Stellarator Reactors

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With material from Aries-CS, NCSX, QPS, LHD/CHS, and W7X Teams

PPPL Grad. Seminar

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Outline

- Motivation
- Physics Issues
- Engineering Issues
- Aries-CS & earlier studies
- Summary

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

$\rightarrow$ Can control rotational transform & shear from external coils

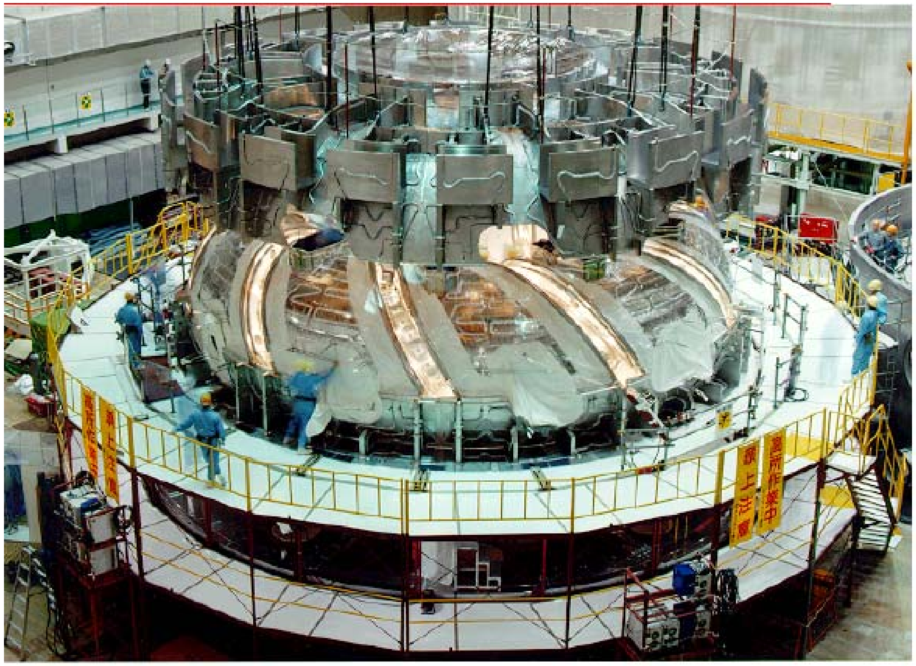
$\rightarrow$ No need for current drive to sustain configuration. Naturally compatible with steady state.

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

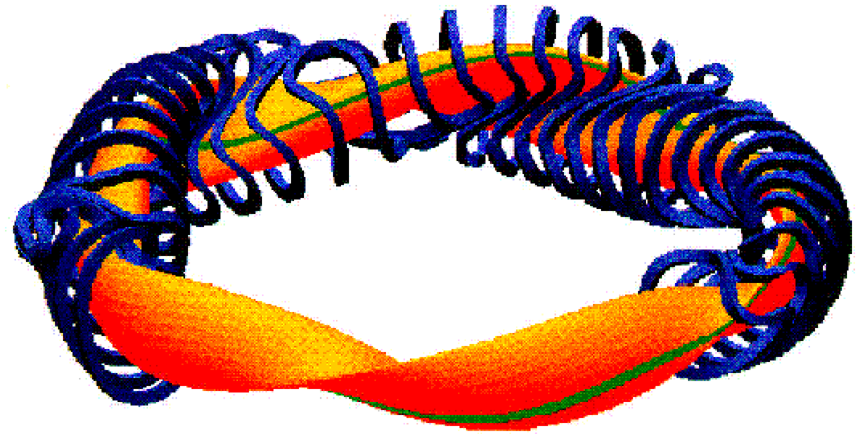
- Can use 3D plasma shaping to control physics properties (~ 40 shape parameters instead of ~4 for axisymmetric)

$\rightarrow$ More flexibility in configuration design

The World Stellarator Program is Substantial



Large Helical Device (Japan)
Enhanced confinement, high β;
$A = 6\text{-}7$, $R=3.9$ m, $B=3\rightarrow4$T

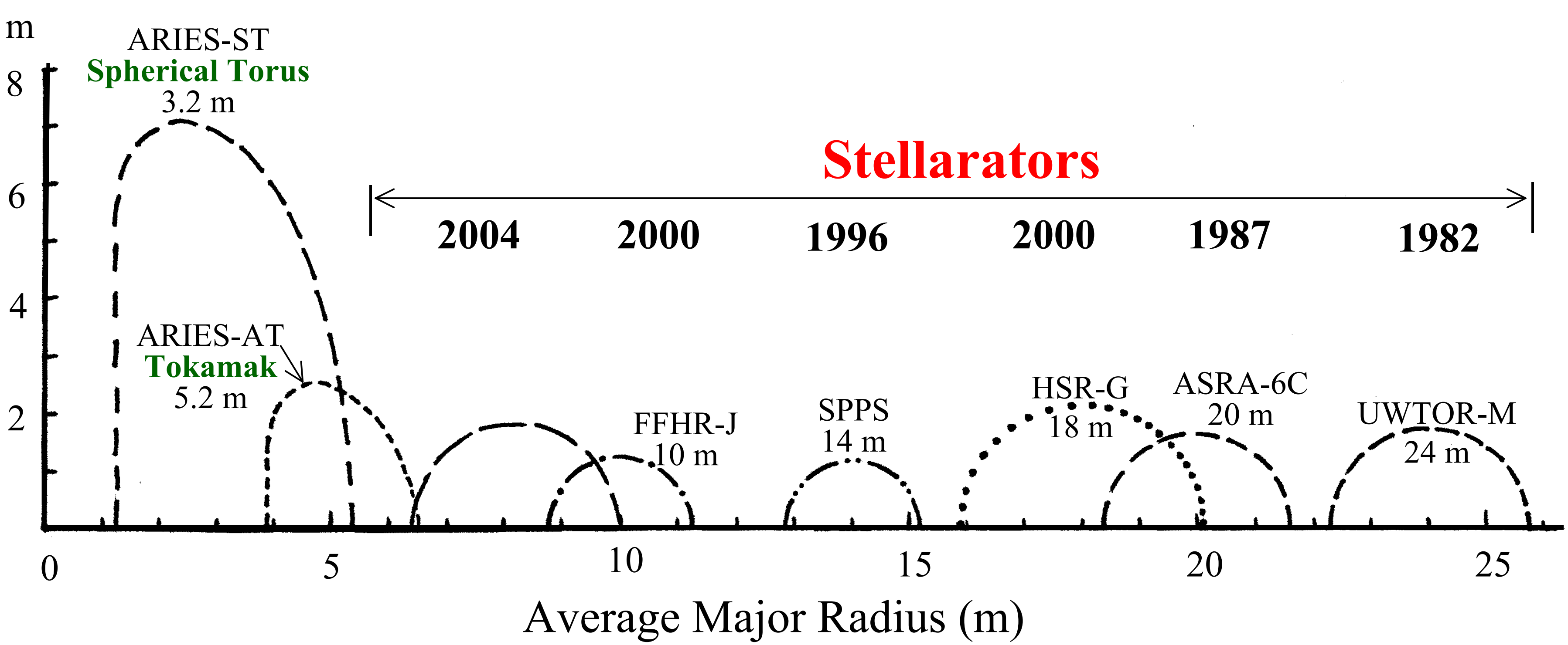


Wendelstein 7-X (Germany) (2010)
non-symmetric optimized design:
no current, $A = 11$, $R=5.4$ m, $B=3$T

- 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.

Stellarator Attractions Have Motivated A Succession of Reactor Studies

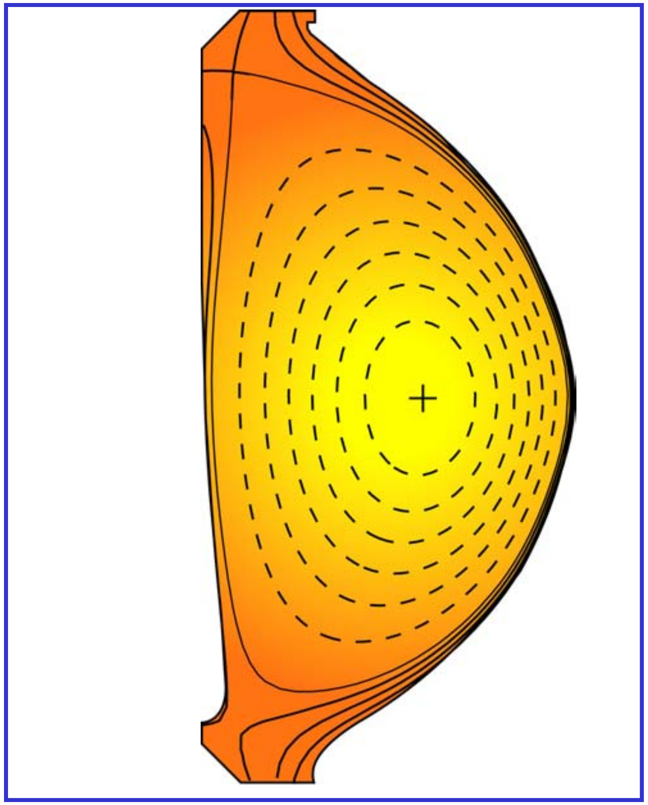


- Typically much larger R than tokamak designs
  => Motivated development of more compact designs
- SPPS projected cost of electricity similar to tokamaks, but higher initial capital cost
- Stellarator reactors expected to operate in true ignition. No need for current drive or profile control.

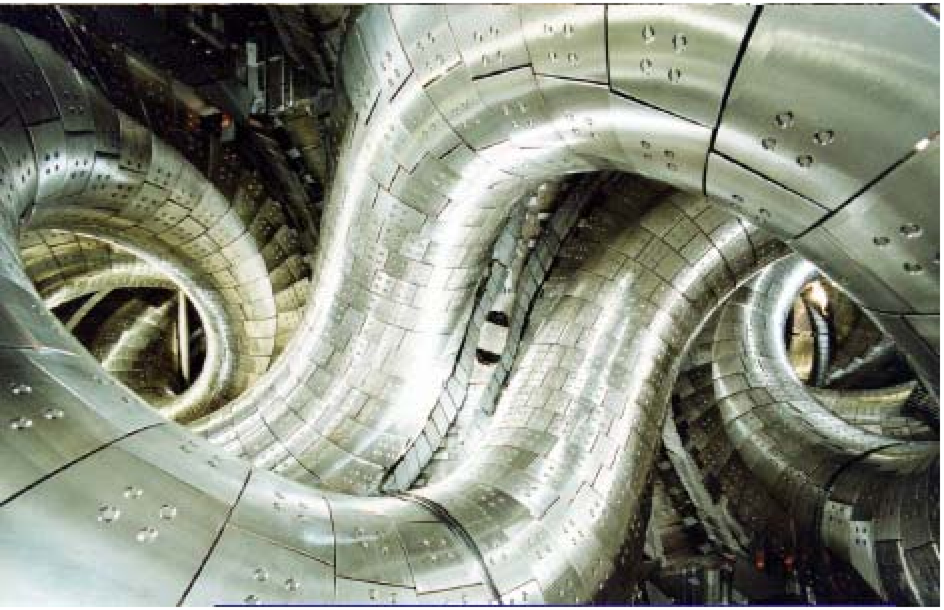
Fusion Plasma Challenges for Reactors

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

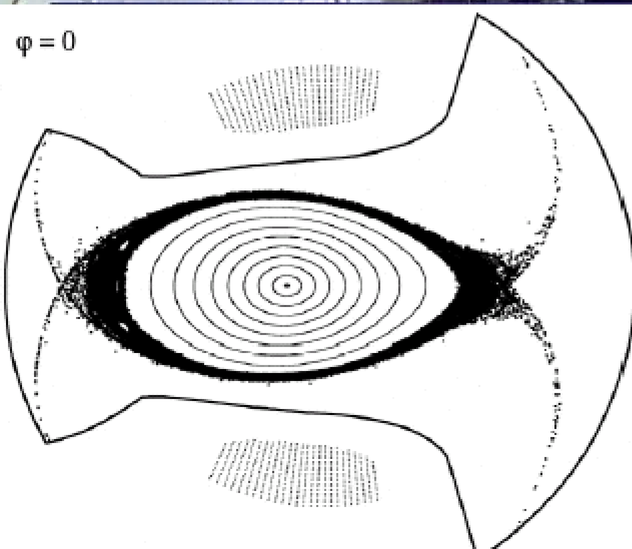


LHD: largest stellarator, record parameters



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

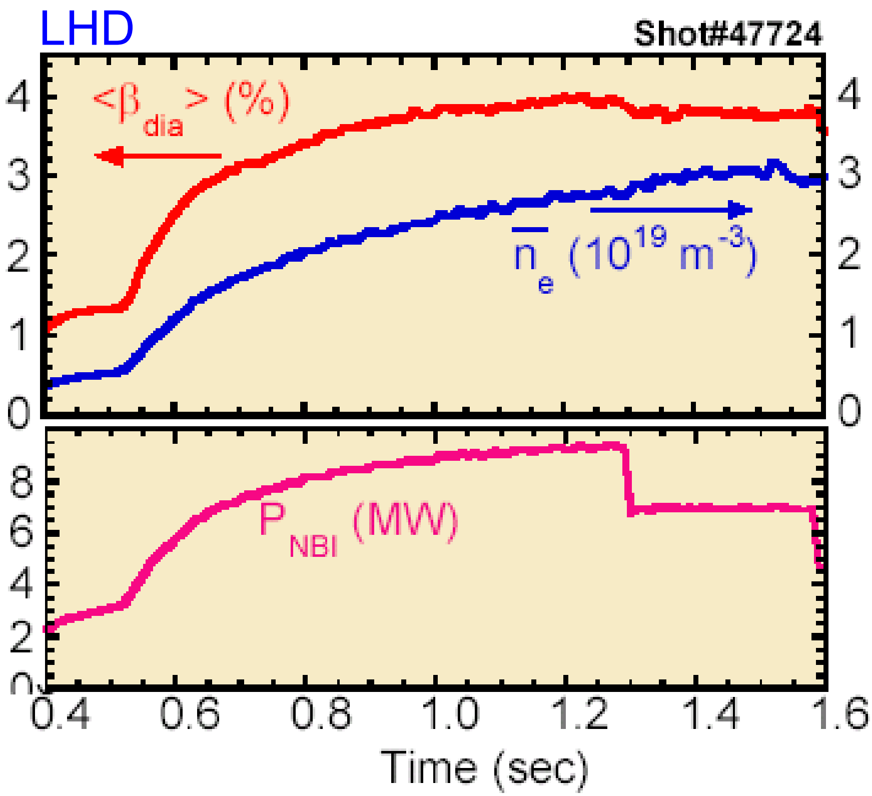
- T_e, T_i up to 10 keV
- $\langle\beta\rangle$ up to 4%, $\beta(0)$ up to 6%
- pulse lengths up to 756 s
- τ_E up to 0.36 s



Worlds largest superconducting coil system

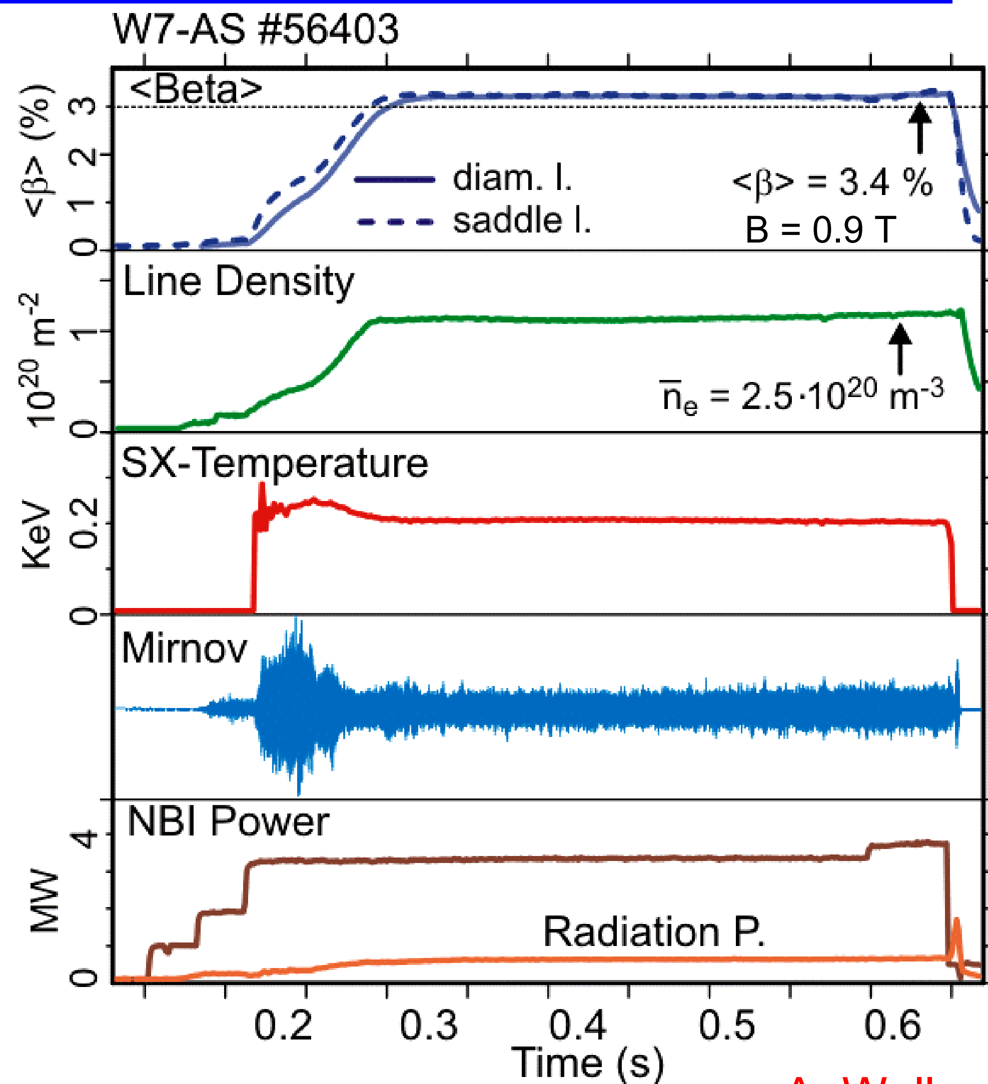
- 1 GJ of magnetic energy
- 850 ton cold mass at 4K

LHD & Wendelstein 7-AS: Quiescent high-β



B = 0.45 T, R=3.6 m

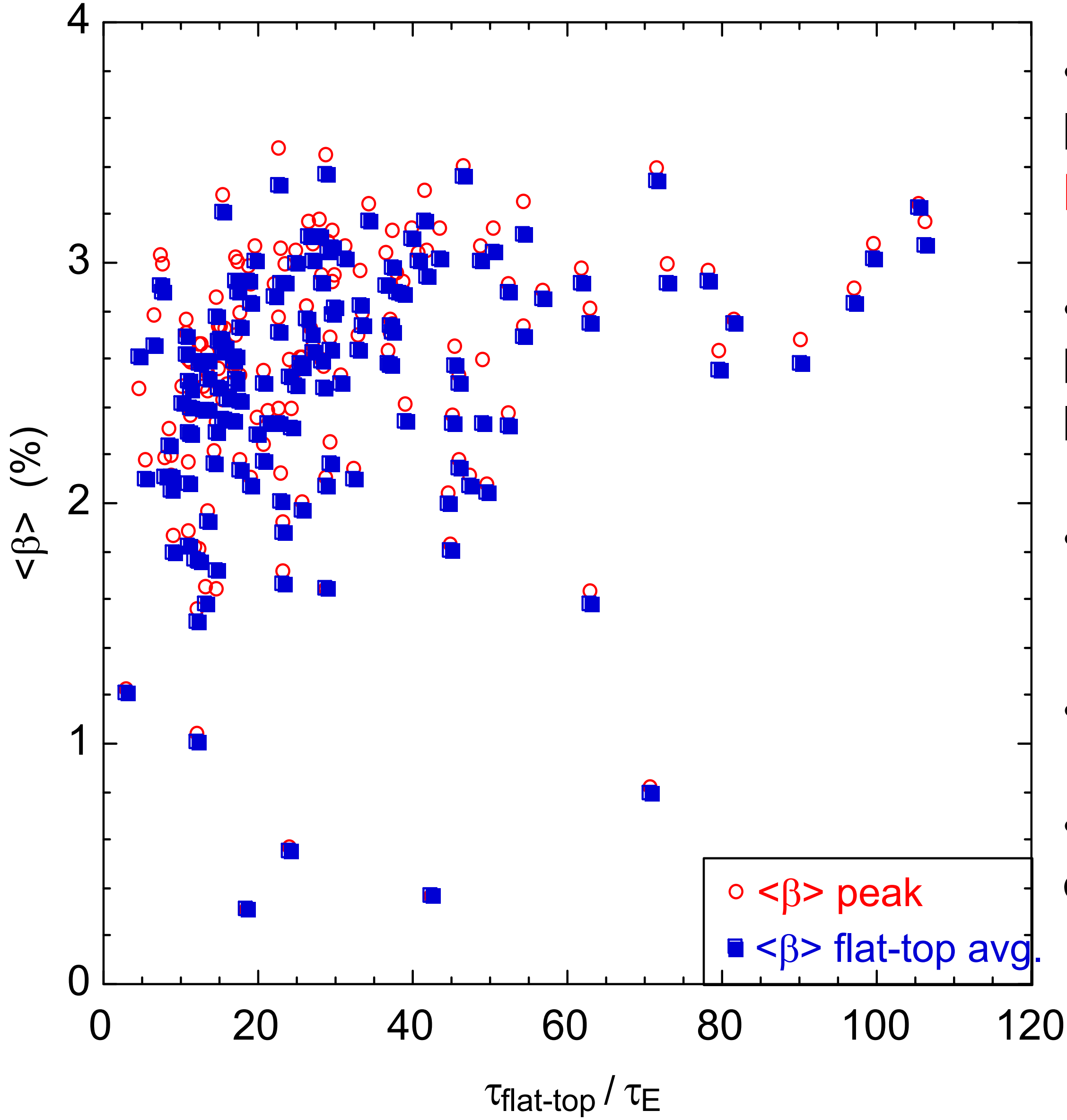
- K. Watanabe



- A. Weller

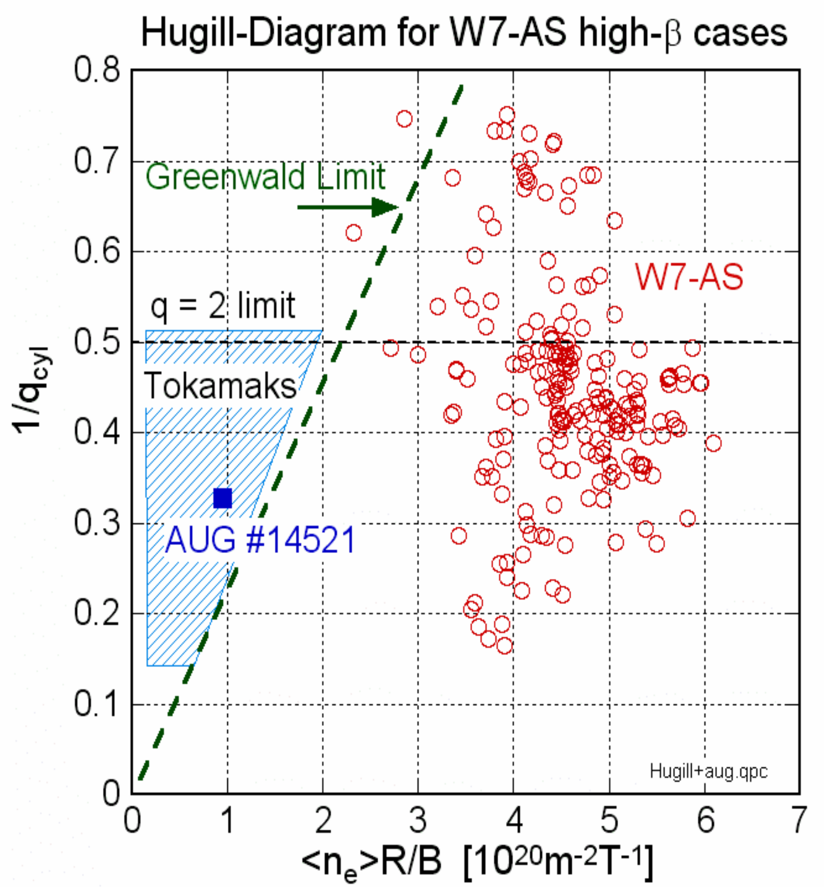
- No disruptions
- In both experiments, β is not limited by observed MHD instabilities
- In both experiments, predicted linear instability threshold is at much lower β

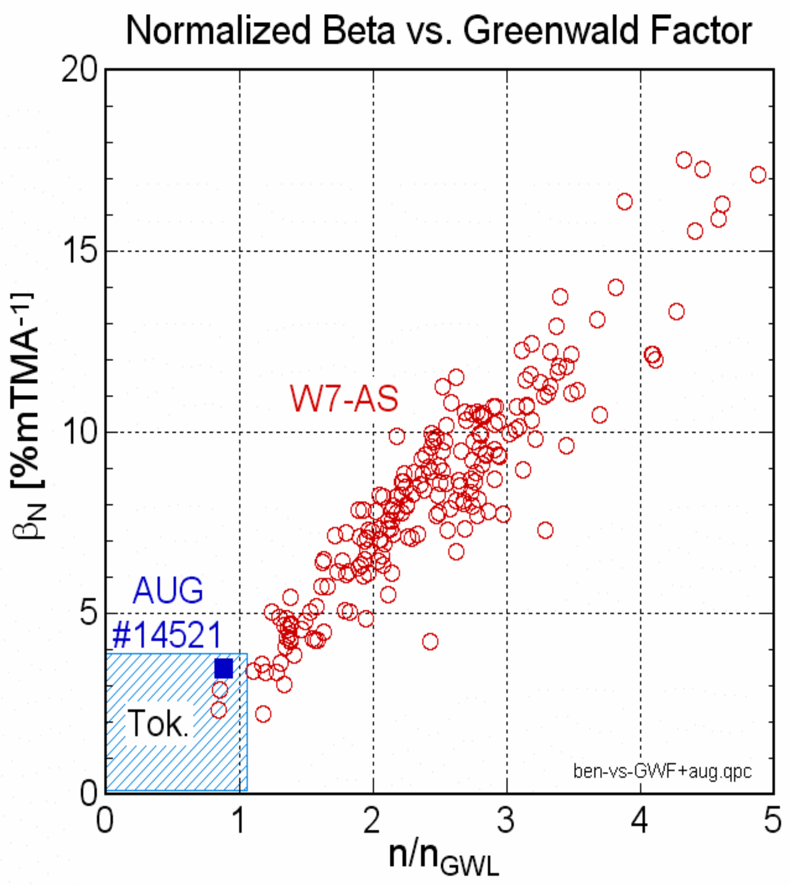
W7AS: ⟨β⟩ > 3.2% maintained for > 100 τ_E



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

W7-AS Operating Range much larger than Tokamaks

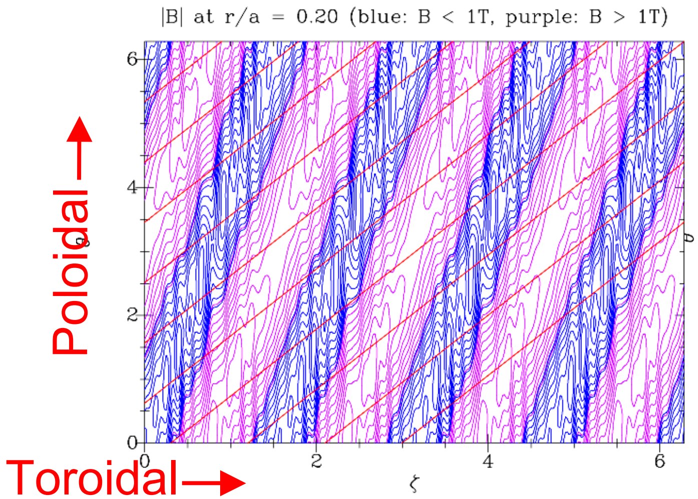


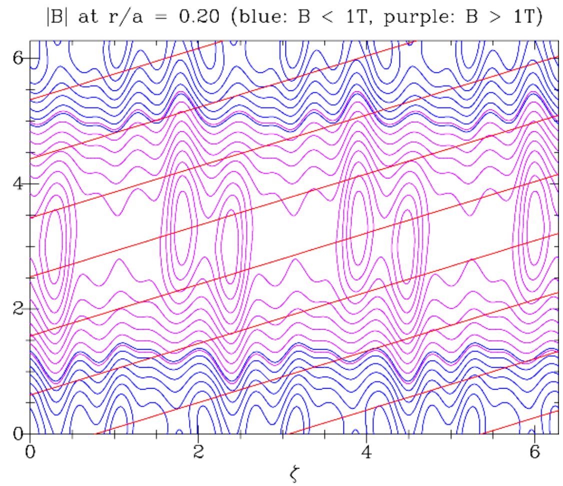


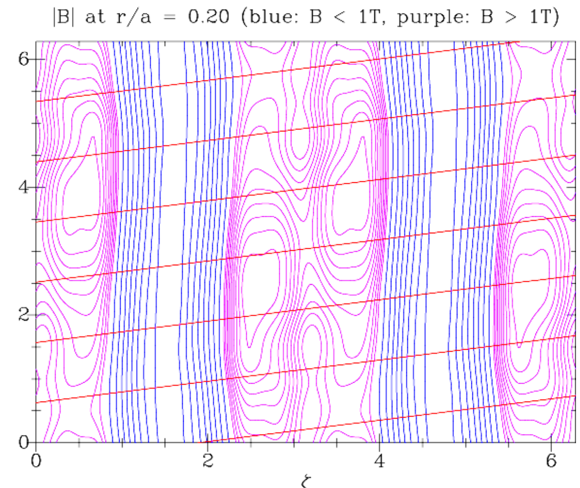
- Using equivalent toroidal current that produces same edge iota
- Limits are not due to MHD instabilities
- high-β is reached with high density (favourable density scaling in W7-AS)
- All **W7-AS** high-β data points beyond operational limits of tokamaks

New designs: Optimize for orbit Confinement

Only 3 strategies: Quasi-symmetric |B| in straight field-line coordinates



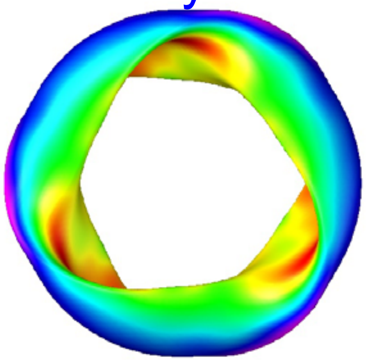




Quasi-Helical

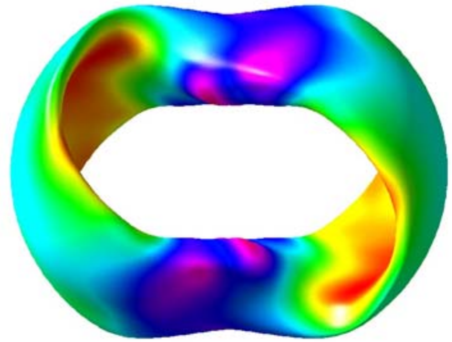
HSX, operating.

Quasi-Axisymmetric



NCSX, construction

Quasi-Poloidal



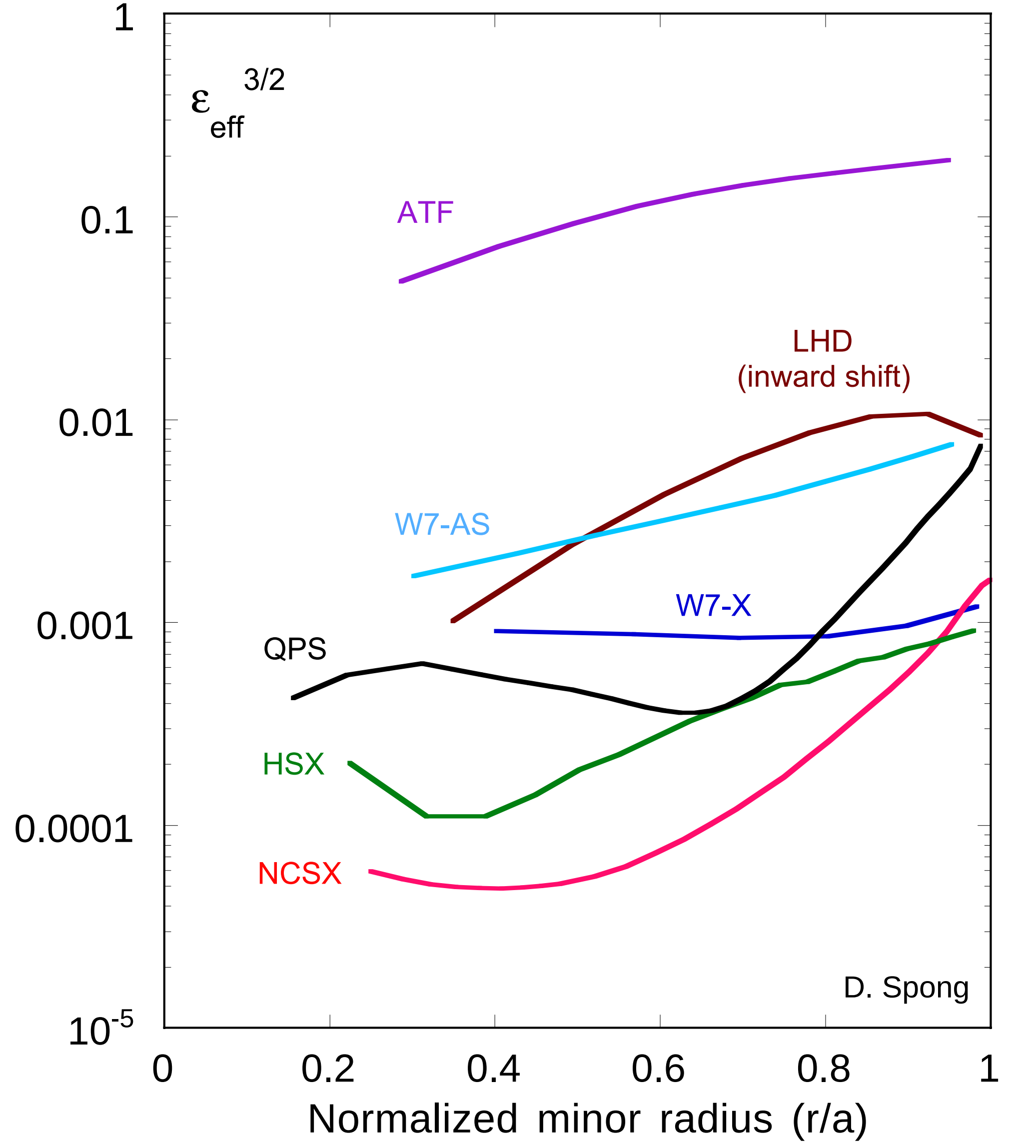
W7-X, construction

QPS, Proposed

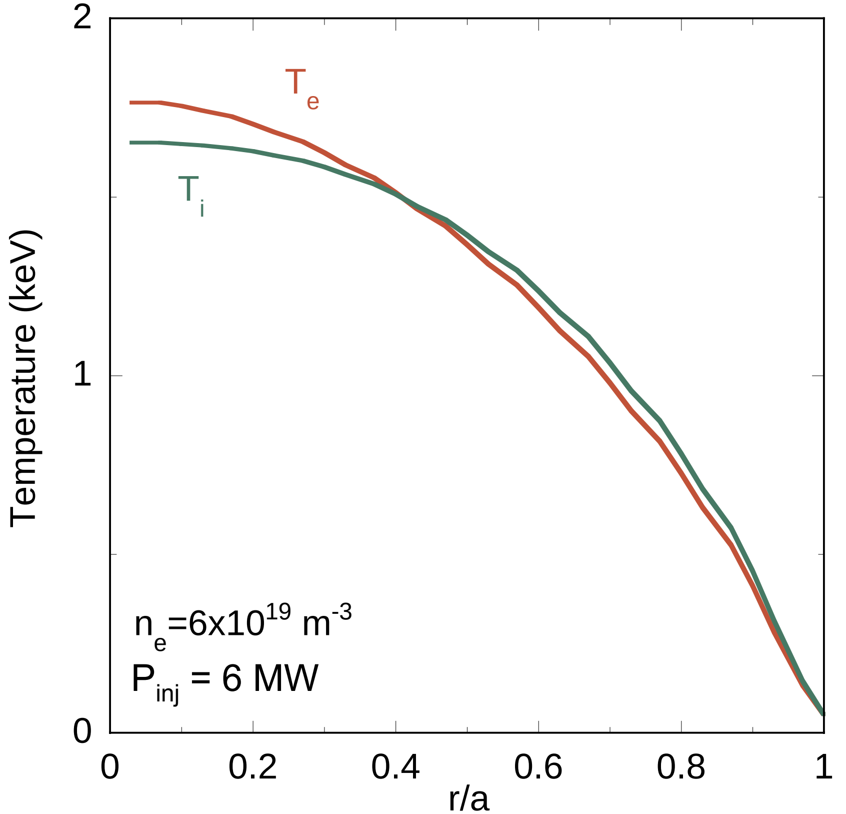
- Reduces ripple-driven cross field transport
- Reduces flow damping in quasi-symmetry direction

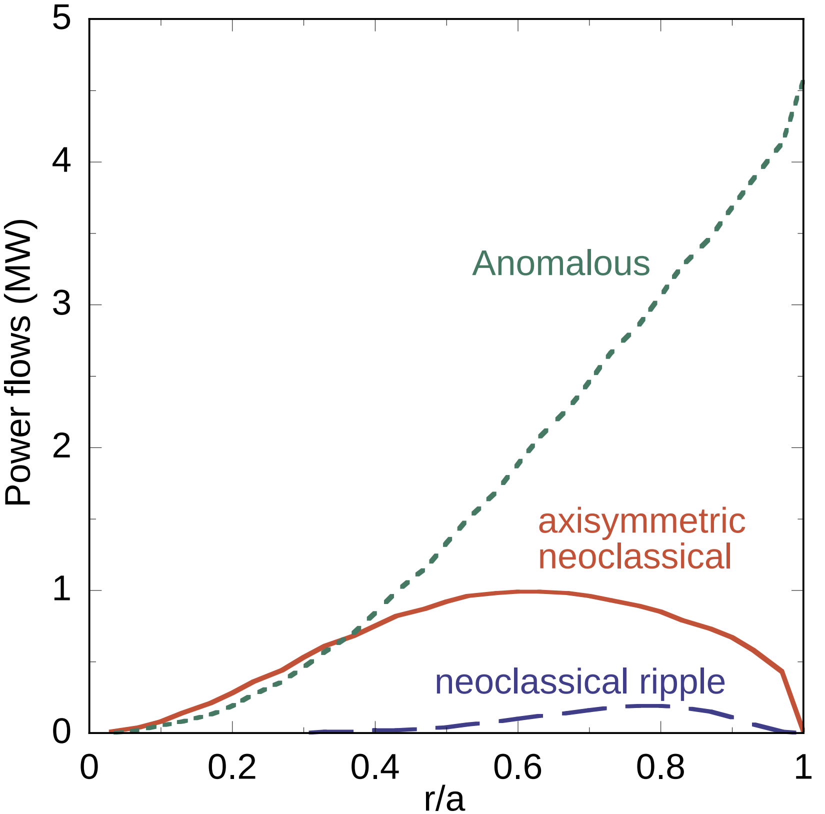
New Experiments: Low Ripple Transport

- $\varepsilon_{eff}^{3/2}$ characterizes collisionless ripple-neoclassical transport
- All new designs numerically optimized to reduce effective ripple
  - W-7X
  - HSX - operating
  - NCSX } Low Aspect Ratio
  - QPS
- Related to improved fast ion confinement.



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





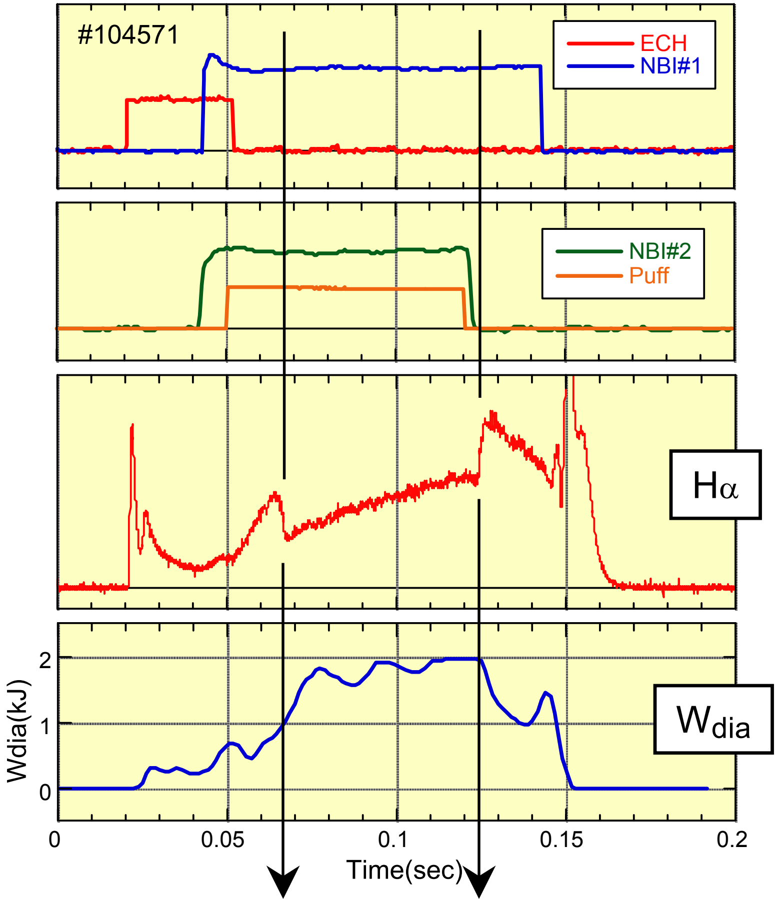
- Helical transport is negligible with self-consistent E_r
- Quasi-axisymmetric

  Collisional-orbit transport is the same as equivalent tokamak
- Calculated: low toroidal flow damping.

  Flow stabilization of turbulence should be similar to tokamaks

Stellarator H-modes and Edge Barriers similar to tokamaks

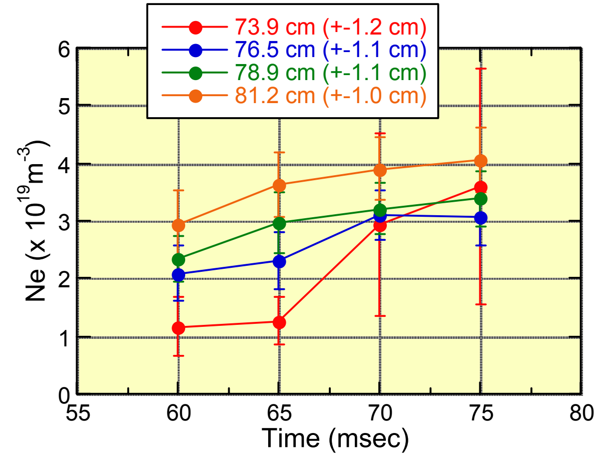
CHS



Transition to ETB

Back Transition

Thomson measurement shows edge density increases at transition



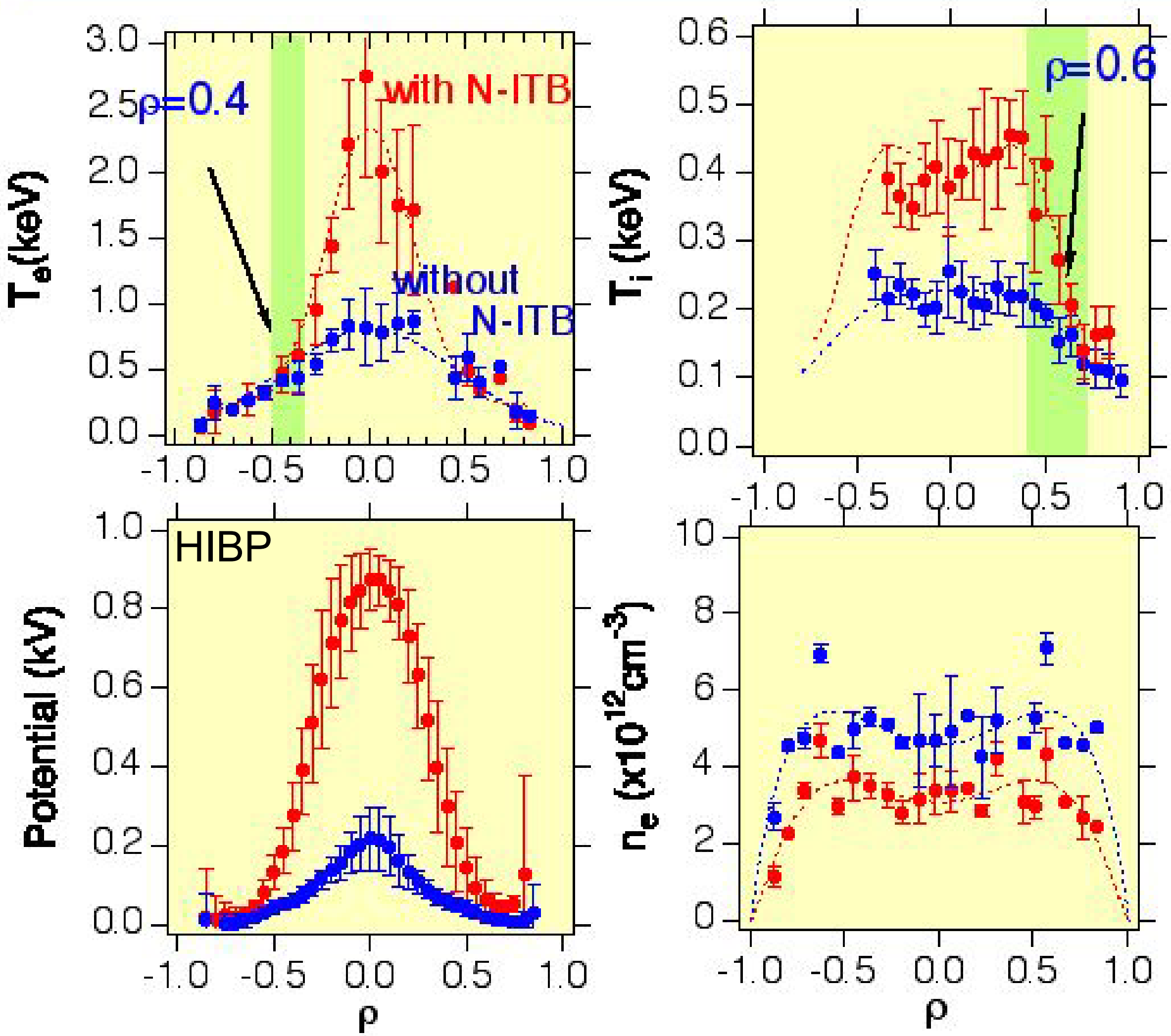
Two NBIs, B = 0.95 T

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

Combined ITBs also Observed



● NBI+ECH with N-ITB
● NBI+ECH without N-ITB

Ion :
The steep gradient increases in the range of ρ~0.4–0.7

Electron :
The electron temperature gradient also increases inside ρ~0.4.

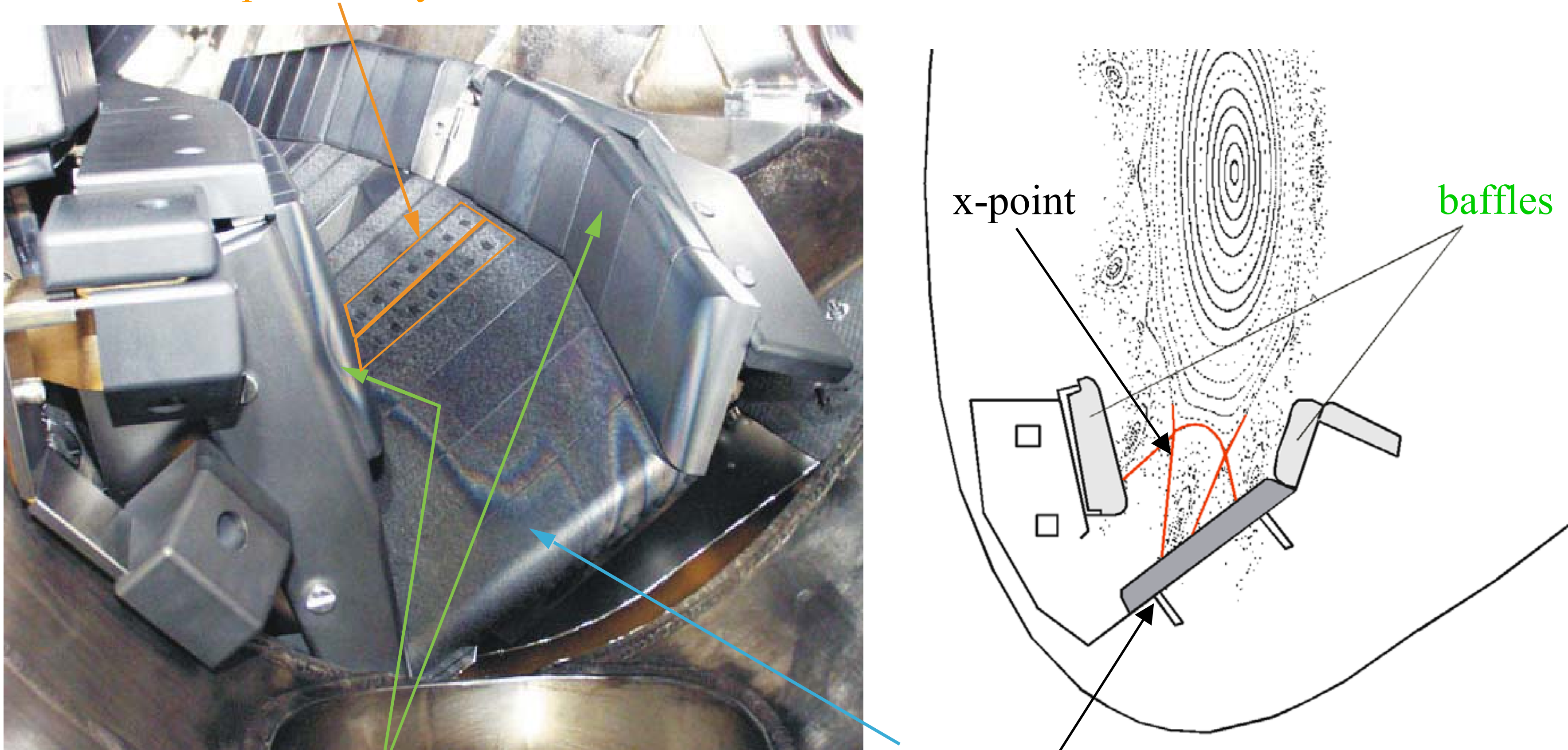
Potential :
The electron root is found inside ρ~0.6 creating the large E_r shear regime.

Fluctuations:
Reduced when ITB present.

- S. Okamura

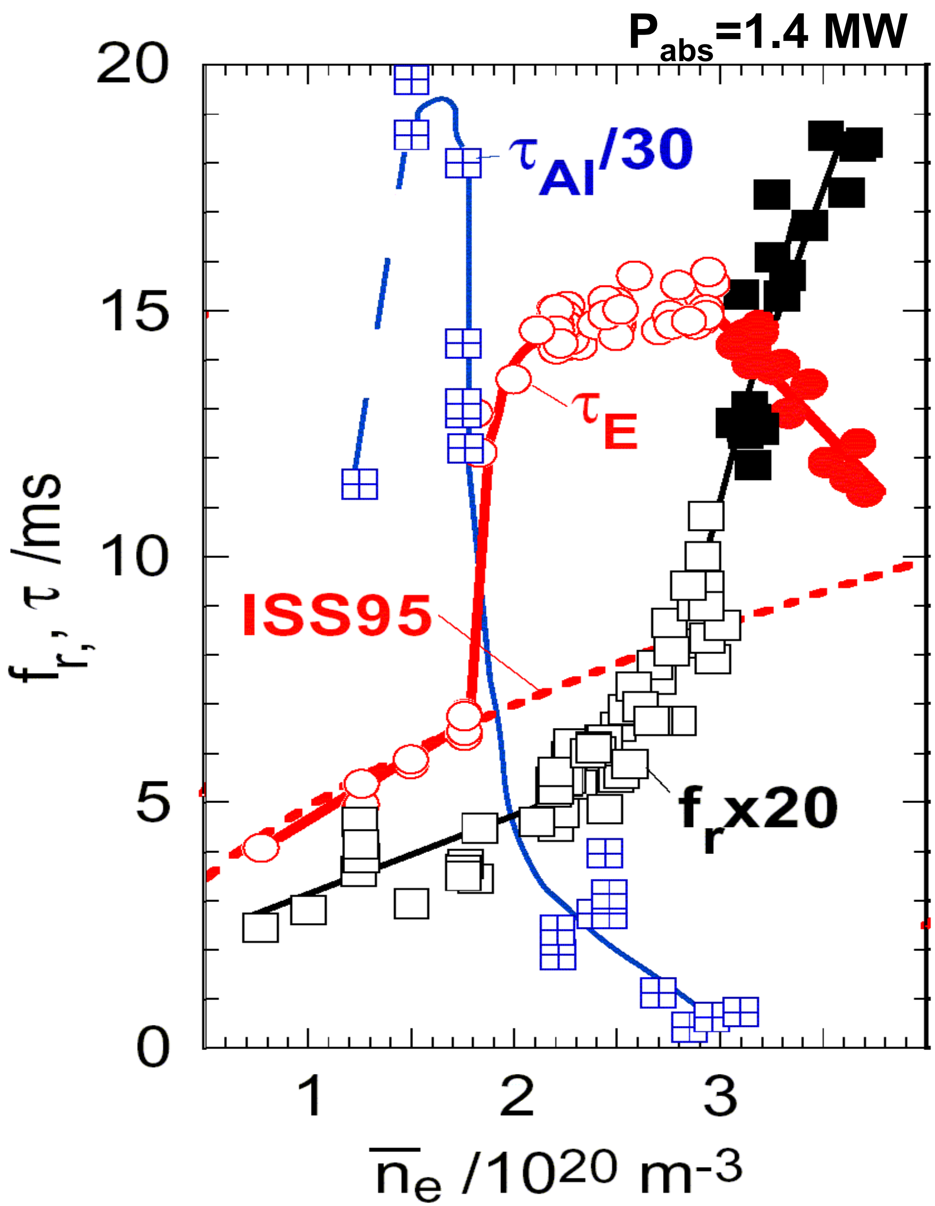
- ITBs observed in both Ion and Electron channels
- Electron-root adds additional mechanism for ITB formation, observed also on LHD, W7AS

W7AS: First Test of Island Divertor Properties



- Natural magnetic islands at the plasma boundary: used to divert magnetic field lines
- Island divertor opened access to long pulse operation at high power, high n_e
- Divertor configurations also being studied on LHD

High Density H-mode in W7AS

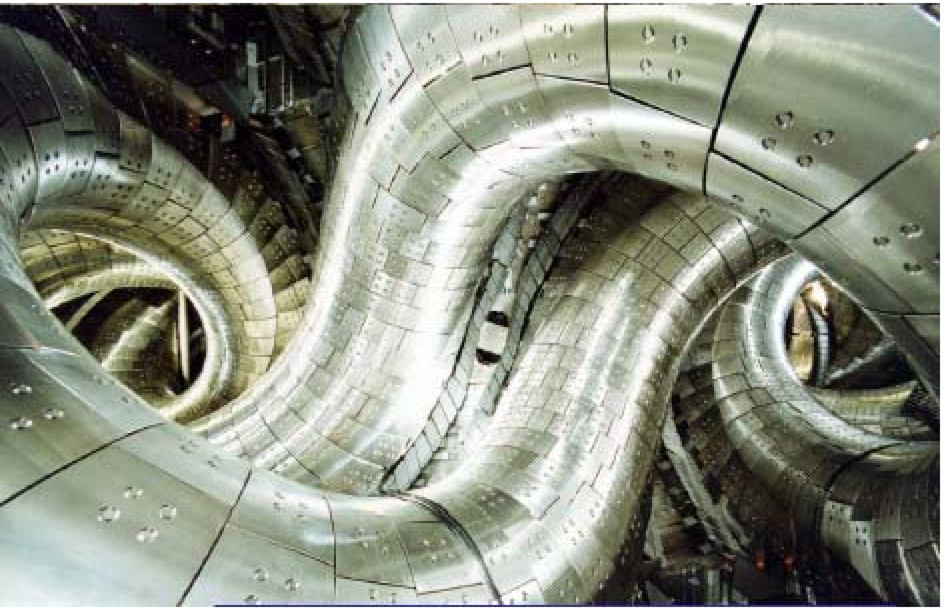


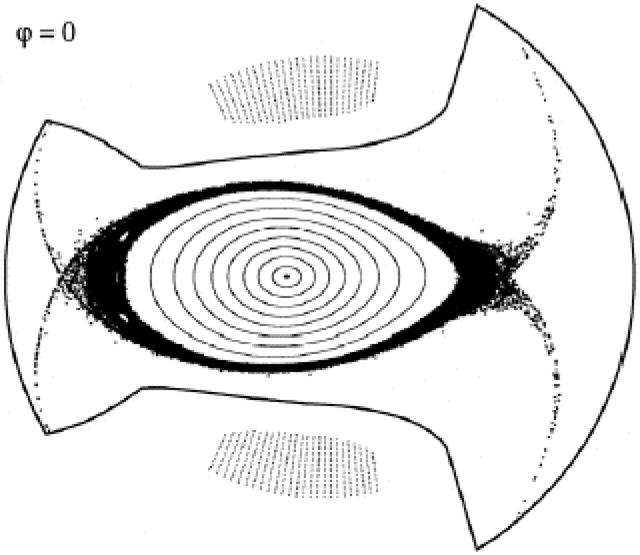
Only observed with divertor edge

Above threshold density:

- τ_E jumps to 2 x τ_E(ISS95)
- τ_{imp} suddenly drops by factor 20-30 to $\tau_{imp} \sim \tau_E$
- Radiated power fraction stable, up to 90 % during detachment
- HDH robust against configuration changes…
- Used for high-β studies

LHD: Helical divertor





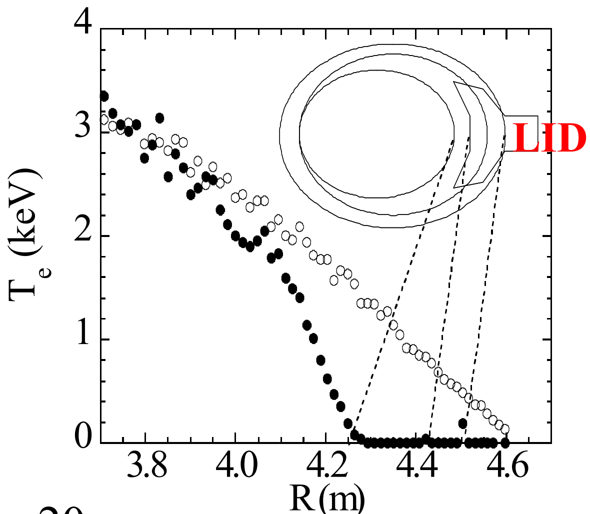
- R = 3.6 – 3.9 m
- minor radius <a> = 0.6m
- B ≤ 3T
- 12 MW NBI, 3MW ICH, 2MW ECH

- Intrinsic helical divertor from coil structure

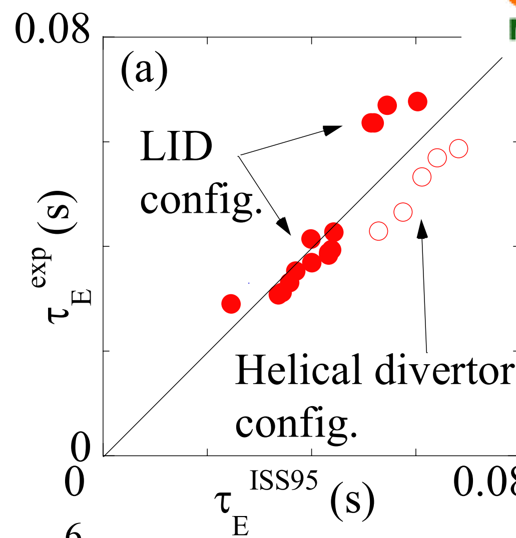
- Can also introduce ‘local island divertor’ and scoop-like divertor structure

LHD LID configuration

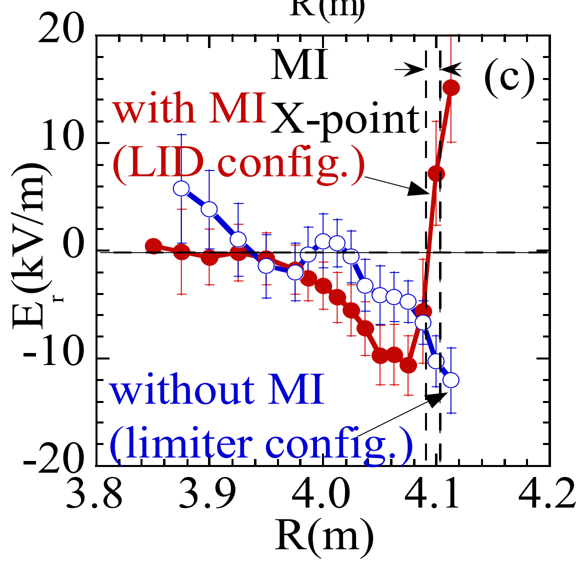
Create sharp edge (large Te gradient)



Confinement improvement

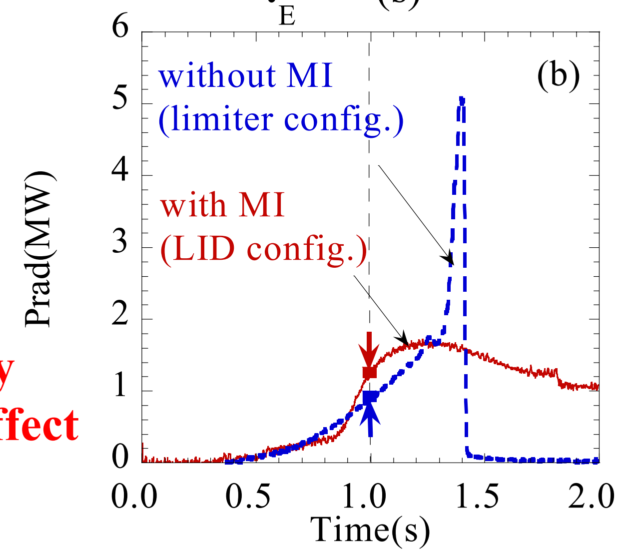


Produce edge Positive E_r



Prevent radiation collapse

by impurity shielding effect



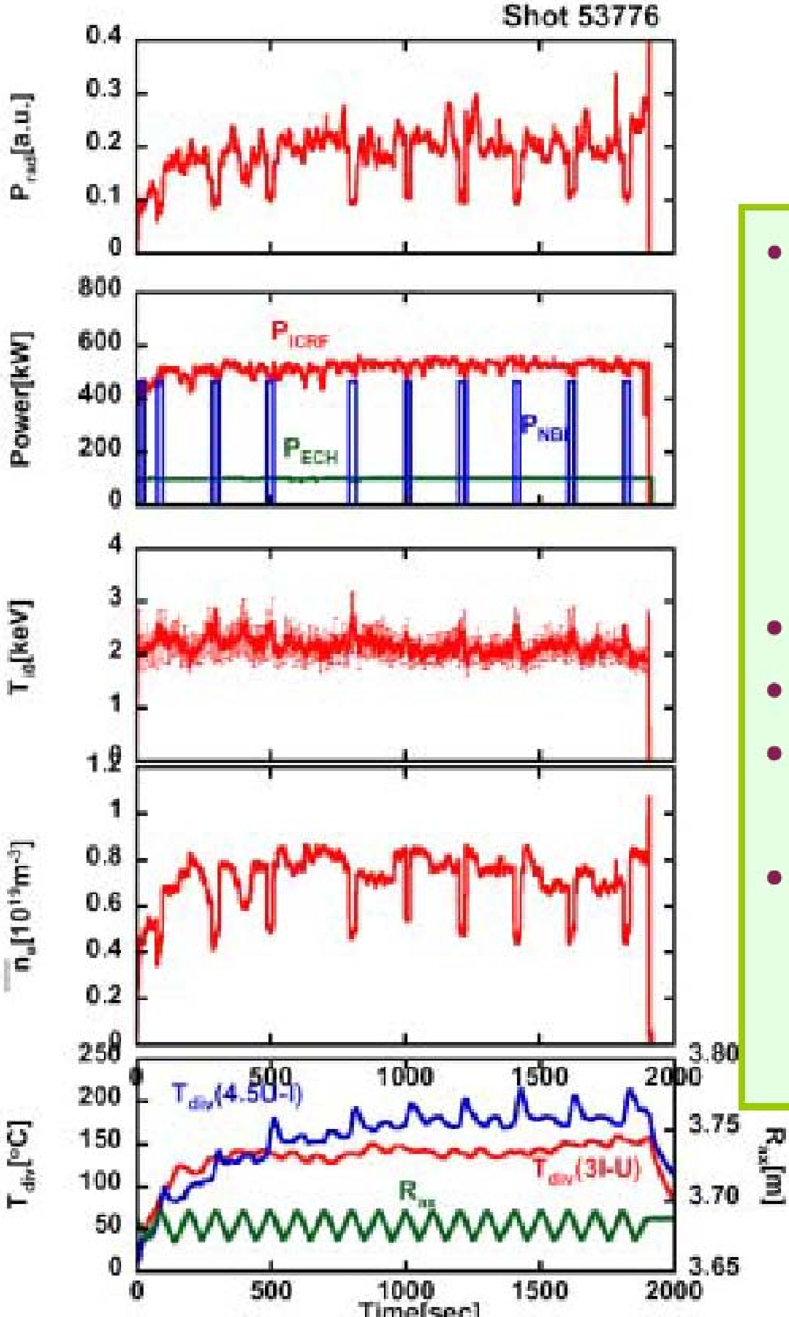
· **Basic function of LID demonstrated**

1) **confinement improvement**

2) **prevent radiation collapse**

Motojima, IAEA 2004

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)

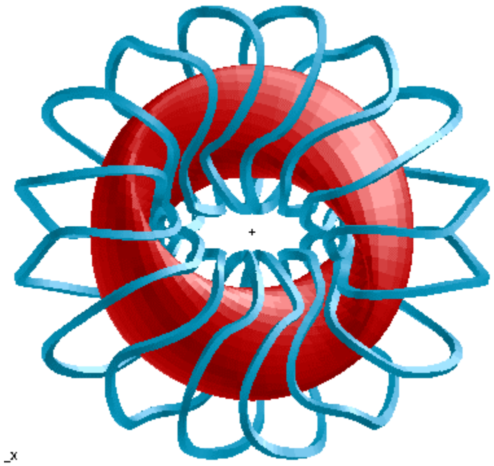
Biggest Engineering Issues: Coils

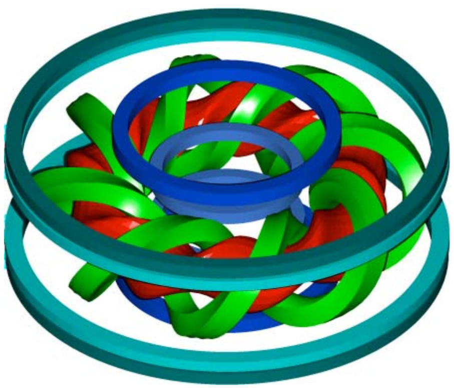
- Stellarators can be made with very simple coils
  - simpler than any tokamak (e.g. CNT)

- Infinite range of coil topologies for making any particular stellarator plasma
  - helical coils
  - modular coils
  - saddle coils

- When optimize plasma shape for transport, stability,… coil shapes become more complex
  - curvature
  - coil-plasma separation
  - coil-coil separation
  - B on coil vs B in plasma

- Modular coils can be used alone
  - no need for PF or TF



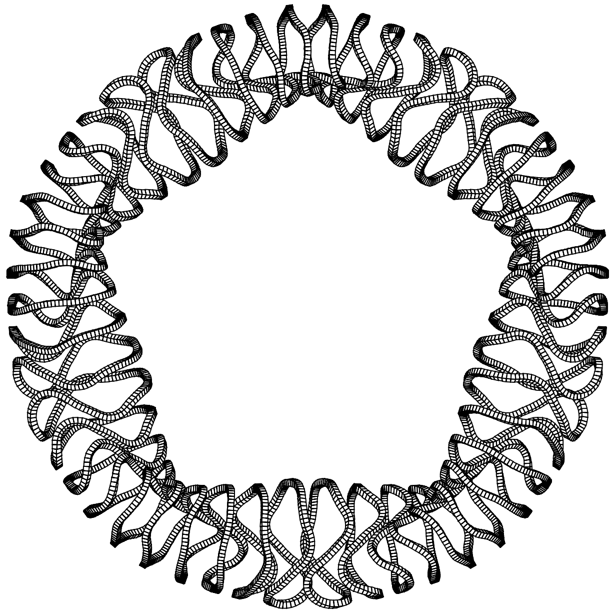


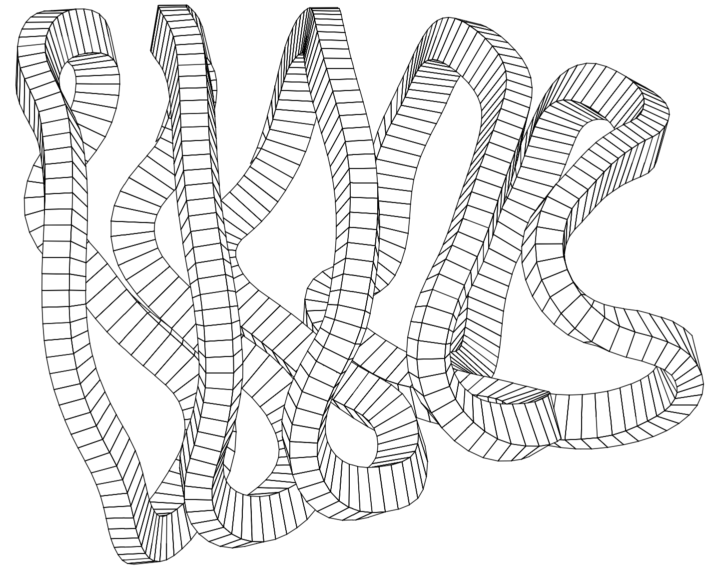
Earlier Reactor Designs were Very Large Motivated US Compact Stellarator Program

HSR
(W7X-like)

R = 22m!
R/<a> = 11
50 coils

ITER fits inside





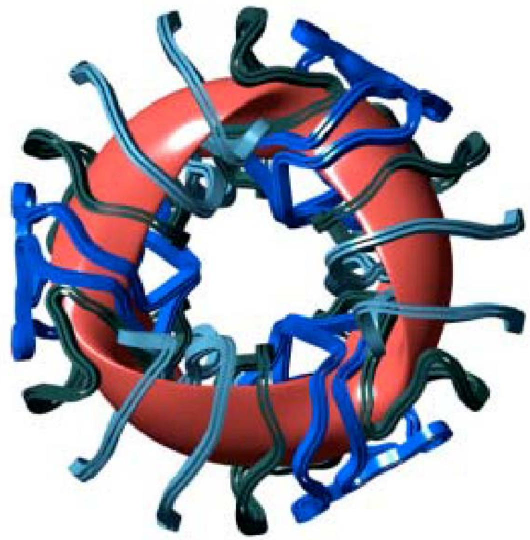
Kisslinger, IAEA 1998

- Large aspect ratio eases maintenance and access
  Reduces wall loading => longer blanket and shield life

- Very high initial capital cost

- Probably would work!

Aries-CS: Examining Compact Alternatives



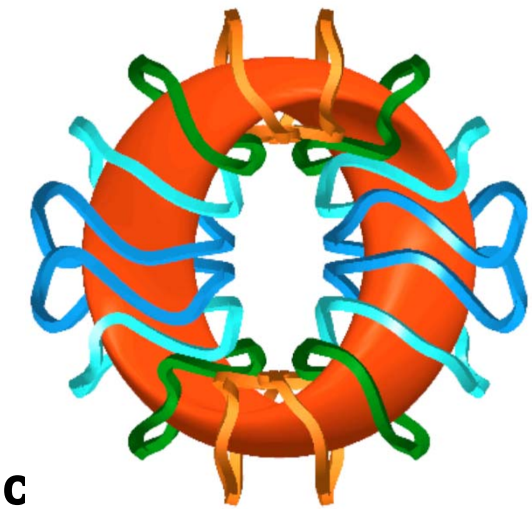
NCSX

port or sector (end) access

MHH2

access through ports

both quasi-axisymmetric



Key Configuration Properties	NCSX-1	NCSX-2	MHH2-8	MHH2-16
Plasma aspect ratio $A_p = \langle R \rangle / \langle a \rangle$	4.50	4.50	2.70	3.75
Wall (plasma) surface area/$\langle R \rangle^2$	11.80	11.95	19.01	13.37
Min. plasma-coil separation ratio $\langle R \rangle / \Delta_{min}$	5.90	6.88	4.91	5.52
Min. coil-coil separation ratio $\langle R \rangle / (c\text{-}c)_{min}$	10.07	9.38	7.63	13.27
Total coil length/$\langle R \rangle$	89.7	88.3	44.1	64.6
$B_{max,coil} / \langle B_{axis} \rangle$ for 0.4-m x 0.4-m coil pack	2.10	1.84	3.88	2.77

0-D Determination of Main Reactor Parameters

- **Fix maximum neutron wall loading $p_{n,wall}$ at 5 MW/m^2**
  - **peaking factor =1.5 → $\langle p_{n,wall} \rangle$ = 3.3 MW/m^2**
- **Maximize $\langle p_{wall} \rangle$ subject to $j_{SC}(B_{max})$ and radial build constraints**
  - **blanket, shield, structure, vacuum vessel ~ wall area ~ $1/\langle p_{n,wall} \rangle$**
  - **volume of coils ~ $L_{coil} I_{coil} / j_{coil}$ ~ $\langle R \rangle^{1.2}$ ~ $1/\langle p_{n,wall} \rangle^{0.6}$**
  - **blanket replacement independent of $\langle p_{n,wall} \rangle$**
- **$\langle p_{wall} \rangle$ = 3.3 MW/m^2 → wall area = 480 m^2 for P_{fusion} = 2 GW**

  $\langle R \rangle$ = 6.22 m for NCSX-1 vs. $\langle R \rangle$ = 14 m for SPPS
- **Chose $\langle \; \rangle$ = 6%:** no reliable instability limit, high equilibrium limit

  $\langle B_{axis} \rangle$ = 5.80 T for NCSX-1
- **B_{max} on coil depends on plasma-coil spacing & coil cross section**
- **$\langle R \rangle$ and $\langle B_{axis} \rangle$ for the other cases are limited by the radial build and coil constraints to $\langle p_{n,wall} \rangle$ = 2.13–2.67 MW/m^2**

Lyon, TOFE 2004

0-D Study Gives Main Reactor Parameters

	NCSX-1	NCSX-2	MHH2-8	MHH2-16
$\langle p_{n,wall} \rangle$ (MW/m^2)	3.33	2.67	2.13	2.4
$\langle R \rangle$ (m)	6.22	6.93	6.19	6.93
$\langle a \rangle$ (m)	1.38	1.54	2.29	1.85
$\langle B_{axis} \rangle$ (T)	6.48	5.98	5.04	5.46
B_{max} (T)	12.65	10.9	14.9	15.2
j_{coil} (MA/m^2)	114	119	93	93
k_{max}	3.30	5.0	2.78	1.87
coil width (m)	0.598	0.719	0.791	0.502
coil depth (m)	0.181	0.144	0.286	0.268
radial gap (m)	0.026	0.012	0.007	0.005
Coil volume (m^3)	60.3	63.4	61.4	60.3
Wall area (m^2)	480	600	750	667

- **Successful in reducing reactor size ($\langle R \rangle$) by factor ~ 2!**
- **Wall (blanket, shield, structure, vacuum vessel) area smallest for NCSX-1 ==> choose for more detailed study**

Lyon, TOFE 2004

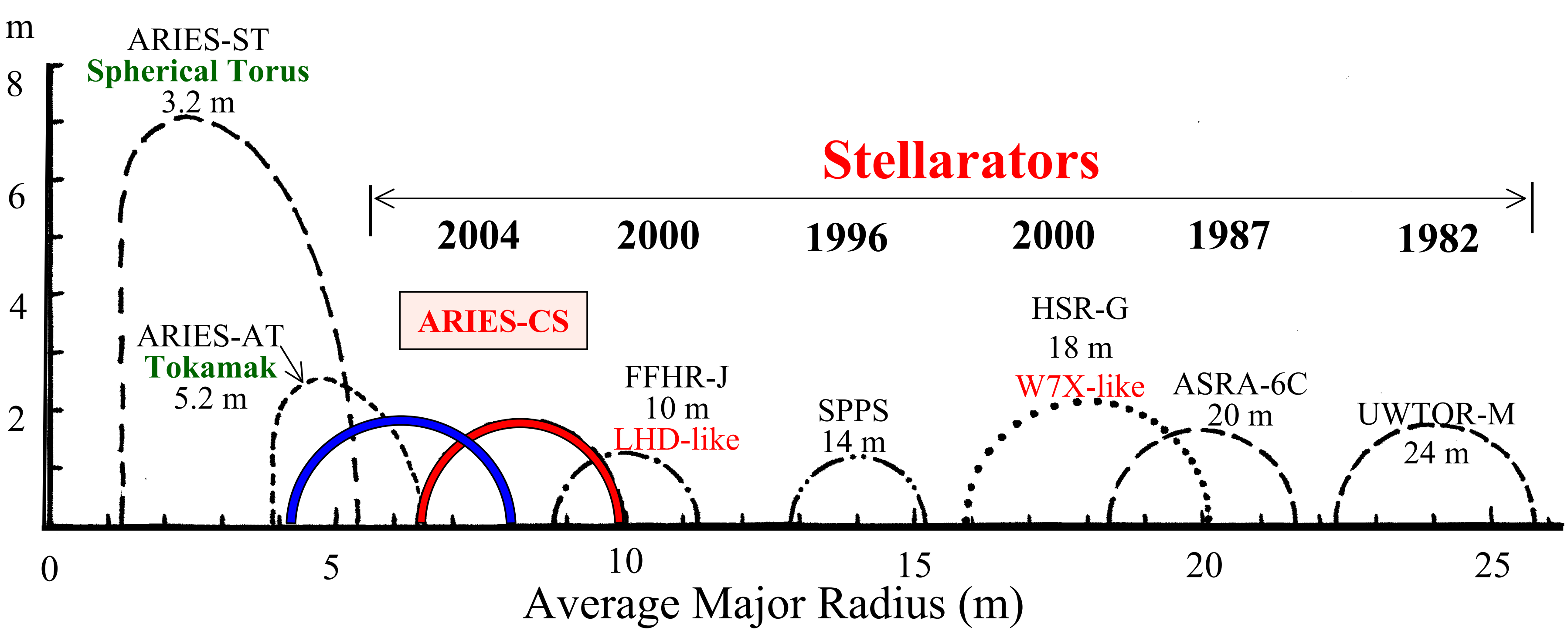
1-D Power Balance Gives Plasma Parameters

	NCSX-1	NCSX-2	MHH2-8	MHH2-16
$\langle R \rangle$ (m)	6.22	6.93	6.19	6.93
$\langle a \rangle$ (m)	1.38	1.54	2.29	1.85
$\langle B_{axis} \rangle$ (T)	6.48	5.98	5.04	5.46
H-ISS95	4.15	4.20	3.75	4.10
n (10^{20} m^{-3})	3.51	2.89	2.05	2.43
f_{DT}	0.841	0.837	0.837	0.839
f_{He}	0.049	0.051	0.051	0.050
T (keV)	9.52	9.89	9.92	9.74
, (%)	6.09	6.12	6.13	6.09

- **ISS-95 confinement improvement factor of 3.75 to 4.2 is required; present stellarator experiments have up to 2.5**
- **ISS-2004 scaling indicates $_{eff}^{-0.4}$ improvement, so compact stellarators with very low $_{eff}$ should have high H-ISS values**

Lyon, TOFE 2004

Optimized Engineering and Physics Design Leads to Compactness of ARIES-CS



Major radius more than halved by advanced physics and technology, dropping from 24 m for UWTOR-M to 6-8 m for ARIES-CS and approaching R of advanced tokamaks.

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, β, A
  - First results from HSX: Quasi-symmetry matters! Flow damping reduced.
  - W7-X and NCSX under construction. Optimized for β and confinement.
  - QPS: proposed for construction. Quasi-poloidal at low aspect ratio.