# *Prospects and Risk Tradeoffs for Steady-State MFE*

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**MFE Roadmapping Workshop**
**9 September 2011**

# Outline

- Introduction

  Plasma Issues for steady-state DEMOs

- Steady-state tokamak
- Steady-state stellarators
- Summary

# Lots of Challenges for a Fusion Energy System

- In US: ReNeW, FESAC studies
- Separate talks here

  Divertor exhaust loads, PFCs

  Materials & technology

  Current drive

- ITER issues continue: ELMs & Disruptions
  - Worse in DEMO: more energy, higher forces
  - PFC armor must be much thinner to achieve TBR > 1

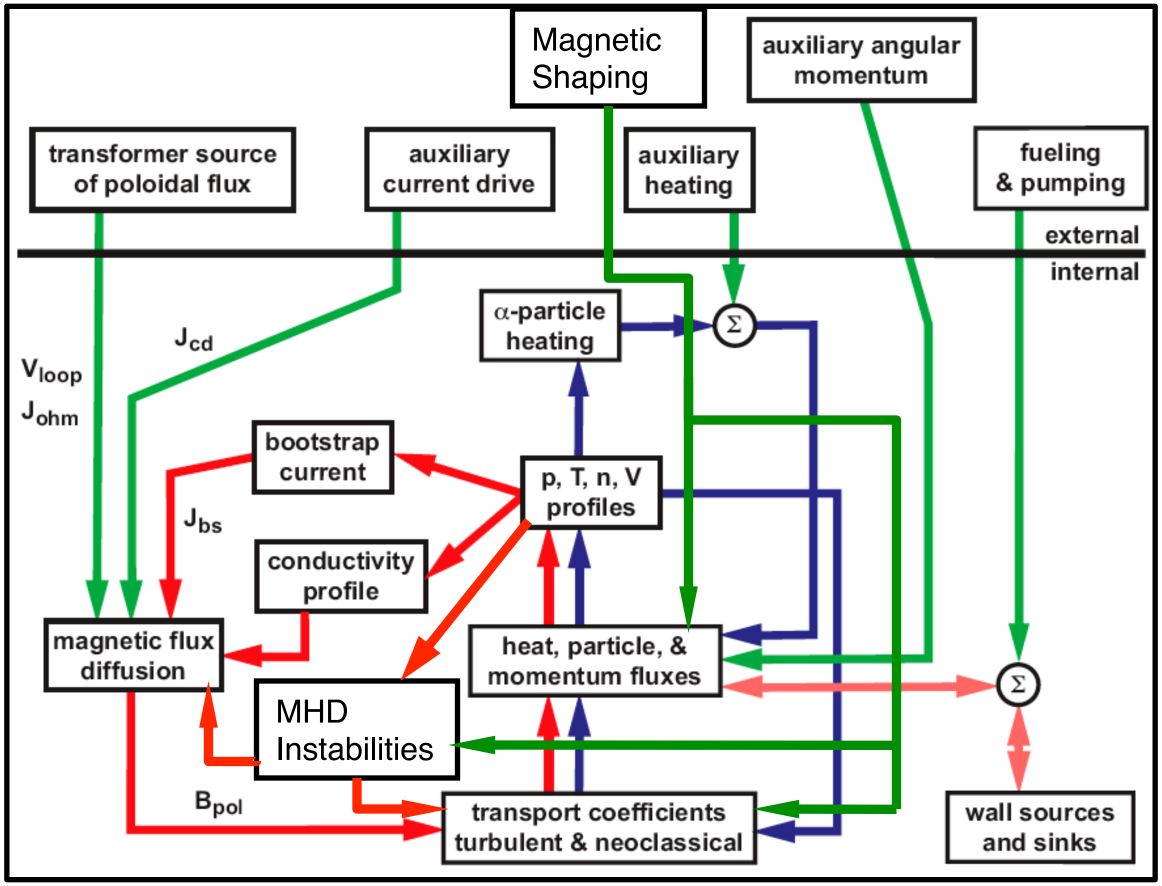
    most reactor designs have 1-3 mm of W armor

    ITER has 1cm Be/W plus 2.5cm of Cu (ΔTBR 12%)

  Disruptions and ELMs must be reliably eliminated

# Burning tokamak plasmas: Very non-linear

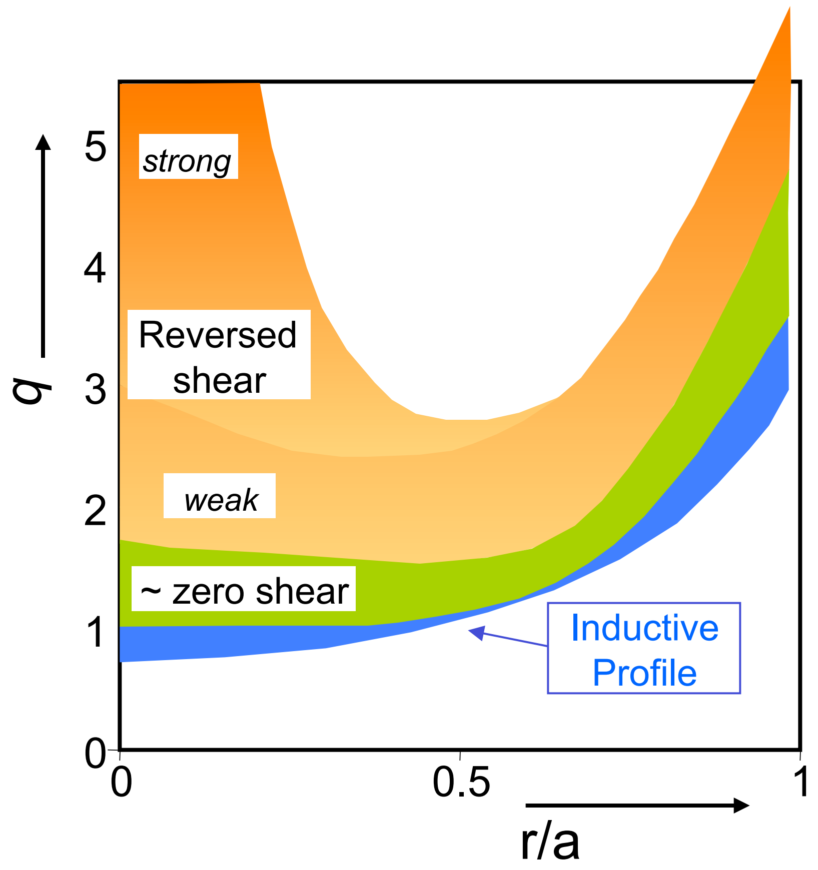
- Fusion heating
- Turbulent transport
- MHD - kinetic interactions
- Evolution of current profile interaction with transport



ala Politzer, 2005

- Need to actively control burning plasma to achieve steady state
- High-Q implies only have weak actuators.

# Steady-state tokamak: how much bootstrap?



- Need to maintain current / q-profile without inductive current
- Highest Q with maximum self-generated bootstrap current
- Large bootstrap current makes hollow profile, changes transport and plasma stability.

Three Advanced Tokamak strategies: zero shear
weak reversed shear
strong reversed shear

# Substantial advances in Steady-State Tokamak Regimes

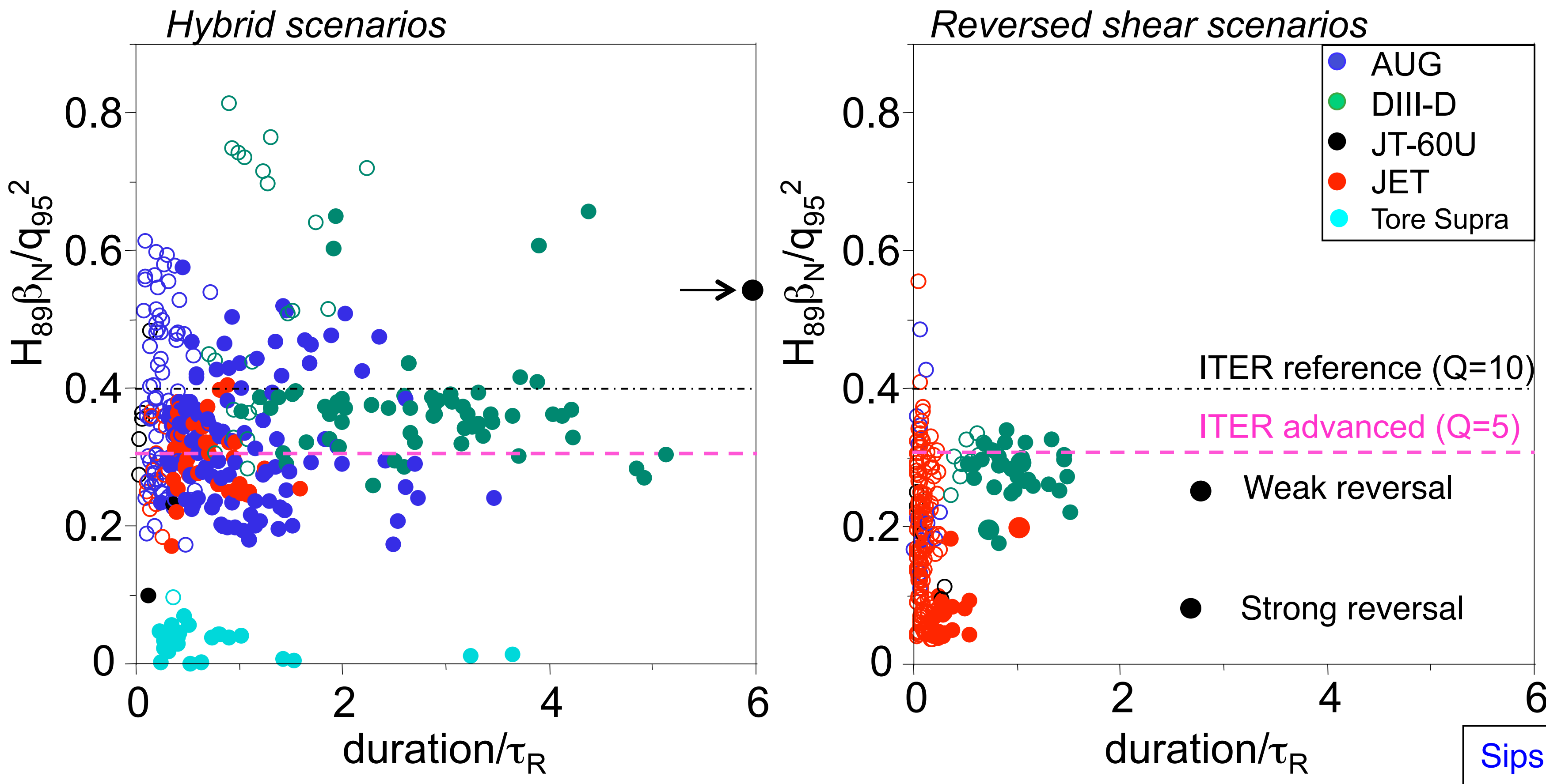
- Lots of significant work by AUG, DIII-D, JET, JT-60U
  in part to prepare for ITER

- 100% Non-inductive plasmas achieved in all three strategies
  ~ stationary for at least ~3 relaxation times for the current profile

- DIIID : extensive shape optimization. DN, $\kappa \sim 1.9$, $\delta \sim 0.6$, $\zeta \sim -0.25$
- JT-60U : extended to almost 30 sec.
- DIII-D, JT-60U, NSTX : above the no-wall limit

Will use $G = \beta_N H / q_{95}^2$ as a dimensionless metric for $nT\tau \sim Q$
using either $H_{89} = \tau_E$ / ITER-89P or $H_{98} = \tau_E$ / ITER-98(y,2)

# Similar Landscape on All Experiments

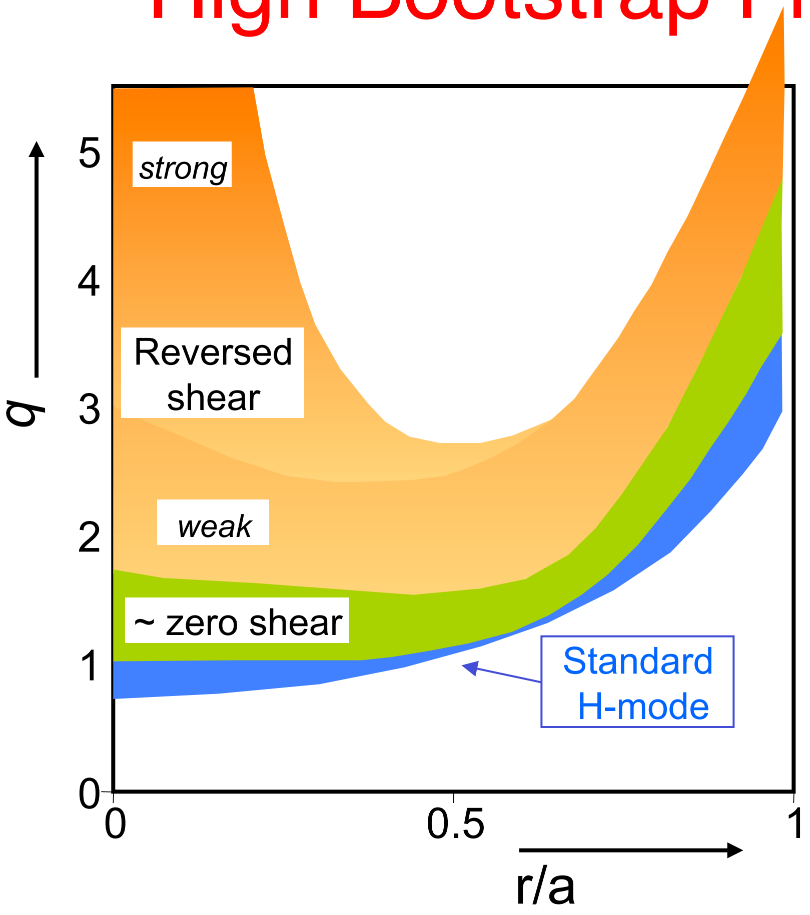


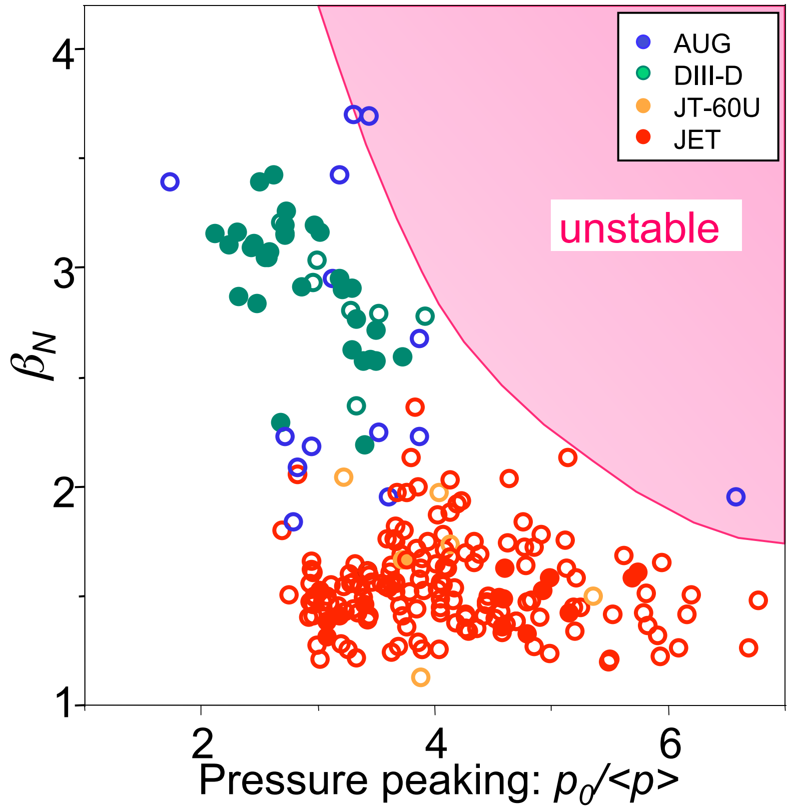
Sips 2005
+ new data

- JT-60U Hybrid sustained for 16 $\tau_R$
- All three regimes sustained to ~ 3 $\tau_R$ or longer, stationary.
- Bootstrap current fractions differ systematically

Hybrid $f_{boot} < 0.5$; Weak reversal $f_{boot} \sim 0.6$; Strong rev. $f_{boot} > 0.7$

# High Bootstrap Fraction More Unstable





- Higher bootstrap fraction => strong shear reversal
- Strong shear reversal => lower transport : ITB (internal transport barrier)
- ITB => pressure gradient driven flow shear & shift => stronger ITB
- Peaked profile / sharp gradient drives internal kink:  Reduced $\beta$ limit

# Limiting process similar on All Experiments

- High bootstrap, strong reversed shear: $\beta_N$ limited by strong ITBs
  produces extremely fast disruptions, often without precursors

- Weak reversed shear is a strategy to avoid ITBs
  limited by when they occur

- Hybrid and Weak shear reversal limited by external kinks / Wall modes

- Current experiments use beta-feedback of heating power to control all three regimes
  - \+ makes bootstrap evolution ~reproducible
  - \+ help control occurrence of non-linear ITB generation
  - \- Not prototypical for burning plasmas.
    Need to assess expected burn control strategies.
    May have slower reaction => impact performance limits.

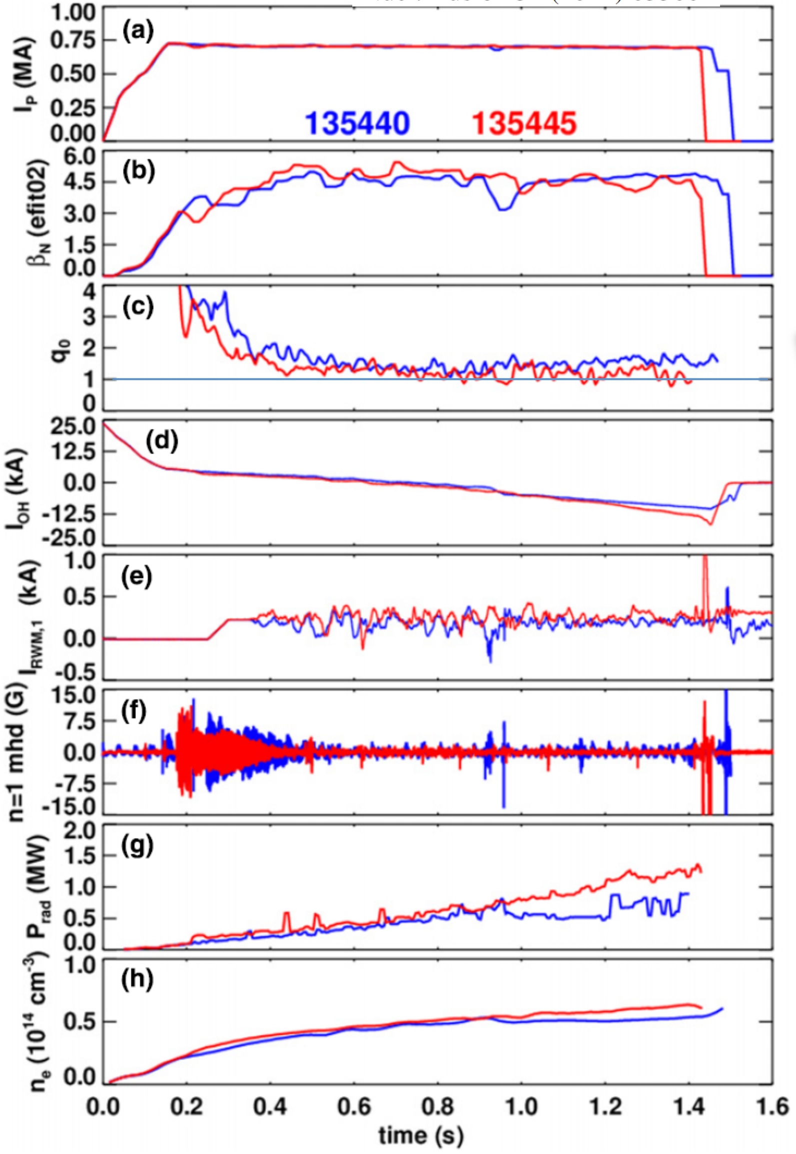
# Reactor Designs are Not Consistent with Sustained AT Characteristics

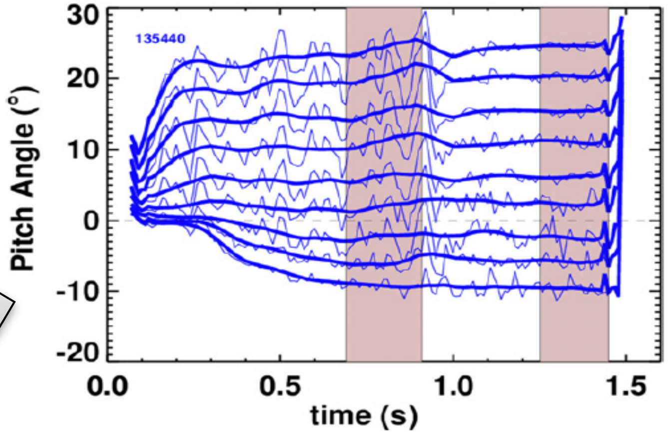
| | **Hybrid** | **Weak Rever** | **Strong Rever** | **Slim CS** | **CREST** | **EU AB** | **EU C** | **Aries-AT** |
|---|---|---|---|---|---|---|---|---|
| | | DIII-D | JT-60 | | Weak rev | | | Strong rev. |
| $q_{95}$ | 3.3 | 6.3 | 8.3 | 5.4 | 4.3 | 3.0 | 4.3 | 3.2 |
| $H_{98}$ | 1.5 | 1.5 | 1.8 | 1.3 | 1.3 | 1.2 | 1.3 | 1.7 |
| $\beta_N$ | 2.8 | 3.7 | 1.7 | 4.3 | 5.5 | 3.5 | 4 | 5.4 |
| $G_{98}$ | 0.38 | 0.14 | 0.044 | 0.19 | 0.39 | 0.47 | 0.28 | 0.90 |
| $f_{bootstrap}$ | ~0.4 | 0.65 | 0.75 | 0.77 | 0.83 | 0.45 | 0.63 | 0.91 |
| $n / n_{GW}$ | 0.4 | 0.5 | | 0.98 | 1.3 | 1.2 | 1.5 | 0.9 |

- Need to iterate designs using more realistic parameters

# NSTX nearly stationary 'hybrid'-like scenarios
# Close to FNSF goals, but still inductively sustained.

S. Gerhardt Nucl. Fusion **51** (2011) 033004





## NSTX Upgrade will extend NSTX to 100% non-inductive in support of ST-FNSF

| Parameter (sustained) | NSTX | NSTX-U | ST-FNSF* |
|---|---|---|---|
| Aspect ratio | 1.55 | 1.6-1.8 | 1.6 |
| $R_0$ [m] | 0.86 | 0.94 | 1.3 |
| Elongation | 2.7 | 2.8-3 | 3 |
| Toroidal field [T] | 0.41 | 0.7-1 | 2.9 |
| $\beta_T$ [%] | 14 | 8-12 | 10 |
| $\beta_N$ | 4.7 | 4.5 | 3.3 |
| $q_{min}$ | 1.3 | 1-2.5 | > 2 |
| q* | 4.4 | 4.5-8 | ≥ 4 |
| $f_{BS}$ [%] | 50 | 60-80 | 50 |
| $f_{NI}$ [%] | 60 | 100 | 100 |
| $H_{98}$ | 1-1.1 | 1-1.1 | ≤ 1.25 |

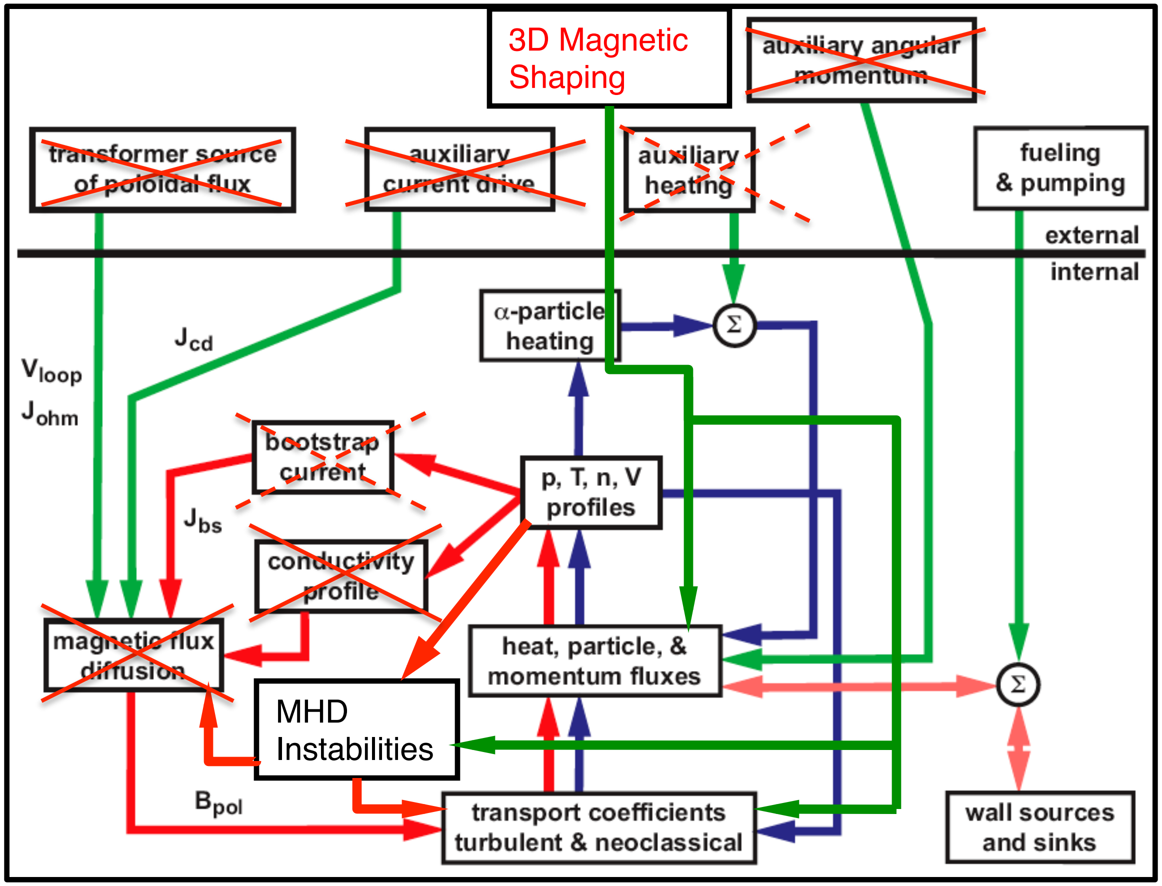
*Stage III-DT, Q=1.7, $P_{fusion}$ = 75MW, 1MW/m$^2$ neutron wall loading (M. Peng)

# RWM Stabilization by Fast Ions

- Fast ion precession can stabilize RWMs, allowing operation above the no-wall limit even at low rotation. [ Hu et al.]
- This has been observed experimentally on DIII-D, JT-60U, and NSTX.
- Analysis indicates that this may provide RWM stabilization in ITER, without external rotation drive [ Sabbagh].
- Experiments on DIII-D and NSTX also observe RWMs being triggered by fast-ion loss from fishbone-like instabilities, forcing the plasma below the no-wall $\beta$-limit.
- In future DEMOs, fast ion instabilities and Alfvenic instabilities may cause alpha-transport, and similarly destabilize the RWM. Need to keep $\beta_\alpha$ low, and assess fast-ion stability and transport.

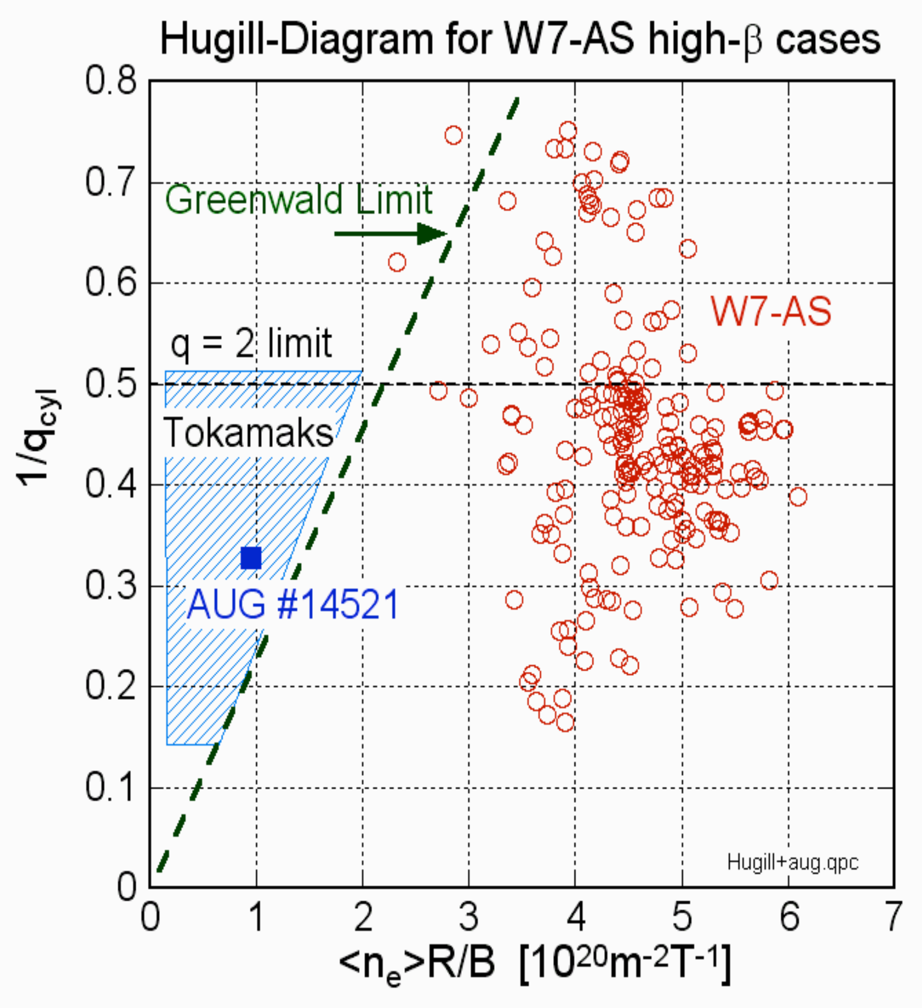
# Stellarators: Eliminate or Weaken Non-linearity

- 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

# Stellarator Operating Range is much larger than for Tokamaks

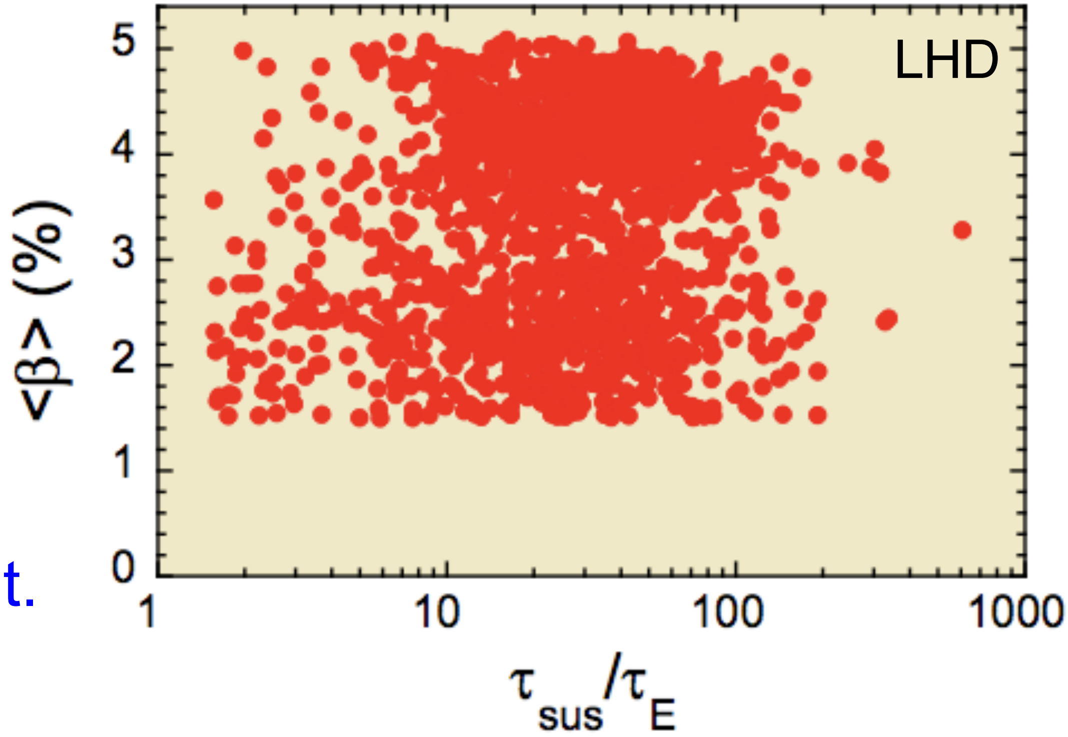


- Density limit ~5 X equivalent Greenwald density limit (from tokamaks).
- LHD $n_{e0} = 10^{21}\ m^{-3}$ at B = 2.7 T
- Can operate with q>2, even q>1
- No disruptions.

  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.



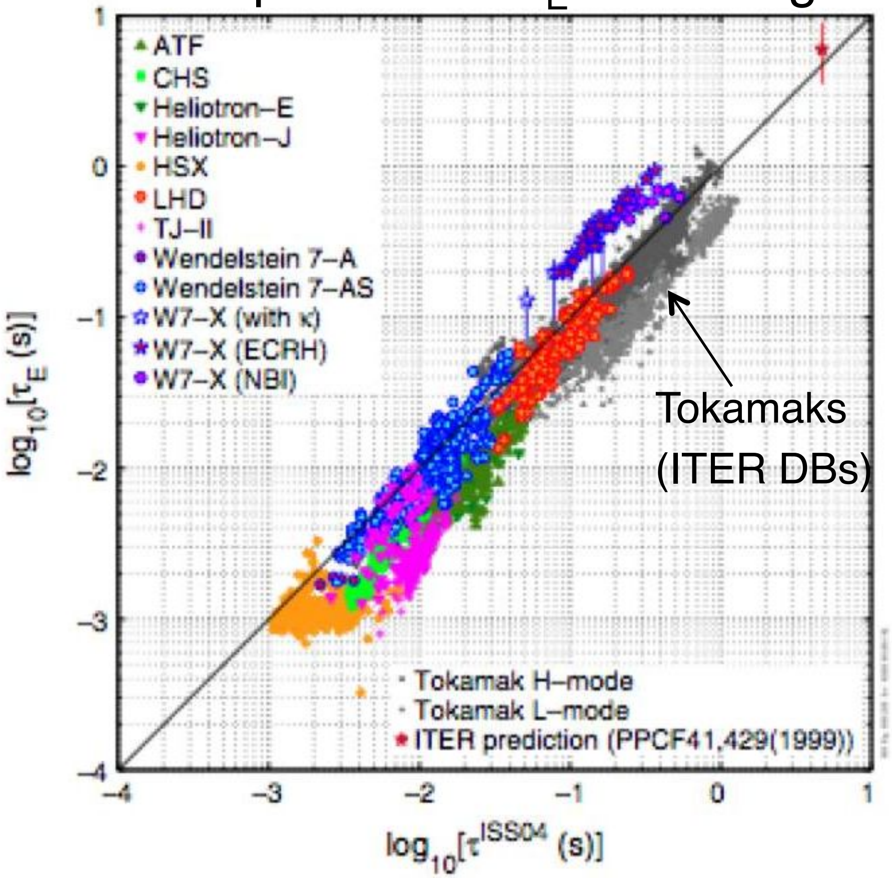
- Soft limit is observed, due
  to saturation in confinement.

- Highest β ~ twice ideal stability threshold. In W7AS: no MHD activity. In LHD: saturated MHD observed.
- What sets β-limit?? May be due to equilibrium limits.
  Can be improved by design.

# Stellarator Energy Confinement Similar to Tokamaks

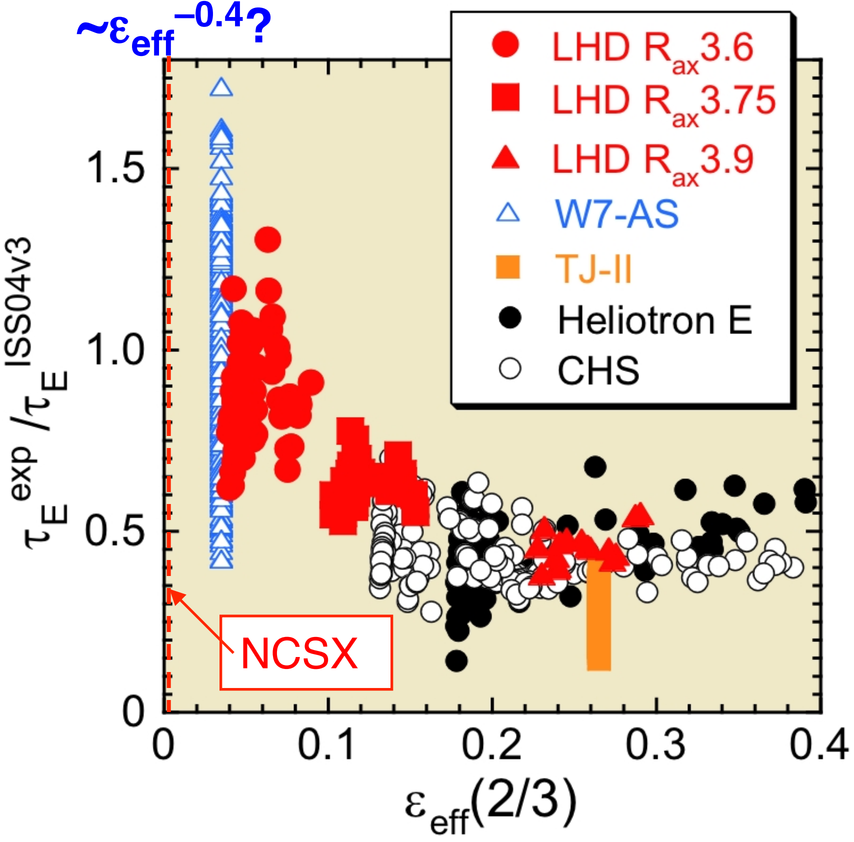
- ISS-04 confinement scaling derived from Stellarator L-mode data base. Gyro-Bohm like.
- Tokamak H-mode data plotted against stellarator scaling relation $\tau_E$-ISS04)
- Stellarator $\tau_E$ data similar to tokamak ELMy H-mode
- $T_i$ = 6.8 keV without impurity accumulation (LHD)

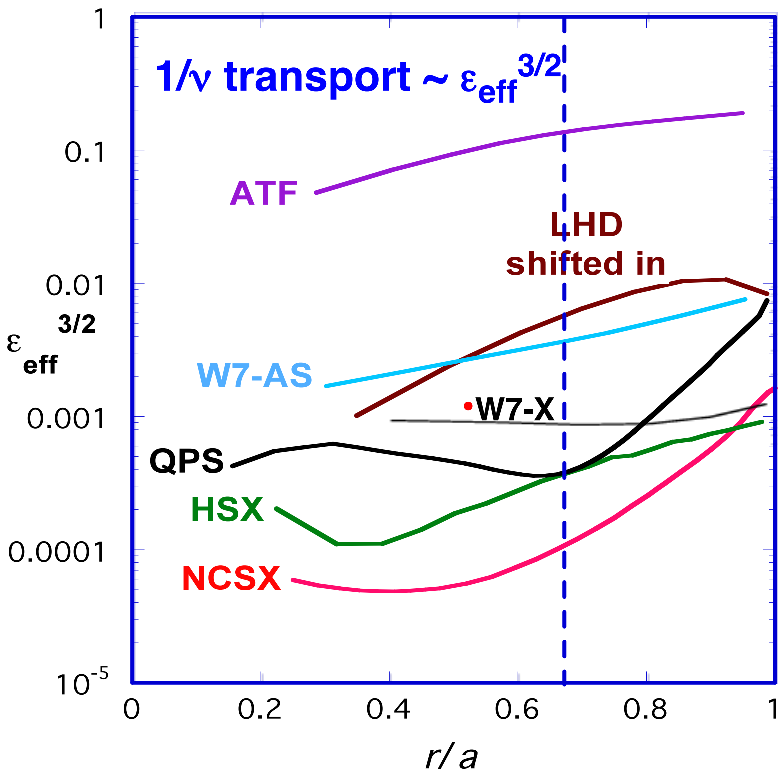
Experimental $\tau_E$ vs scaling



Tokamaks (ITER DBs)

# Low Ripple Gives Good Confinement





- Global confinement scaling for stellarators (ISS04v3) found strong dependence on ripple magnitude. Must involve anomalous transport also.
- H(ISS04) up to 1.5 obtained at low ripple
- H(ISS04) = 1.1 adequate for reactor, simultaneous with high beta.

# 3D Configurations: Need to Optimize for Good Confinement

3D: No symmetry ⇒ no conserved canonical momenta ⇒ lost orbits
⇒ rotation is strongly damped

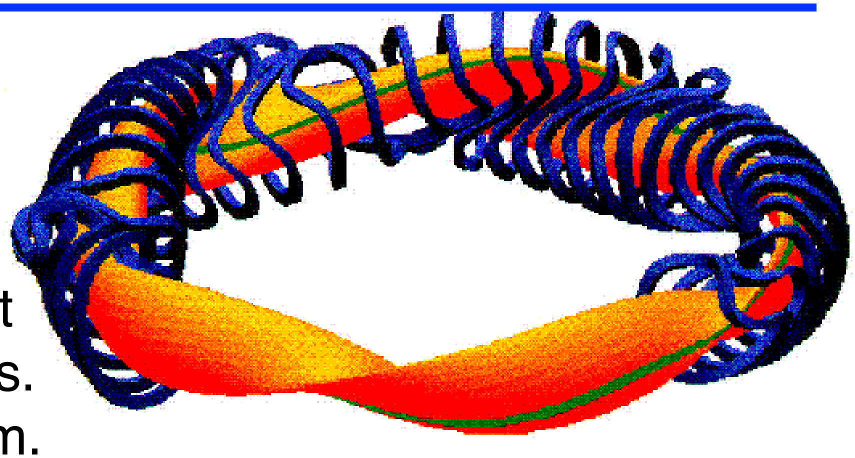
- ‘Quasi-symmetry’
  - (Boozer,1983) Orbits & neoclassical transport depend on variation of IBI within flux surface, not the vector components of B !
  - If IBI is symmetric in flux coordinates, get confined orbits like tokamak
  - Can be perfected on one surface in toridal system; degrades mildly

  ⇒ Neoclassical transport very similar to tokamaks (theoretically), undamped rotation

- Quasi-axisymmetry, Quasi-helical symmetry, Quasi-poloidal symmetry

  Differ in drift orbit widths and other physics characteristics

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



- 5 periods, R/⟨a⟩=11, R=5.4 m
  Superconducting coils

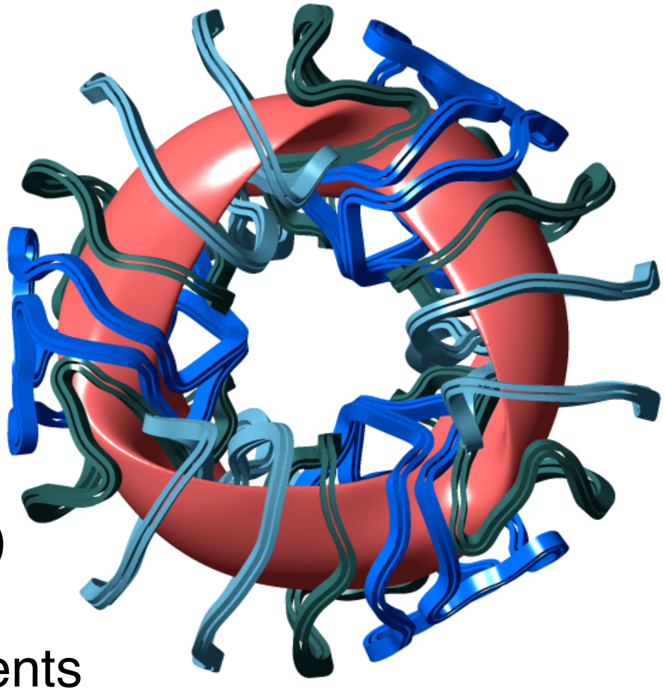
- Quasi-isodynamic: neoclassical transport
  minimized by minimizing drift-orbit widths.
  An approximation to quasi-poloidal symm.

- 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/⟨a⟩=4.4, ⟨κ⟩~1.8 , ⟨δ⟩~1
- Quasi-axisymmetric: tokamak with 3D shaping ripple-induced thermal transport insignificant. Build on ITER results.
- 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
- Designed to keep ~perfect flux surfaces to β=4.1% 2-fluid calculations indicate it may continue to β > 7%
- Passive disruption stability: equilibrium maintained even with total loss of β or $I_P$

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# Issues for Stellarators

US Assessment (ReNeW & FESAC):

1. Simplify coil designs

   Simplify maintenance strategies for blanket

2. Demonstrate integrated high performance: high-$\beta$, low collisionality

3. Confinement predictability

4. Effective 3D divertor design

# Compactness: How important

- Main interest in compactness is to reduce capital costs, increase mass-power density, improve competitiveness.
  - Non-trivial, given ITER's costs and budgeting challenges

- Clearly, compactness aggravates some engineering challenges

- Most design studies show shallow minimum & hard constraints (e.g. blanket thickness).

- In energy system, drives minimum power size.

Personal perspective:

- Any design will compromise between cost, engineering risk, perceived attractiveness to customer. Need to assess variations, maintain contingency. Compactness is only one of the characteristics.

# Summary

- Substantial advances in last 10 yrs. in understanding steady-state tokamaks and stellarators.
- AT experiments have achieved 100% non-inductive sustainment in 3 q-profiles, with varying amounts of bootstrap current. Very similar characteristics across all experiments.
- AT steady-state performance levels are lower than assumed in reactor designs. Reactor design groups should assess realistic performance, combined with realistic current drive efficiencies.
- Need to assess performance limits of control strategies that will be used for burning plasmas.

# Summary (2)

- Stellarators simplify physics non-linearities. Plasma equilibrium determined by coils.
- Simplify & reduce auxiliary technology needs
  - Don’t require steady-state neutral beams and RF-launchers in burning environment
- Steady-state, high-beta plasmas already demonstrated. Minimal recirculating power required.
- Robust confinement: no disruptions, can avoid edge instabilities (ELMs)
- Need to simplify coil engineering, maintainability.
- Need to demonstrate integrated performance, incl. divertor. How to best build on ITER?