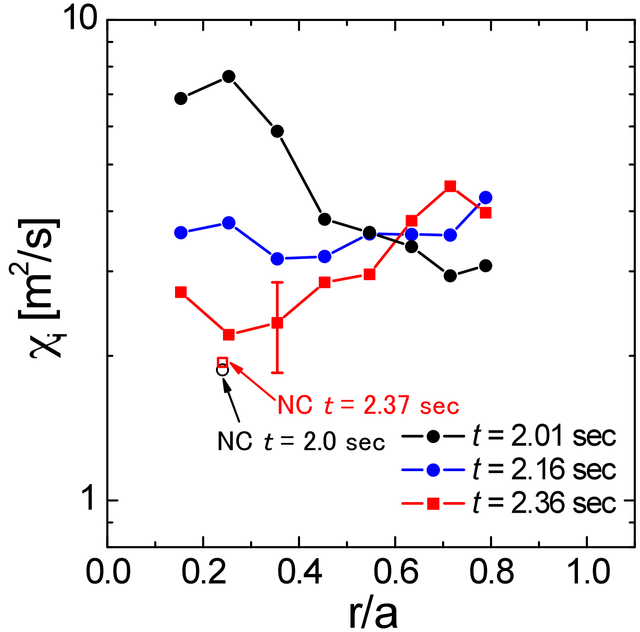
Confinement Optimization Approaches

Different types of quasi-symmetry

- Quasi-axisymmetry
  - Tokamak like confinement properties
  - Optimized designs down to A~2.5
  - Turbulence predicted to be similar to reversed shear in tokamaks
- Quasi-helical symmetry
  - Neoclassical transport ~1/3 of tokamaks
  - Optimized designs down to A~8
  - Predicted turbulence >> tokamaks in evaluations so far
- Quasi-isodynamic / Quasi-poloidal symmetry
  - Neoclassical transport ~ eliminated!
  - Optimized designs down to A~8
  - Predicted turbulence lower than comparable tokamak

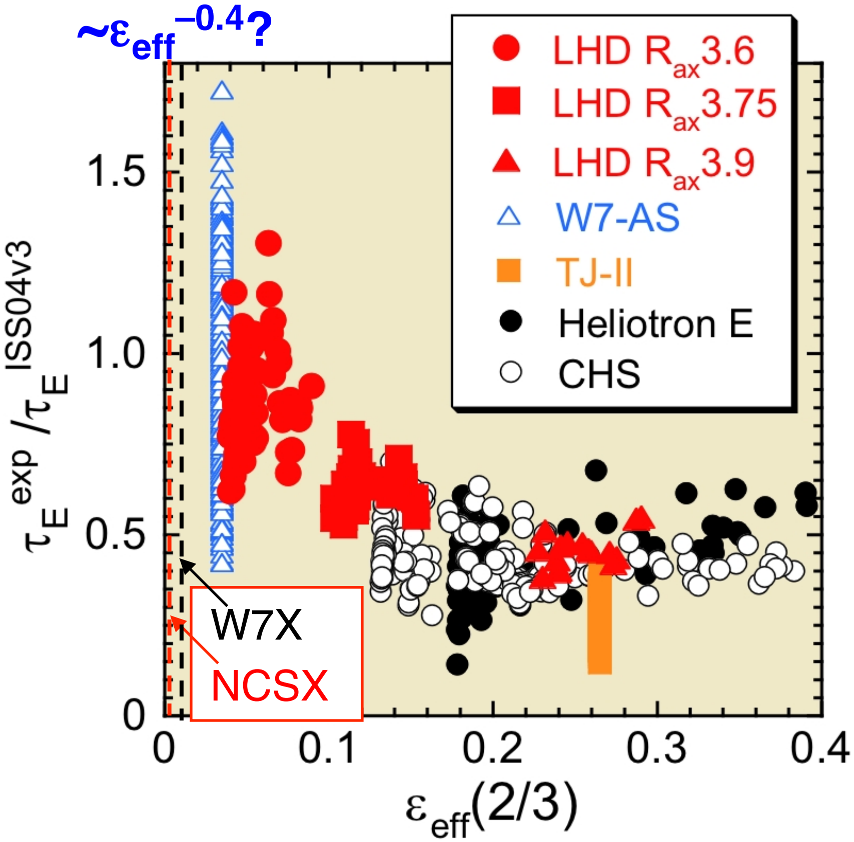
T_i = 7keV in LHD: χ_i at 3D Neoclassical Levels

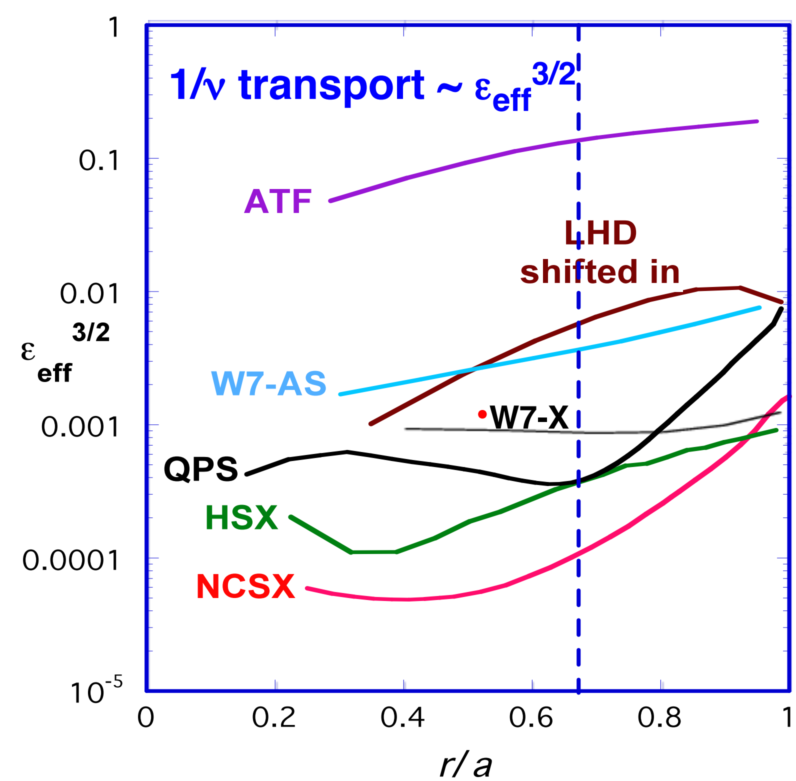




- **Peaked profiles of V_ϕ as well as T_i. ITB forms**
- **The ion thermal diffusivity (χ_i) decreases** and reaches **neoclassical level** -> **reduction of anomalous transport**
- Core χ_i-neo ~ 2 m^2/sec due to ripple.

Low Ripple Magnetic Field Important for Good Confinement

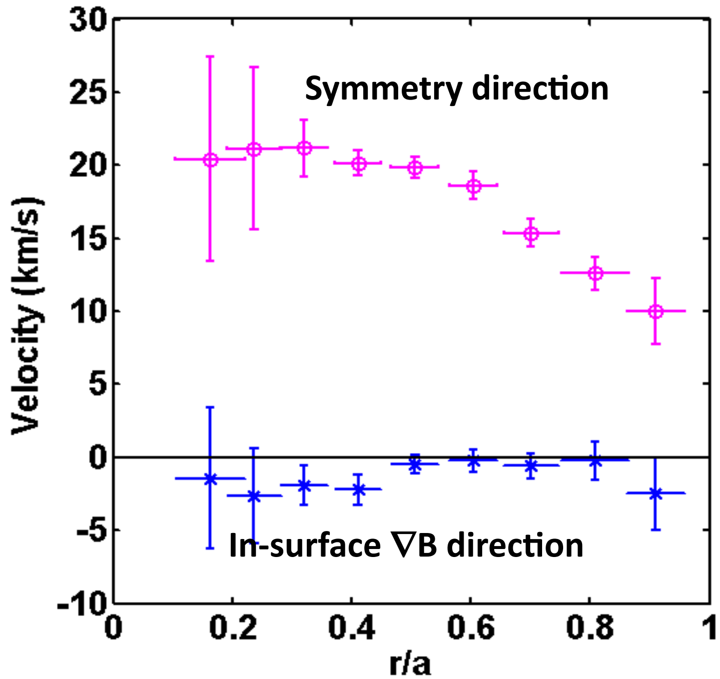




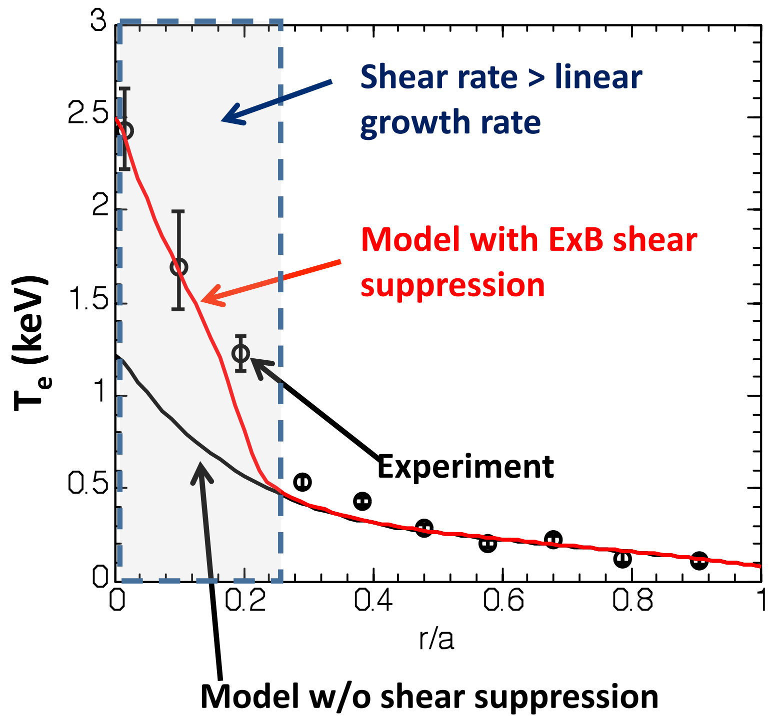
- Global confinement scaling for stellarators (ISS04v3) found strong dependence on ripple magnitude.
- Gyrokinetics: turbulence reduced at low ripple.
- H (ISS04) up to 1.5 obtained at low ripple
- All new configurations designed for low ripple (HSX, W7X, NCSX)

Quasi-helical Symmetry in HSX has Similarities to 2D Transport Physics

- HSX demonstrated quasi-symmetry reduces transport of momentum, heat, and particles, compared to conventional stellarator.
- Present focus is on exploring neoclassical and anomalous transport in ECRH plasma & 3D equilibrium reconstruction.

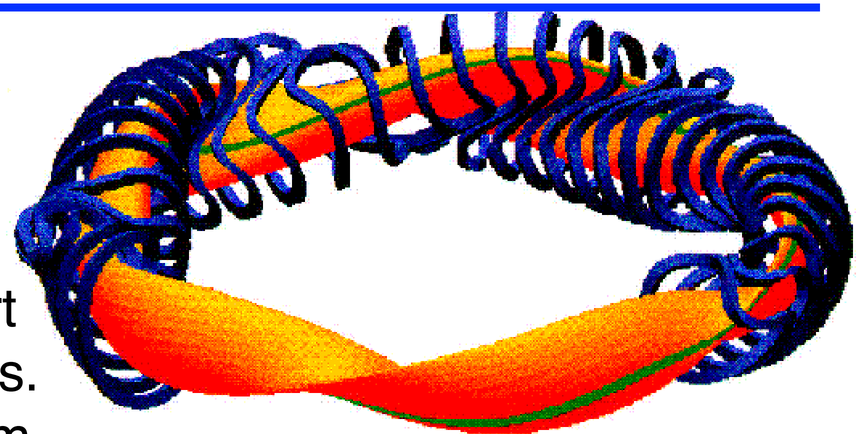


Large (~20 km/s) flows in direction of symmetry measured by CHERS.



Internal transport barrier due to neoclassical E_r shear.

W 7-X Optimized for High-β, Quasi-Isodynamic



- 5 periods, R/⟨a⟩=11, R=5.4 m
  Superconducting coils
- Quasi-isodynamic: neoclassical transport minimized by minimizing drift-orbit widths. An approximation to quasi-poloidal symm. Theory projections H_{ISS04} : 2-3
- Bootstrap current & Pfirsch-Schluter current minimized to minimize change in equilibrium with increasing β. This also implies strong rotation damping (including zonal flows)
- MHD Stable for β = 5%
- Designed for good vacuum flux surfaces. Current minimization keeps good surfaces to β=5%

NCSX: Optimized Design for High-β, Quasi-Axisymmetry

- 3 periods, R/$\langle a \rangle$=4.4, $\langle \kappa \rangle$~1.8 , $\langle \delta \rangle$~1

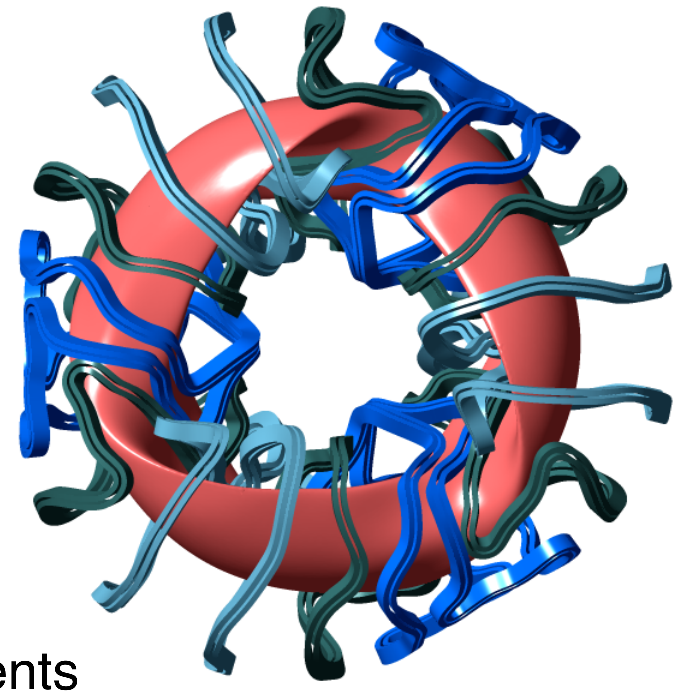
- Quasi-axisymmetric: tokamak with 3D shaping
  Ripple-induced thermal transport insignificant.
  Build on ITER results. Allows lower Aspect ratio.

- 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 $\beta > 6.5\%$ by adjusting coil currents

- Designed to keep ~perfect flux surfaces to β=4.1%
  2-fluid calculations indicate it may continue to $\beta > 7\%$

- Passive disruption stability: equilibrium maintained even with total loss of β or bootstrap current.



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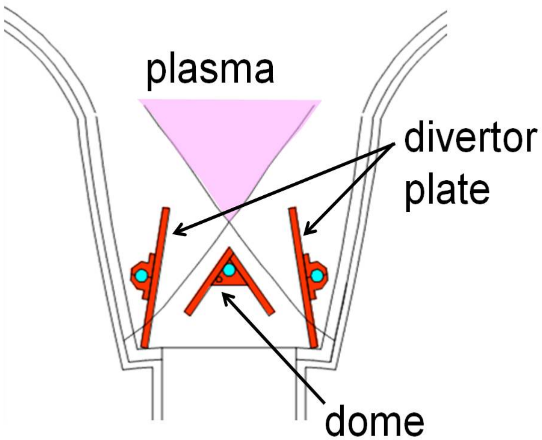
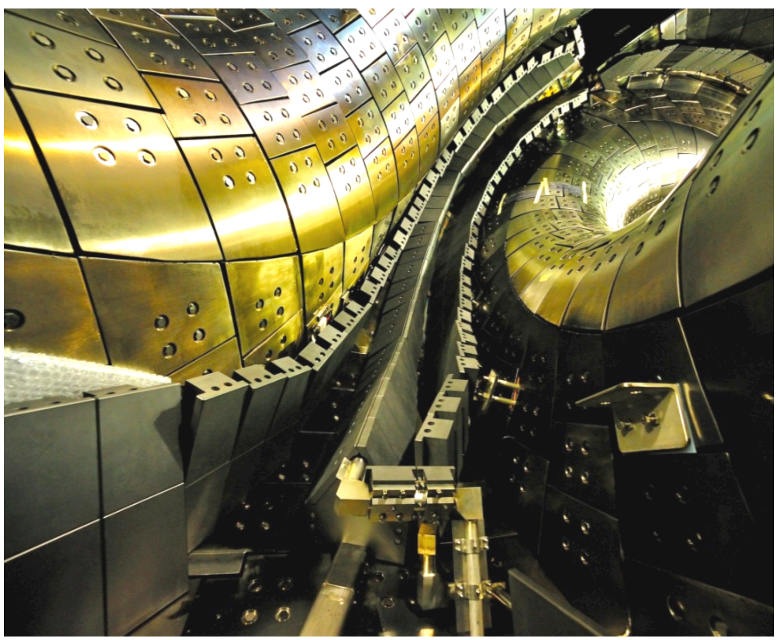
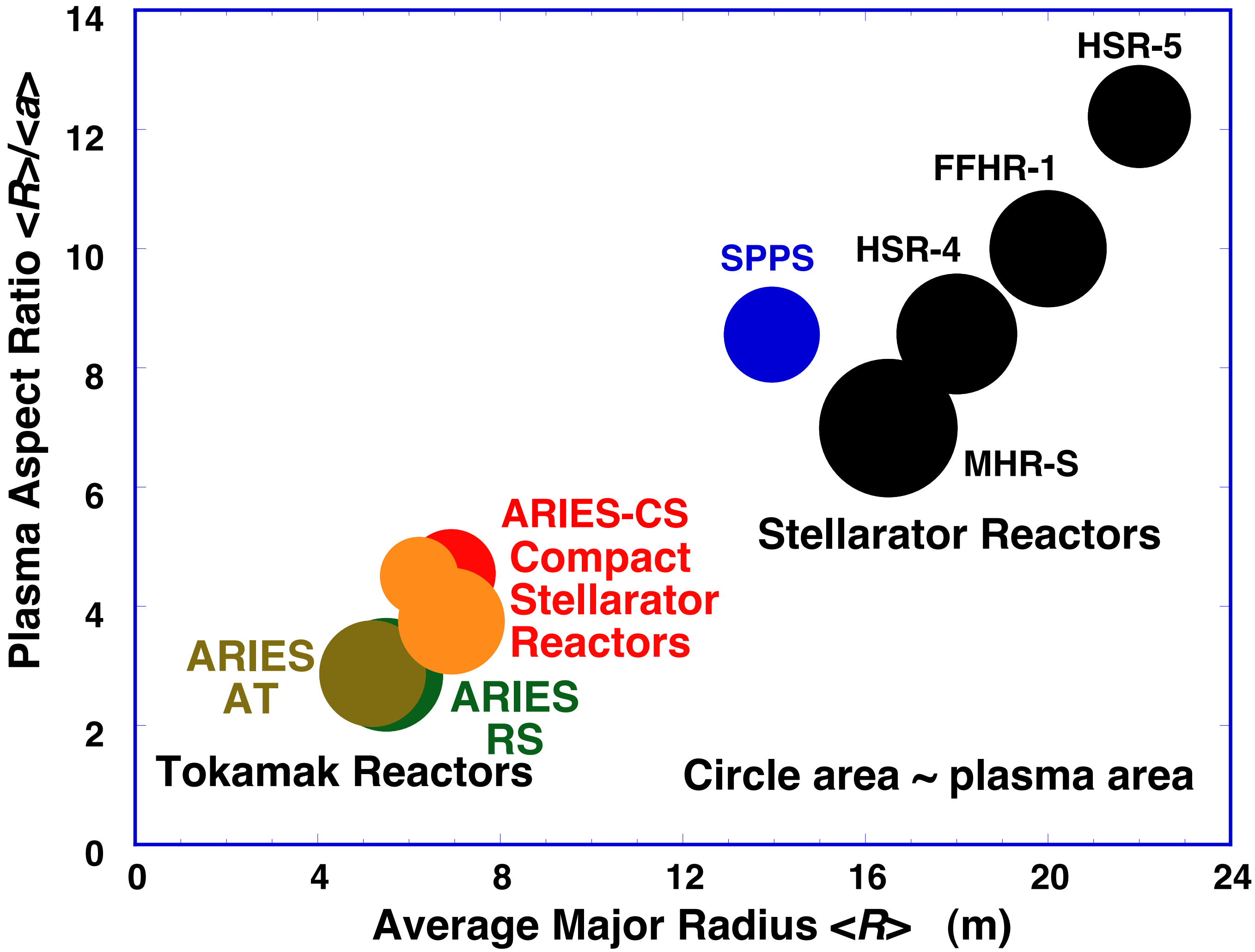
Stellarator Divertors Offer New Characteristics





- W7-AS and LHD have operated with closed helical divertors. W7-X will start up with divertor. Substantial modeling developed.
- Stellarator divertors have much longer connections length > 100m
  Easier to decouple from main plasma
- High density operation drops divertor temperature, reduces sputtering
- Ergodic divertor region expands footprint on divertor plates.

Stellarator Reactor Studies: QA comparable to Tokamak Designs

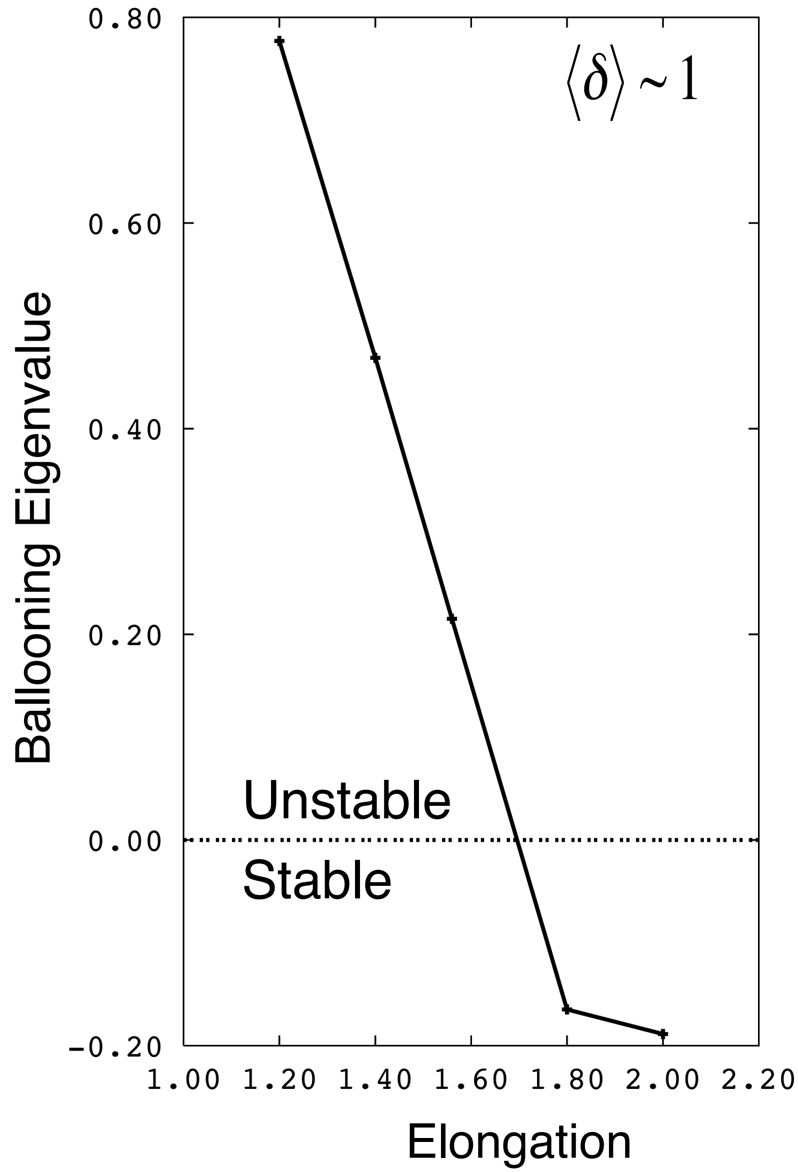


Focus on Quasi-axisymmetry (QA) for 3D tokamaks

- $|B| = |B|(\theta, \rho)$, like axisymmetry
- Can be added continuously to axisymmetry, keeping good orbits

- MHD stability characteristics similar for QA and tokamaks:
  - Elongation, at high triangularity, stabilizes ballooning
  - QA can access 2nd stablity (Hudson & Hegna)
  - NTMs stabilized by negative shear

<ins>Even without disruptions</ins>, this is important to prevent confinement degradation by MHD



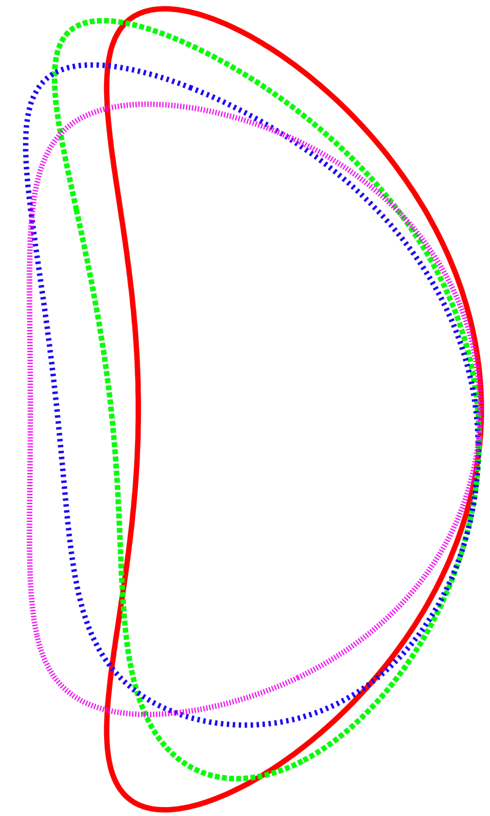
Method: Add QA shaping to Achieve Goals

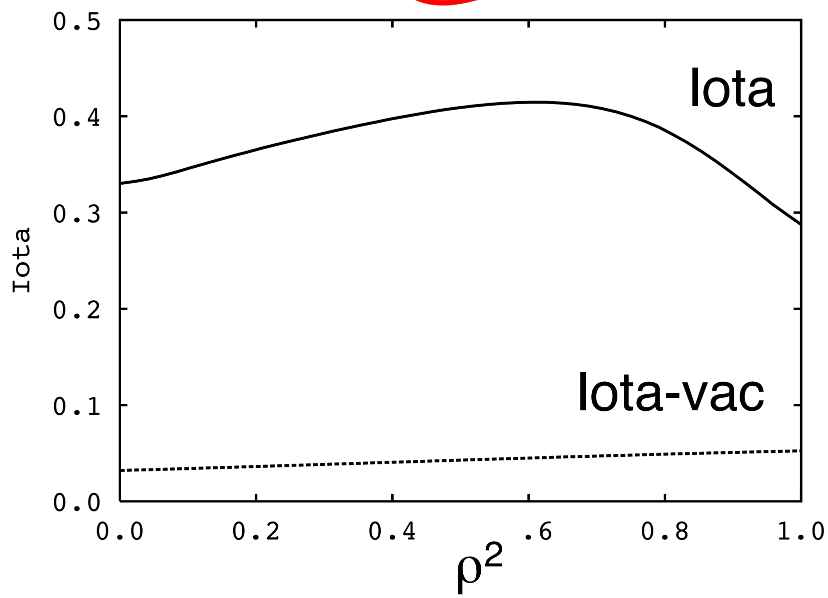
- Base case: Advanced Tokamak $\kappa = 1.8$, $\delta \sim 1$, $A = 4$, $\beta = 4\%$
- Add 3D QA shaping, producing vacuum rotational transform, iota-vac: 0.05 – 0.3. Use 3 field periods.
- Optimize 3D shaping to keep
  – residual ripple $\varepsilon_{eff} < 1\%$ in plasma core for good orbits
  – vacuum magnetic well of 2-3%, for general stability

- Vary shape, aspect ratio (and # periods)
- Examine impact on coil complexity

Iota-vac = 0.05 : Vertical stability

- Need externally driven current to keep good equilibrium
- No need for vertical position control.
- May eliminate VDEs
- If eliminate Ip, lose radial position. Probably will disrupt.

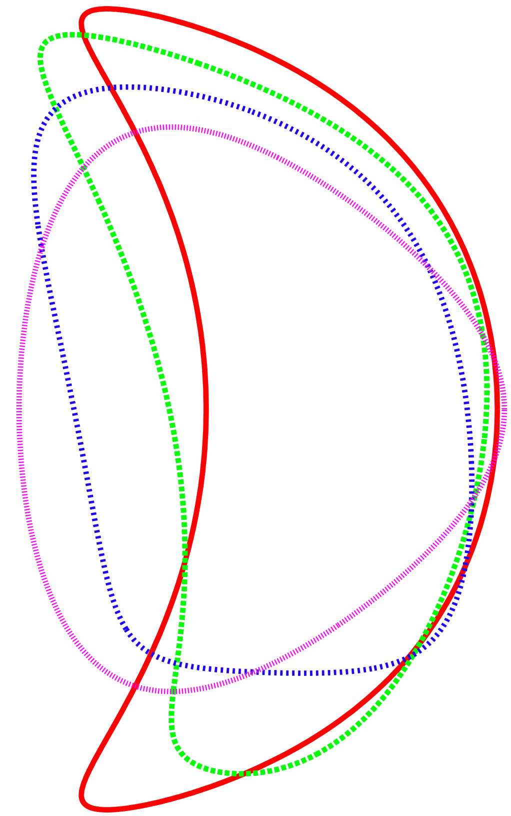


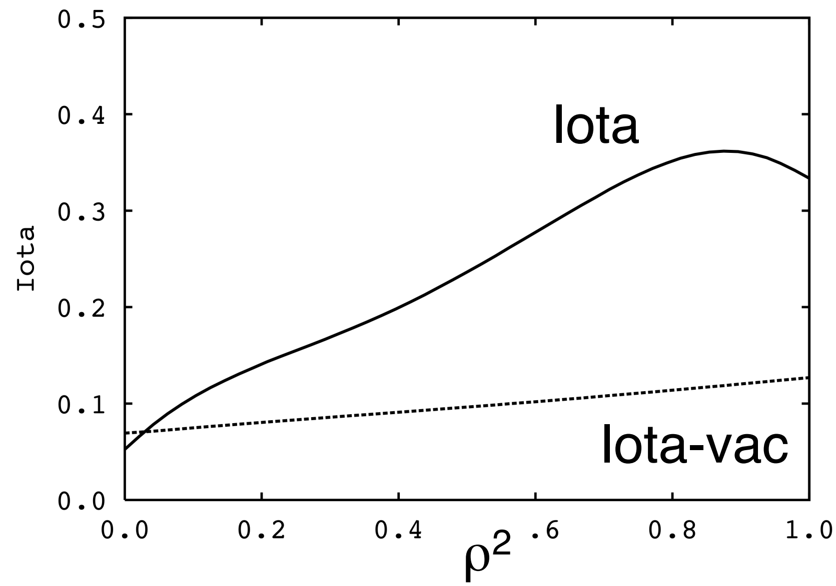


Vertical stability can also be achieved just with 3D shaping, with iota-vac=0 (Reiman 2007)

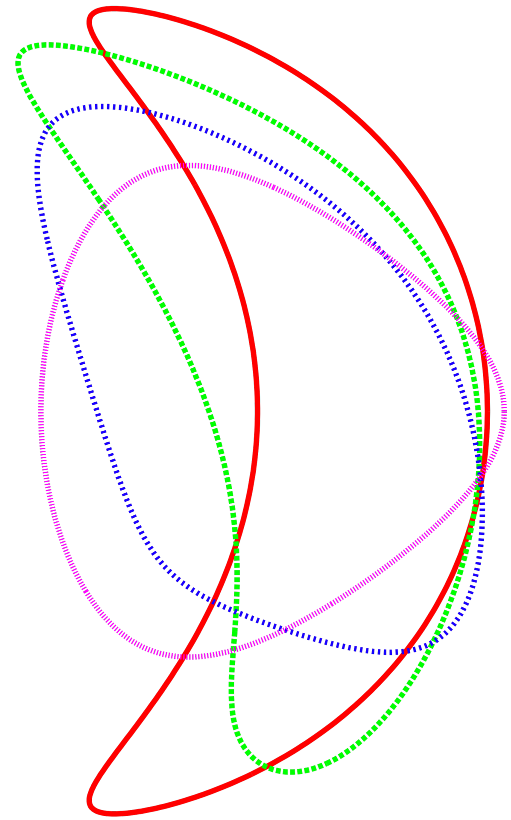
Iota-vac ~ 0.1 : Sustainment without CD

- Good equilibrium with bootstrap current alone. Shafranov shift ~ 35% of minor radius.
- Steady-state compatible with simple startup
- NTM stable
- Probably not disruption-proof





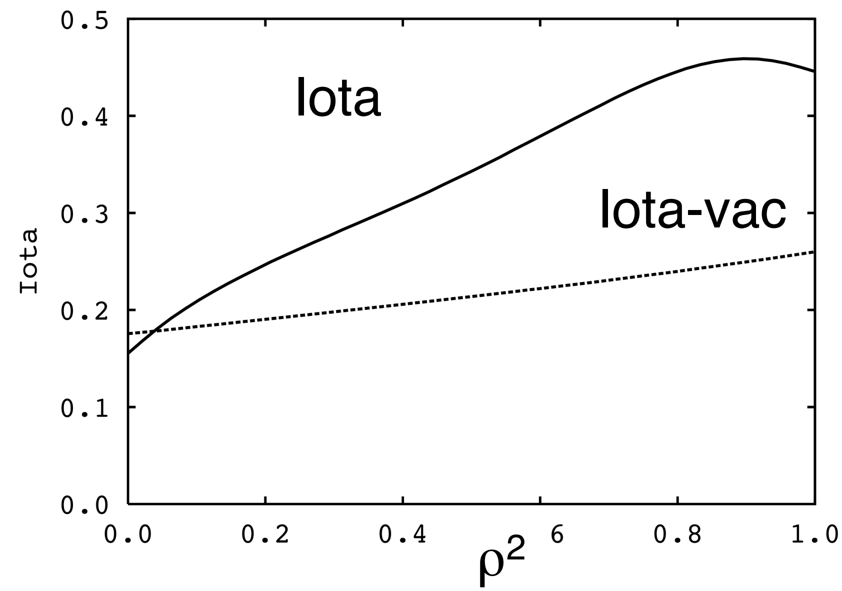
Iota-vac ~ 0.2 : Robust equilibrium



- Bootstrap current only. No external CD.
  Simple steady-state
  NTM stable

- Flux surfaces only displace slightly if abruptly loose I_P and β
  $\Rightarrow$ should be robust against disruption

- Iota-vac > 0.14 the empirical disruption stability criteria from W7-A



Iota-vac ~ 0.28 : External kink & RWM stabilized

- No need for external CD.

  Simple steady-state

  NTM stable

  No disruptions

- External kink / RWM stabilized

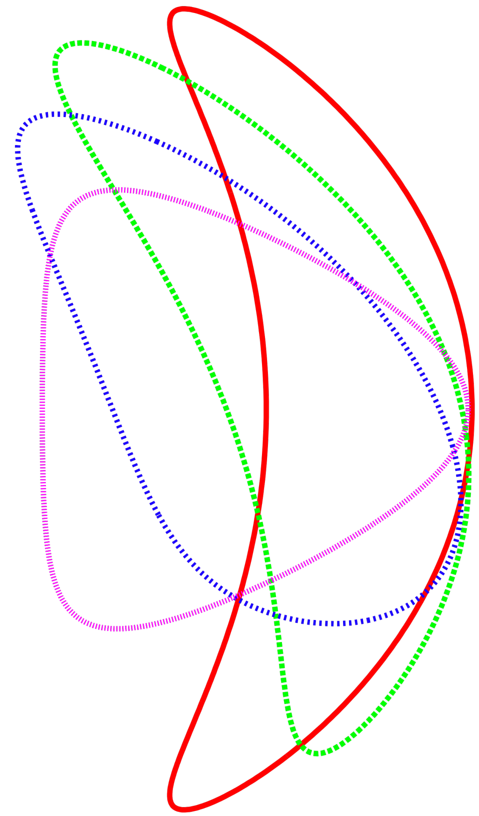
  shaping not feedback or

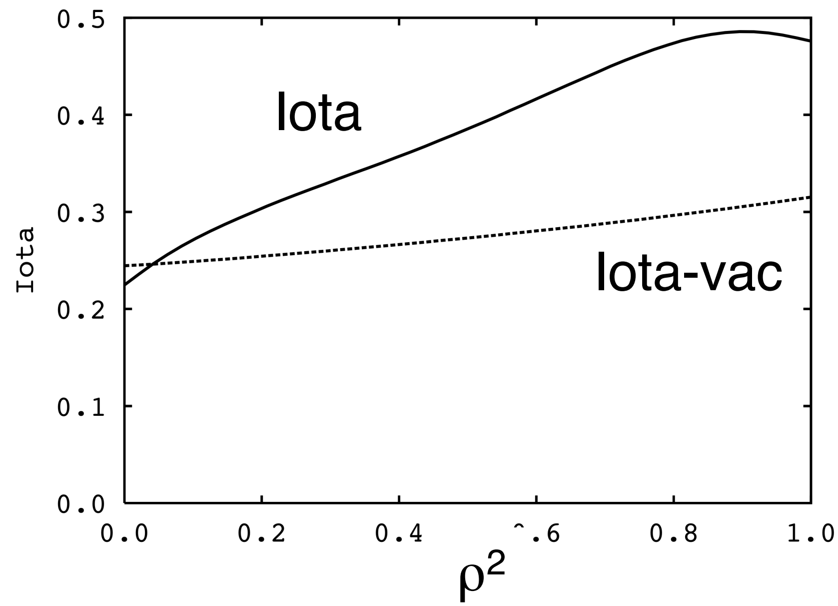
  nearby wall.

- Can further increase iota_vac

  to 0.6 $\Rightarrow$ only 10% of transform

  from bootstrap current





Stellarators Reduce Risks for Pilot Plants

- Plasma configuration sustained by coils
  - Don't require steady-state neutral beams and RF-launchers in burning environment
- Sustained, quiescent high-beta plasmas already demonstrated
- Robust confinement: no disruptions, can avoid edge instabilities (ELMs)
  - Allows thin first wall for breeding
  - No need for conducting wall in blanket
  - ⇒ Increase TBR & reduce wall complexity

Stellarators Reduce Risks (2)

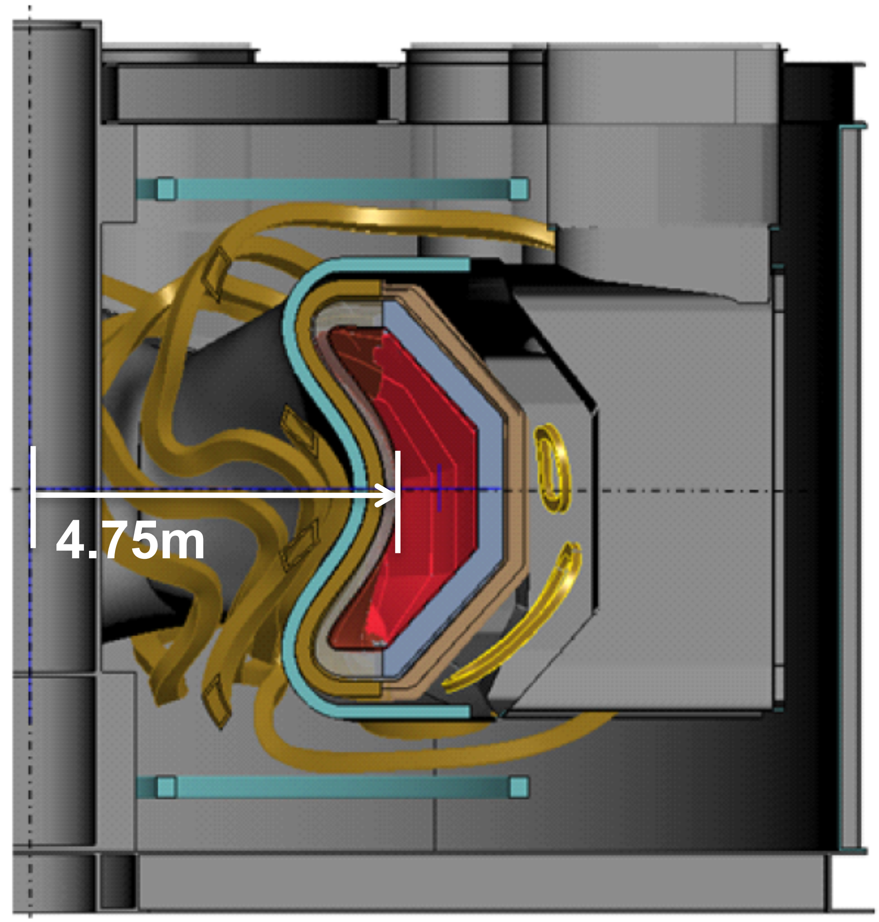
- Don't need instability or profile feedback control
  - Reduce need for diagnostics, feedback actuators in burning environment.
  - Higher reliability

But:

- Higher coil complexity.
- Small database of optimized experiments

Example Quasi-Axisymmetric Stellarator Pilot Plant

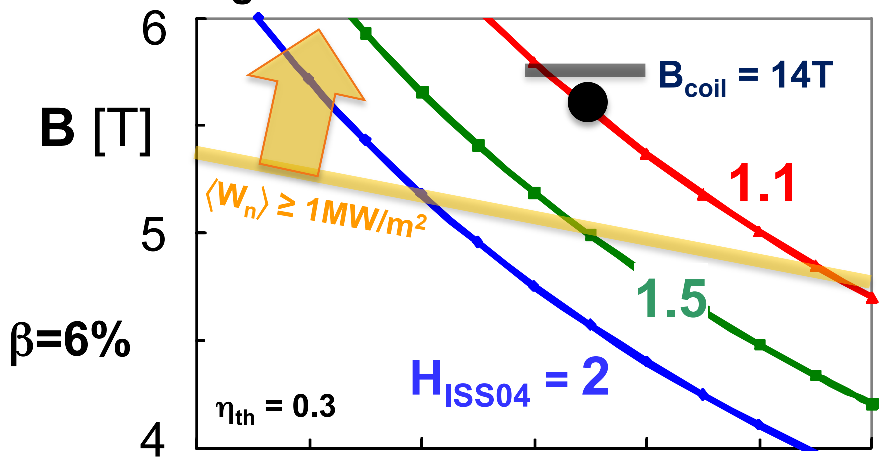
- $A = 4.5 = 4.75m / 1.05m$
- $B_T = 5.6T$, $I_P = 1.7MA$ (BS)
- Avg. $W_n = 1.2\text{-}2\ MW/m^2$
- Peak $W_n = 2.4\text{-}4\ MW/m^2$
- $Q_{eng} = 1.1$



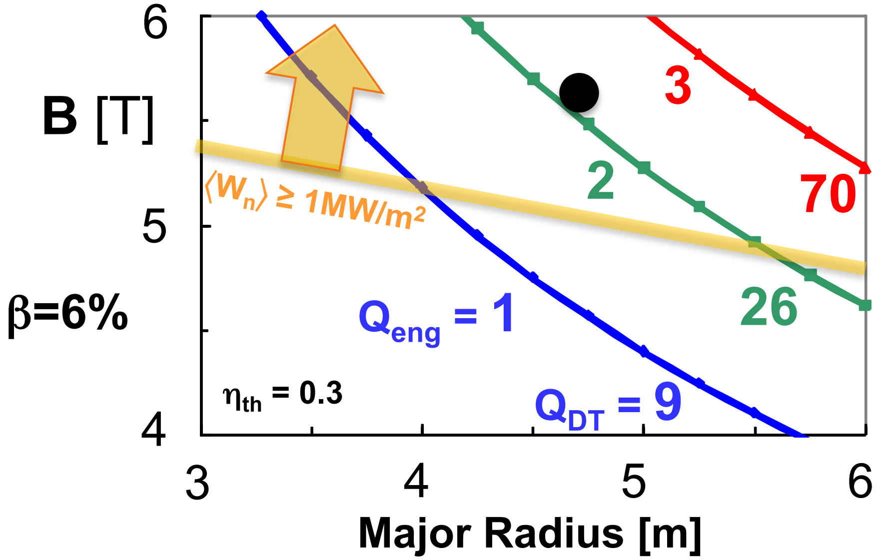
- Based on ARIES-CS design
- Divertor power flux < 10 MW/m^2
- Vertical maintenance between coils

Stellarator Pilot Plant can Operate $Q_{eng} > 1$ with L-mode Confinement.

Q_{eng}=1.1 contours vs H_{ISS04}



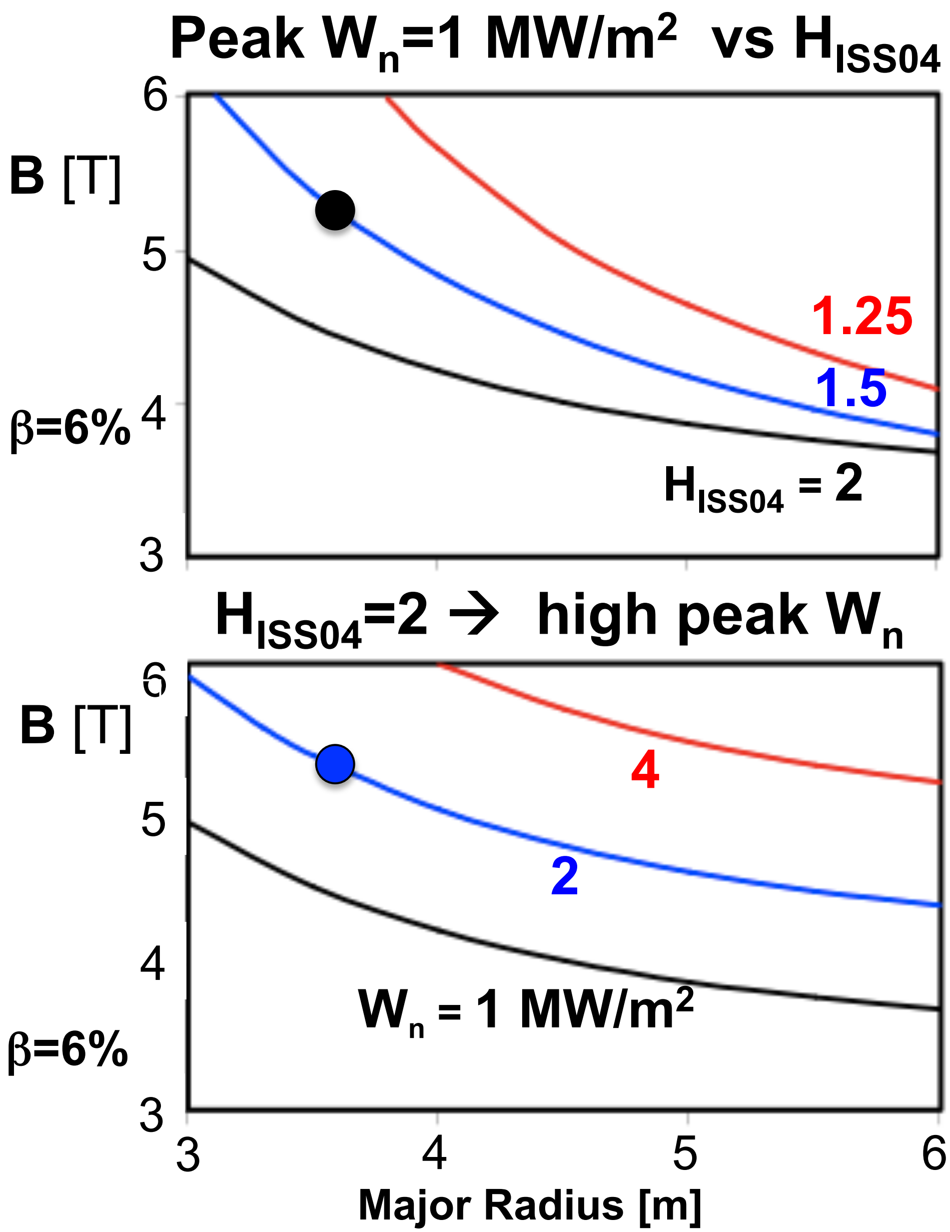
H_{ISS04}=2 → Q_{eng}=2-3, high Q_{DT}



● Pilot design point

- H_{ISS04} an L-mode scaling, Comparable to $H_{ITER97P}$
- $Q_{eng} > 1$ with $H_{ISS04} \sim 1$. Due to low recirculating power.
- Can operate $Q_{eng} > 1$ at low fusion power ~ 100 MW.
- Flux sufficient for blanket testing
- Expect higher H, gives higher Q_{eng}, provides margin & reliability.

Stellarator Pilot Plants Available at Reduced Size



- H_{ISS04} =1.5 attained on non-optimized stellarators
- Could allow CTF with R=3.5m, <a> = 0.77m, B_T=5.4T P_{fus} = 72MW
- H-mode confinement H=2 gives Peak W_n=2 MW/m^2, P_{fus}=144 MW
- Optimal size depends on blanket thickness and magnet technology.
- W7X will give data on low-ripple, optimized stellarator confinement.

But, not at low aspect ratio, nor QA
⇒ need experiment to validate calc.

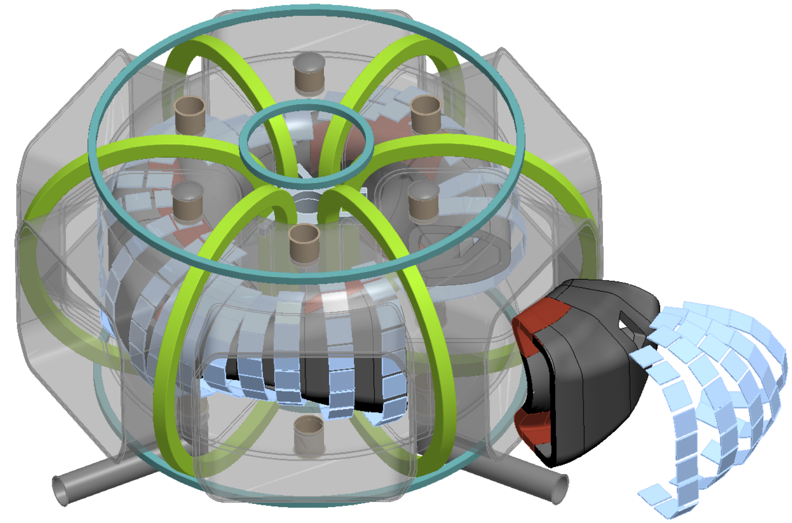
Simplify Coils and Maintenance

- Want sector-maintenance to increase availability and reliability.
- Any stellarator configuration can be made using range of coil configurations: helical; modular; TF+saddle coils; combinations.

New strategy for coils:

- Use passive magnetic materials or saddle coils to shape magnetic field.
- Mount on outside of blanket+shield module.

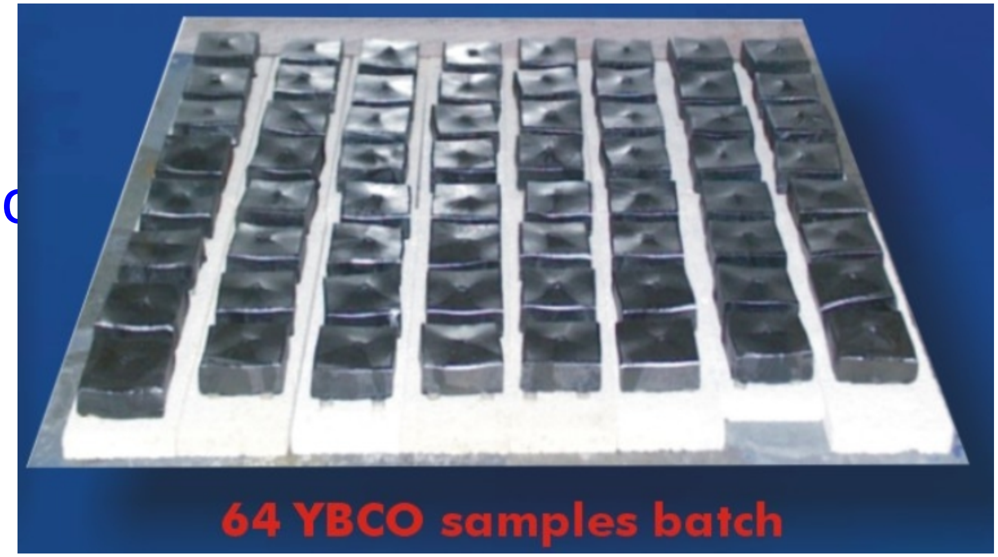
ARIES CS using HTS tiles for shaping.



L. Bromberg,
M. Zarnstorff, *et al.*
TOFE-19, Las Vegas, Nov. 2010

Passive 3D shaping: Diamagnetic Tiles

- Use bulk high temperature superconducting tiles as diamagnets to shape magnetic field
- Commercially available, up to 25 cm diameter
- Position and orient tiles to produce desired field shape, reacting to field from simple coils.
- Also can use other magnetic materials.



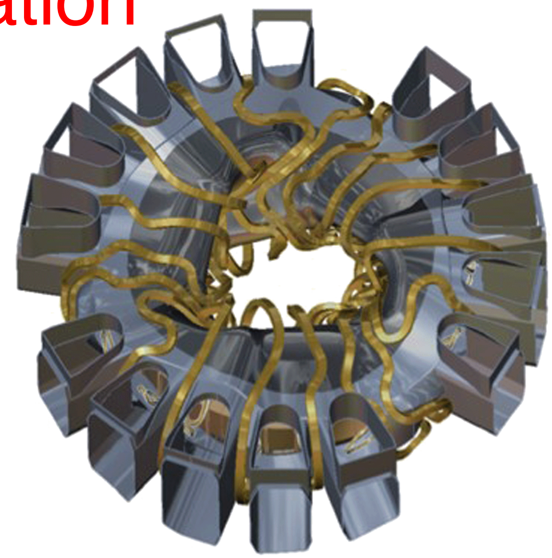
Engineering Improvements for High Availability And Simplification

Starting from ARIES-CS:

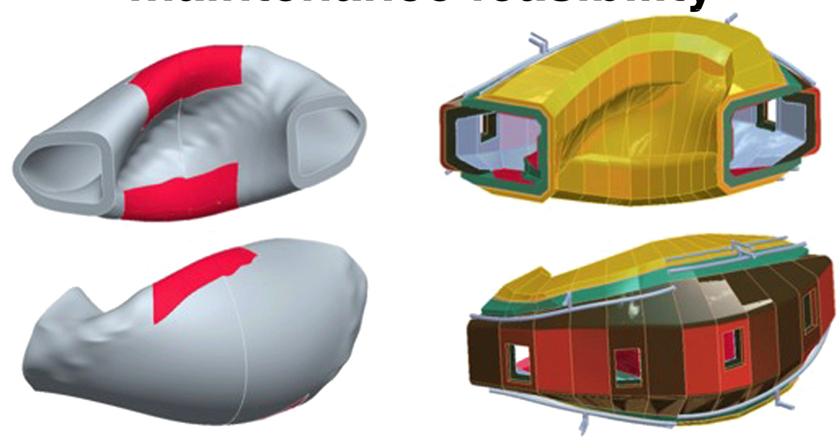
- Reduce the number of internal components (from ~200 modules in ARIES-CS to 30-70), by increasing size.
- Widen inter-coil openings on the outboard side; straighten the outboard legs.

  Maintenance between coils.

- Simplify the in-vessel blanket-shield geometry: 3D→2D shapes where possible.



Modified ARIES CS to improve maintenance feasibility



ARIES-CS blanket **HTS wf/bkt/shld**

Needed R&D for Stellarators

Sustained high-beta, robust confinement already achieved.

US Assessment (ReNeW & FESAC):

1. Simplify coil designs *(US design studies)*
   Simplify maintenance strategies for blanket
   Need to complete design and test materials.

2. Demonstrate integrated high performance: high-β, low collisionality *(W7X) (NCSX completion?)*

3. Confinement predictability *(LHD, W7X)*

4. Effective 3D divertor design *(LHD, W7X)*

Summary

- Stellarators can reduce the risks to advance magnetic fusion
  - Steady-state, disruption free, high beta demonstrated
  - No need for steady-state NBI, in-vessel RF launchers in burning environment
  - Reduce/eliminate need for feedback, diagnostics, actuators in burning environment
- Compact stellarator project to conservative Pilot Plants
  - Pilot plants, even with ~L-mode confinement; R=4.75 m
  - Pilot plants with P_{fusion} ~100 MW, or smaller size with higher confinement
- Concepts identified to simplify coils and provide sector maintenance. Need to complete engineering design & testing.

- Because needed physics characteristics are already demonstrated, stellarators offer a quicker route to a fusion pilot plant

Supplemental

Method: Vary QA shaping to Achieve Goals

- Base case: Advanced Tokamak $\kappa = 1.8$, $\delta \sim 1$, $A = 4$, $\beta = 4\%$
- Add 3D QA shaping, producing vacuum rotational transform, iota-vac: 0.05 – 0.3. Use 3 field periods.
- Optimize 3D shaping to keep
  - residual ripple $\varepsilon_{eff} < 1\%$ in plasma core for good orbits
  - vacuum magnetic well of 2-3%, for general stability
- Equilibrium: VMEC Optimization: STELLOPT

  Ideal low-n stability: TERPSICHORE, ballooning: COBRA

- Vary shape, aspect ratio (, # periods)
- Examine impact on coil complexity

ARIES-CS: a Competitive, Attractive Reactor

Reference parameters for baseline:

NCSX-like: 3 periods

$\langle R \rangle$ = 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

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

$\langle \beta \rangle$ = 6.4%

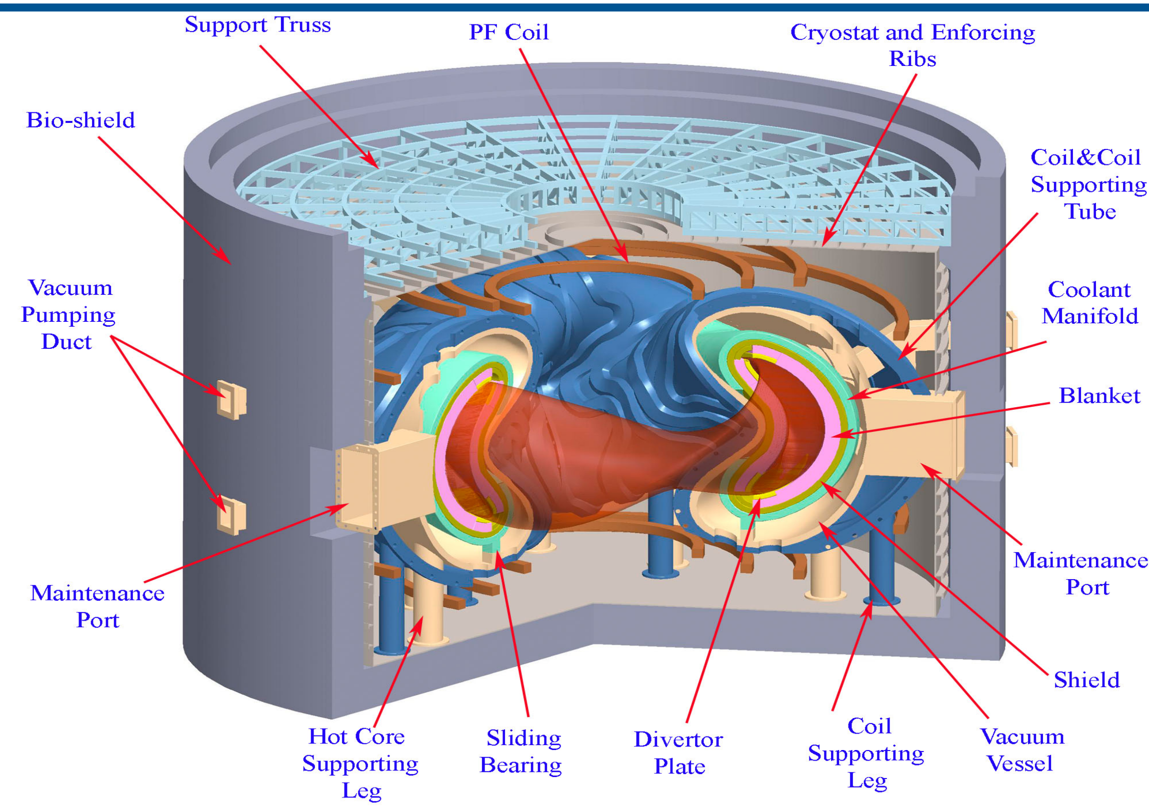
H(ISS04) = 1.1

I_{plasma} = 3.5 MA (bootstrap)

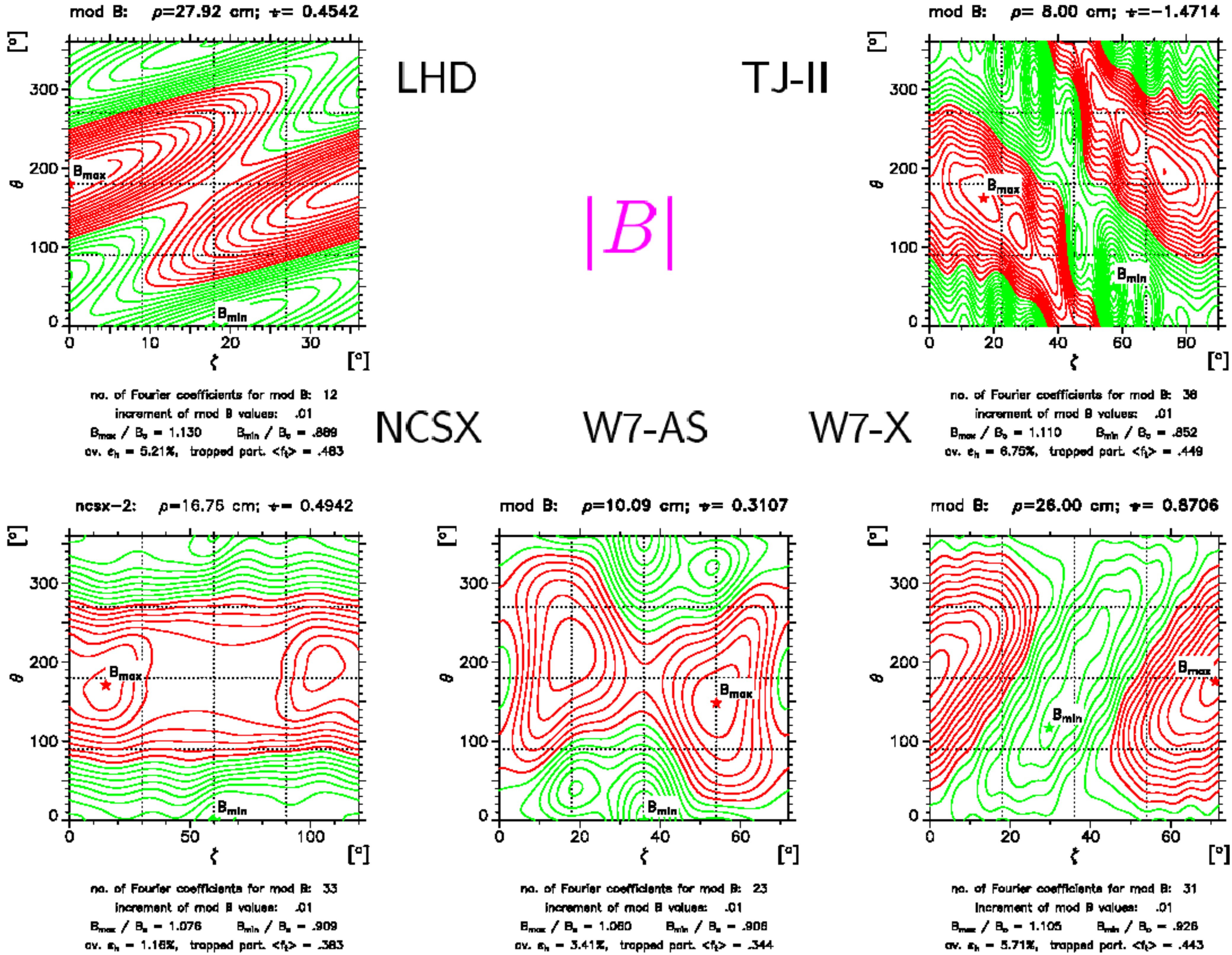
P(fusion) = 2.364 GW

P(electric) = 1 GW

Ignited



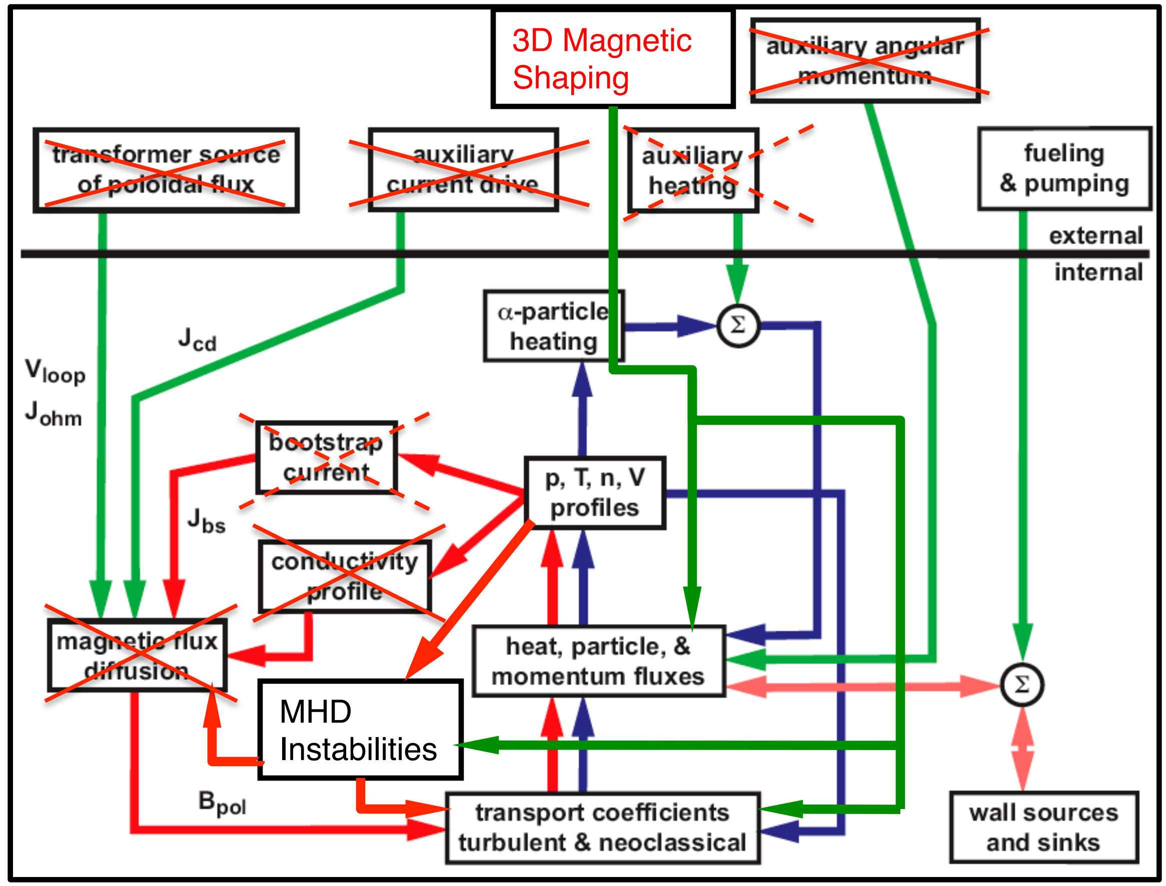
Aries-	-I	-RS	-CS	-AT	-CS
Blanket			LiPb/FS	LiPb/SiC	LiPb/SiC
COE(92)	99.7	75.8	61.3	47.5	48.



mod B: ρ=27.92 cm; ι= 0.4542
LHD
TJ-II
mod B: ρ= 8.00 cm; ι=−1.4714
|B|
B_max
B_min
θ
ζ
[°]
no. of Fourier coefficients for mod B: 12
increment of mod B values: .01
B_max / B_0 = 1.130 B_min / B_0 = .869
av. ε_h = 5.21%, trapped part. <f_t> = .483
NCSX
W7-AS
W7-X
no. of Fourier coefficients for mod B: 38
increment of mod B values: .01
B_max / B_0 = 1.110 B_min / B_0 = .852
av. ε_h = 6.75%, trapped part. <f_t> = .449
ncsx-2: ρ=16.76 cm; ι= 0.4942
mod B: ρ=10.09 cm; ι= 0.3107
mod B: ρ=26.00 cm; ι= 0.8706
no. of Fourier coefficients for mod B: 33
increment of mod B values: .01
B_max / B_0 = 1.076 B_min / B_0 = .909
av. ε_h = 1.16%, trapped part. <f_t> = .383
no. of Fourier coefficients for mod B: 23
increment of mod B values: .01
B_max / B_0 = 1.080 B_min / B_0 = .906
av. ε_h = 3.41%, trapped part. <f_t> = .344
no. of Fourier coefficients for mod B: 31
increment of mod B values: .01
B_max / B_0 = 1.105 B_min / B_0 = .926
av. ε_h = 5.71%, trapped part. <f_t> = .443

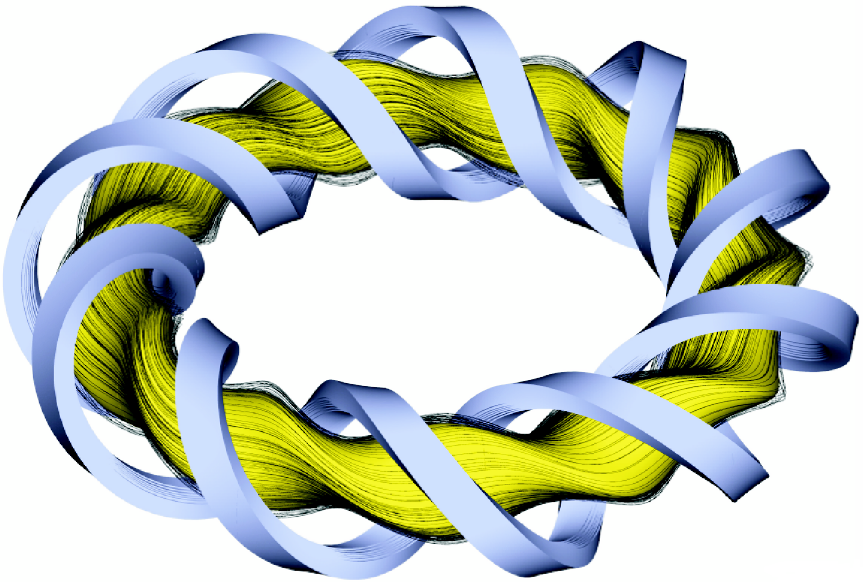
Stellarators: Eliminate or Weaken Non-linearity

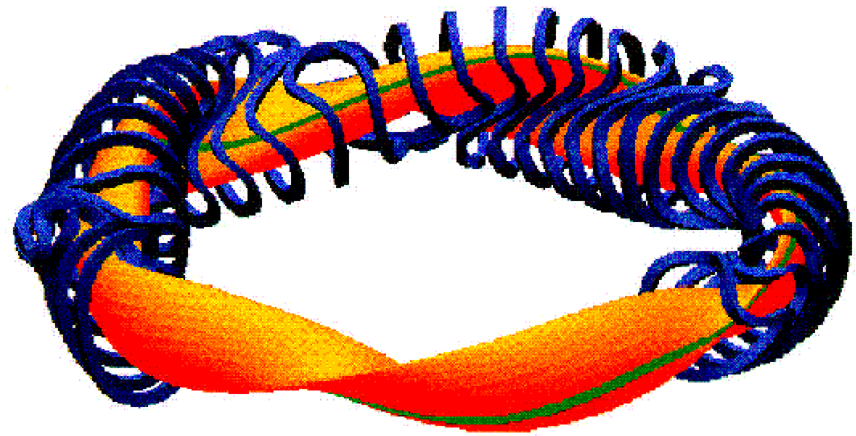
- Equilibrium maintained by coils, not current drive. Simple steady-state.
- Equilibrium maintained without plasma.
- Not limited by MHD instabilities. No need to control profiles.
- Greatly simplify plasma control needs.



ala Politzer, 2005

Large International Superconducting Stellarators





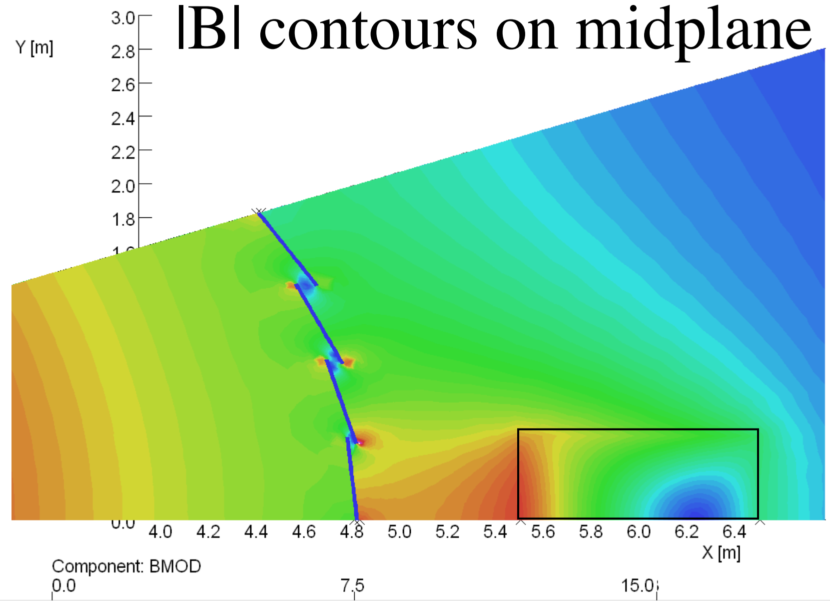
- **Large Helical Device (Japan)**
  - Non-symmetric
  - A = 6-7, R=3.9 m,   B=3T

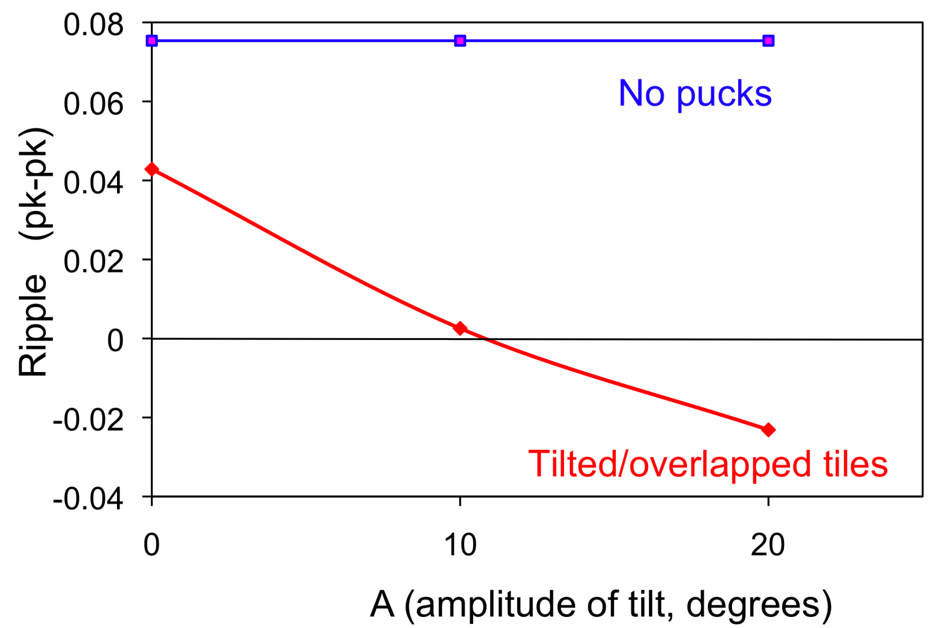
- **Wendelstein 7-X (Germany)**
  - QP optimized design
  - A = 11, R=5.4 m,   B=3T

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

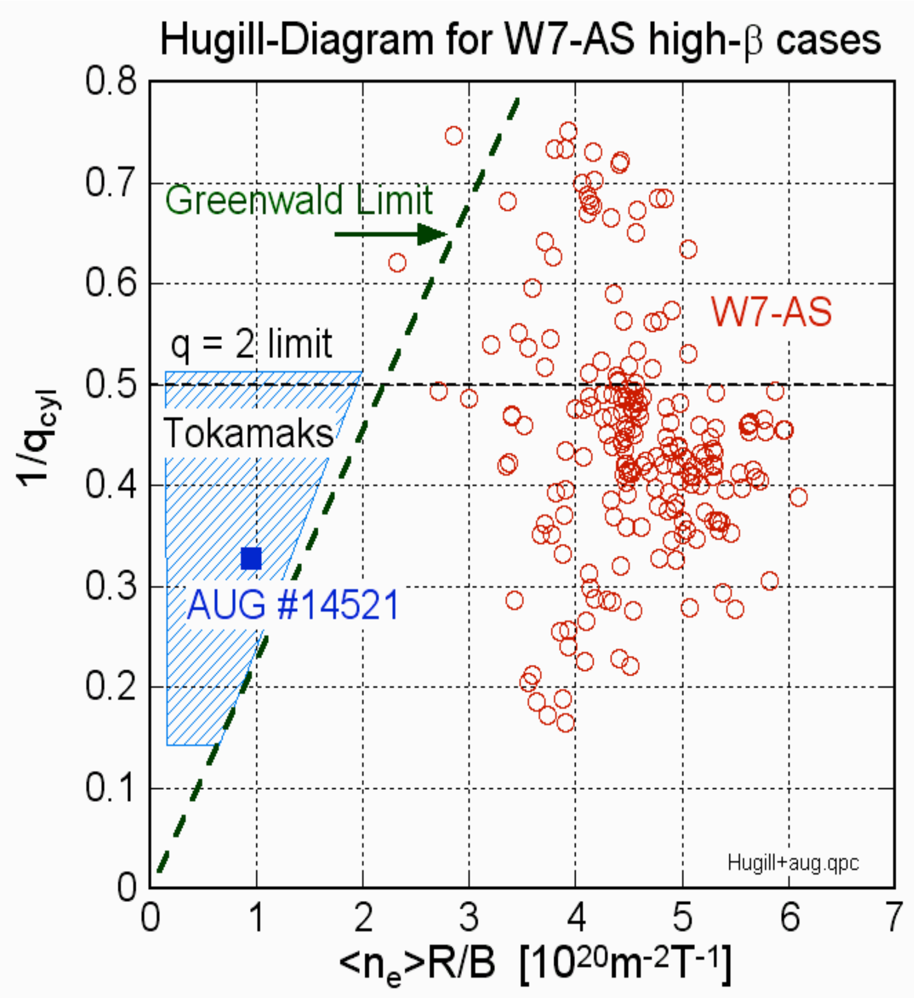
Trial Problem: Eliminate TF Ripple For 8-coil TF

- Simple geometry
  - 8 TF-coils at R=6m axisymmetric
  - Use HTS tiles at R=4.8m to eliminate ripple at R≤4m
- Tilt tiles so that they interact with toroidal field
- Can zero or reverse ripple Magnitude of |B| change similar to need for stellarator





Stellarator Operating Range is much larger than for Tokamaks

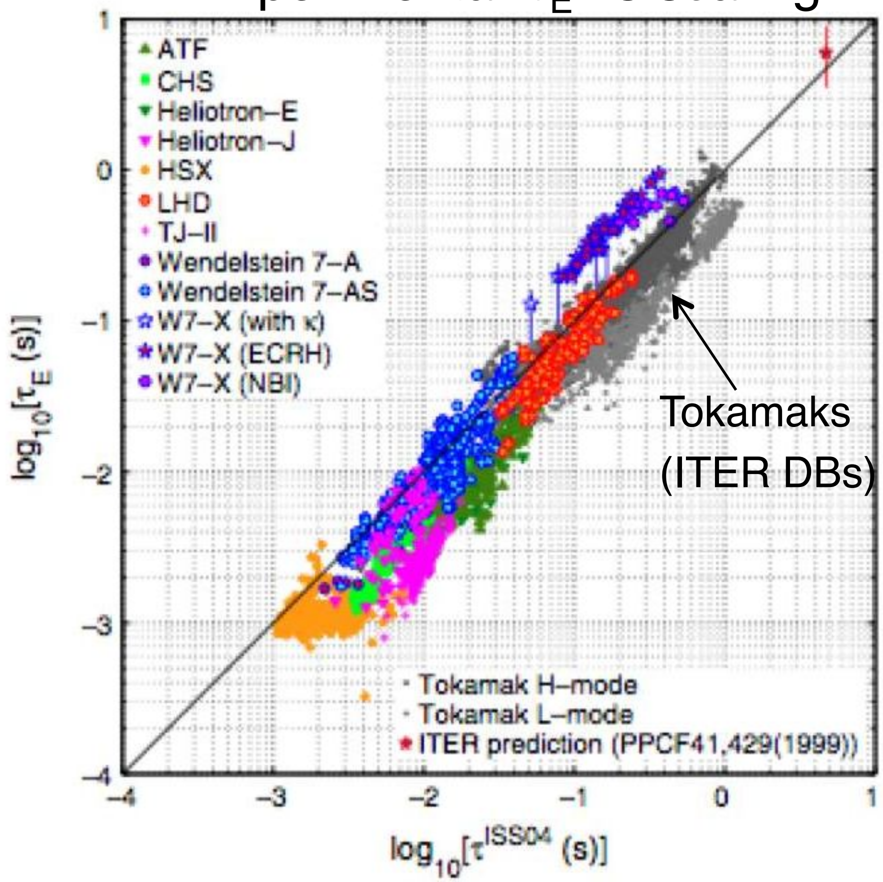


- Using equivalent toroidal current that produces same edge iota in Greenwald evaluation.
- LHD $n_{e0} = 10^{21}\ m^{-3}$ at B = 2.7 T
  3-5 X Greenwald limit
- No disruptions.
  Limits are not due to MHD instabilities.
- High density favorable:
  - Lower plasma edge temperature, Eases edge design
  - Reduces energetic particle instability drive

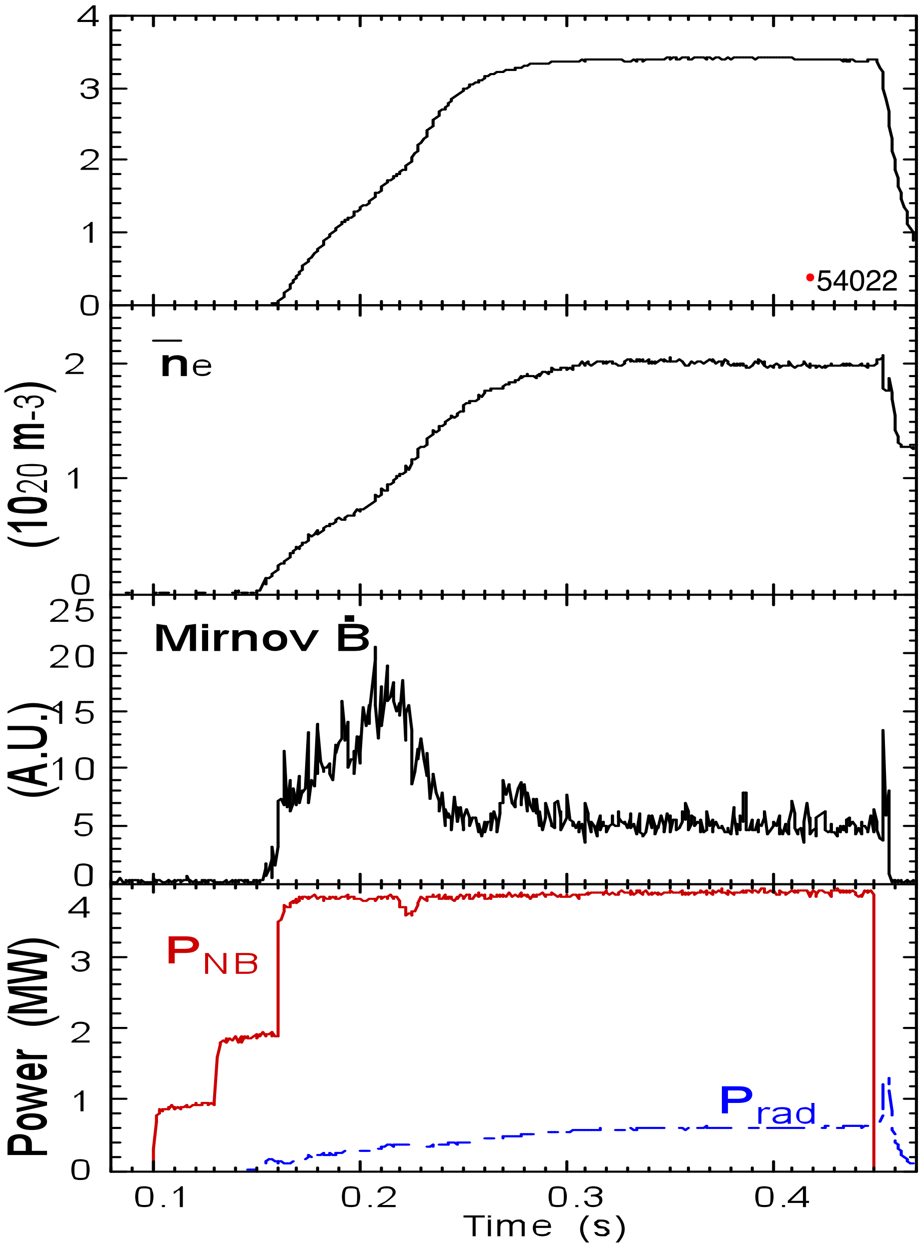
Stellarator Energy Confinement Similar to Tokamaks

- Stellarator τ_E similar to ELMy H-mode
- T_i = 6.8 keV without impurity accumulation (LHD)
- Discharge duration ~ 1 hr with P ~ 0.6 MW, limited by PWI

Experimental τ_E vs scaling

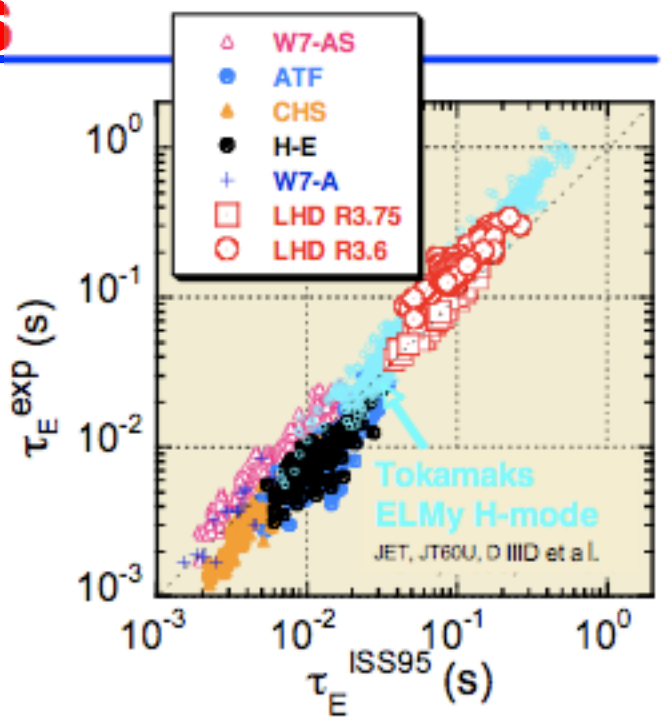


• $\langle\beta\rangle \approx 3.4$ % : Quiescent, Quasi-stationary

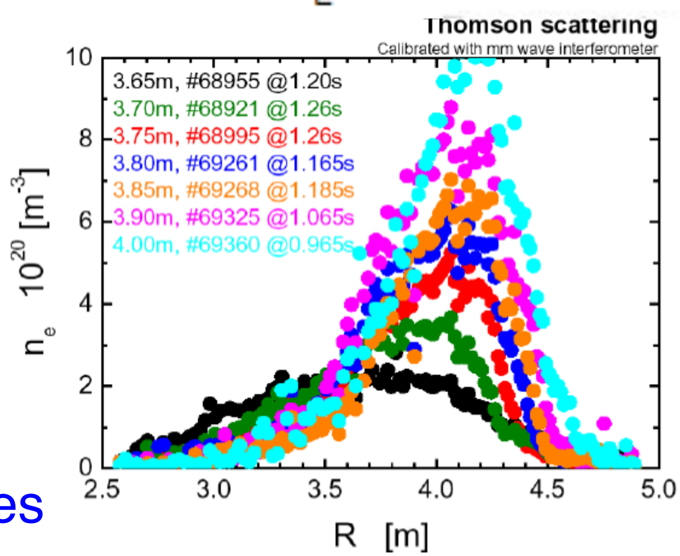


- B = 0.9 T, $iota_{vac} \approx 0.5$
- Almost quiescent high- β phase, MHD-activity in early medium-β phase
- In general, β not limited by any detected MHD-activity.
- $I_P = 0$, but there can be local currents
- Peak β~8%
- Similar plasmas with B = 0.9 – 1.1 T, either NBI-alone, or combined NBI + OXB ECH.
- Much higher than predicted linear stability β limit ~ 2%

Stellarators are Achieving Outstanding Results

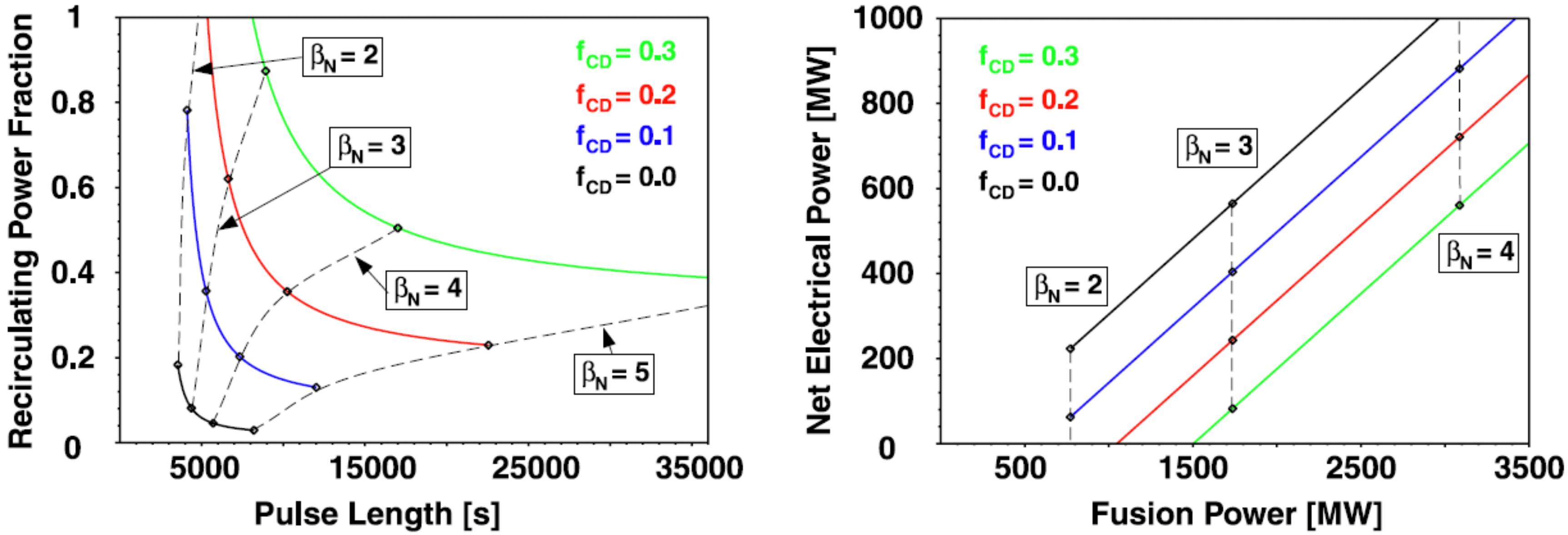


- Quiescent high beta plasmas,
  limited by heating power & confinement
  - LHD β = 5.2% transiently; 4.8% sustained
  - W7AS β > 3.2% for 120 τ_E
- τ_E similar to ELMy H-mode
- Improved confinement with quasi-symmetry
  - HSX finds reduced transport of momentum, particles, and heat with quasi-symmetry.
- Very high density operation, limited only by heating power, without confinement degradation
  - Up to 5 x equivalent Greenwald density (W7AS)
  - LHD $n_e(0)$ up to 10^{21} m^{-3} at B=2.7T !
- Importance of divertors to control recycling
- Steady state: LHD pulse lengths up to 55 minutes



Example: ITER-like case with R_0=7.5 m

Vary β_N between 2 and 5 and f_{CD} between 0 (ohmic) and 0.3 and assume conventional technology (η_{WP}=0.25, η_{TD}=0.3, P_{BOP}=50 MW, η_{BOP}=0)



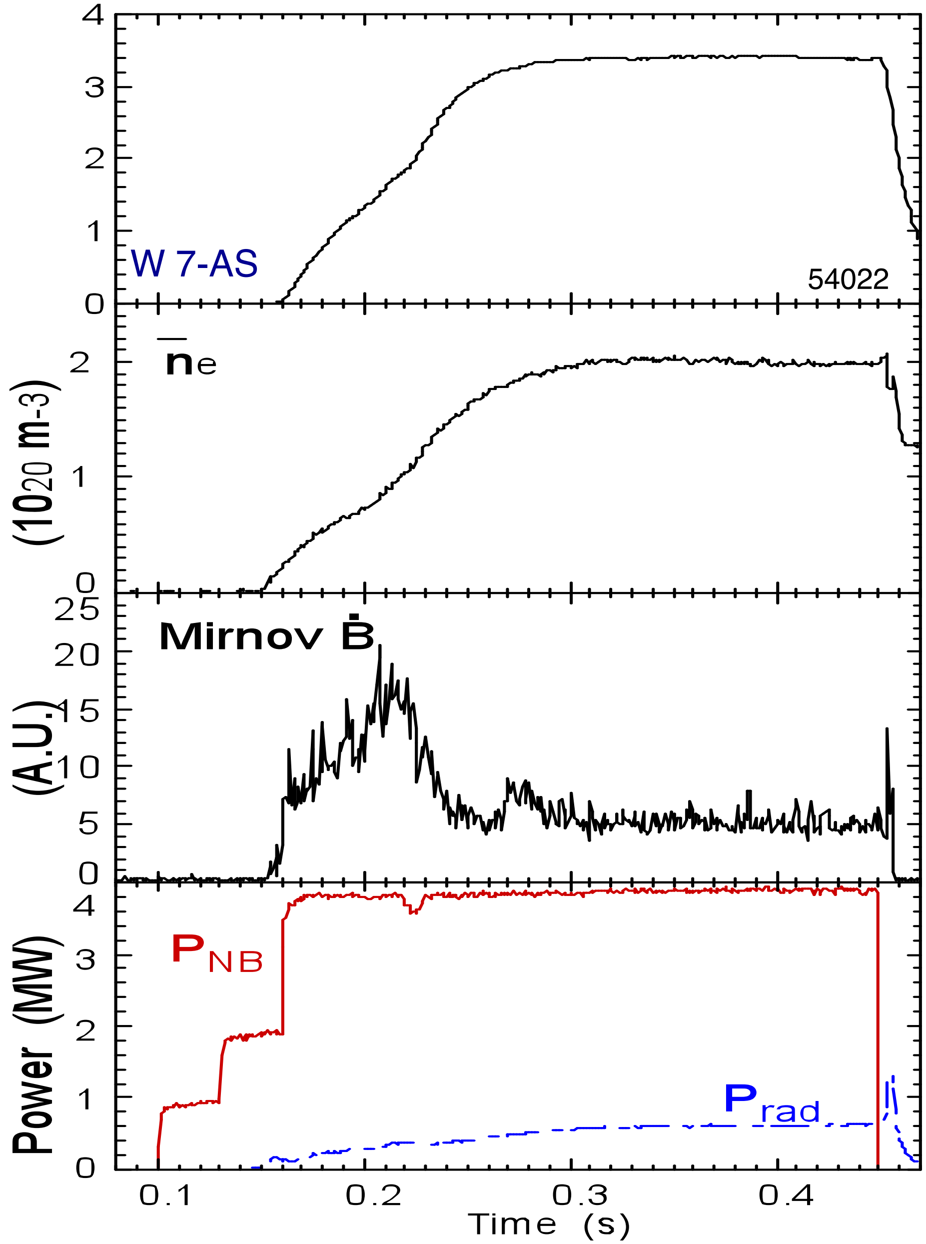
The objectives of acceptable f_{rec} and significant $P_{el,net}$ can be fulfilled relatively easily (e.g with f_{CD}=0.1 and β_N=3, $P_{el,net}$=350 MW with f_{rec}<0.4), but pulse length is nowhere near the target!

Even P_{fus}=3 GW (β_N=4.2, f_{CD}=0.2, f_{rec}=0.33) only gives $\tau_{pulse} \approx$ 3 hrs

Zohm 2011

H. Zohm, Fus. Sci. Technology 58 (2010) 613.

High β: Quiescent, Quasi-stationary



- Almost quiescent high-β phase, Peak β~8%

  MHD-activity only in early medium-β phase

- No disruptions: β not limited by any detected MHD-activity.

- Similar plasmas with B = 0.9 – 1.1 T, either NBI-alone, or combined NBI + OXB ECH.

- Much higher than predicted linear MHD stability β limit ~ 2%