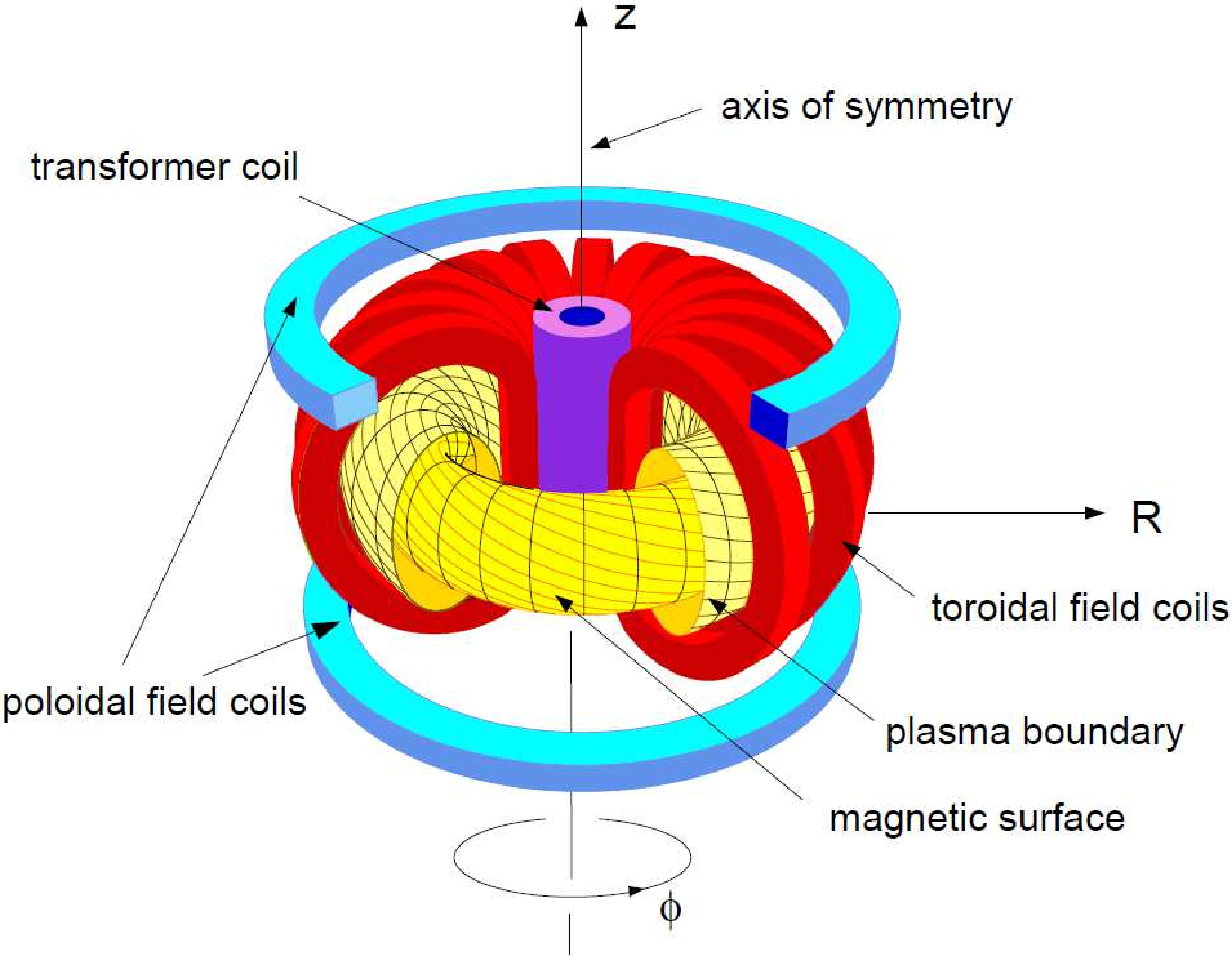
Stellarators

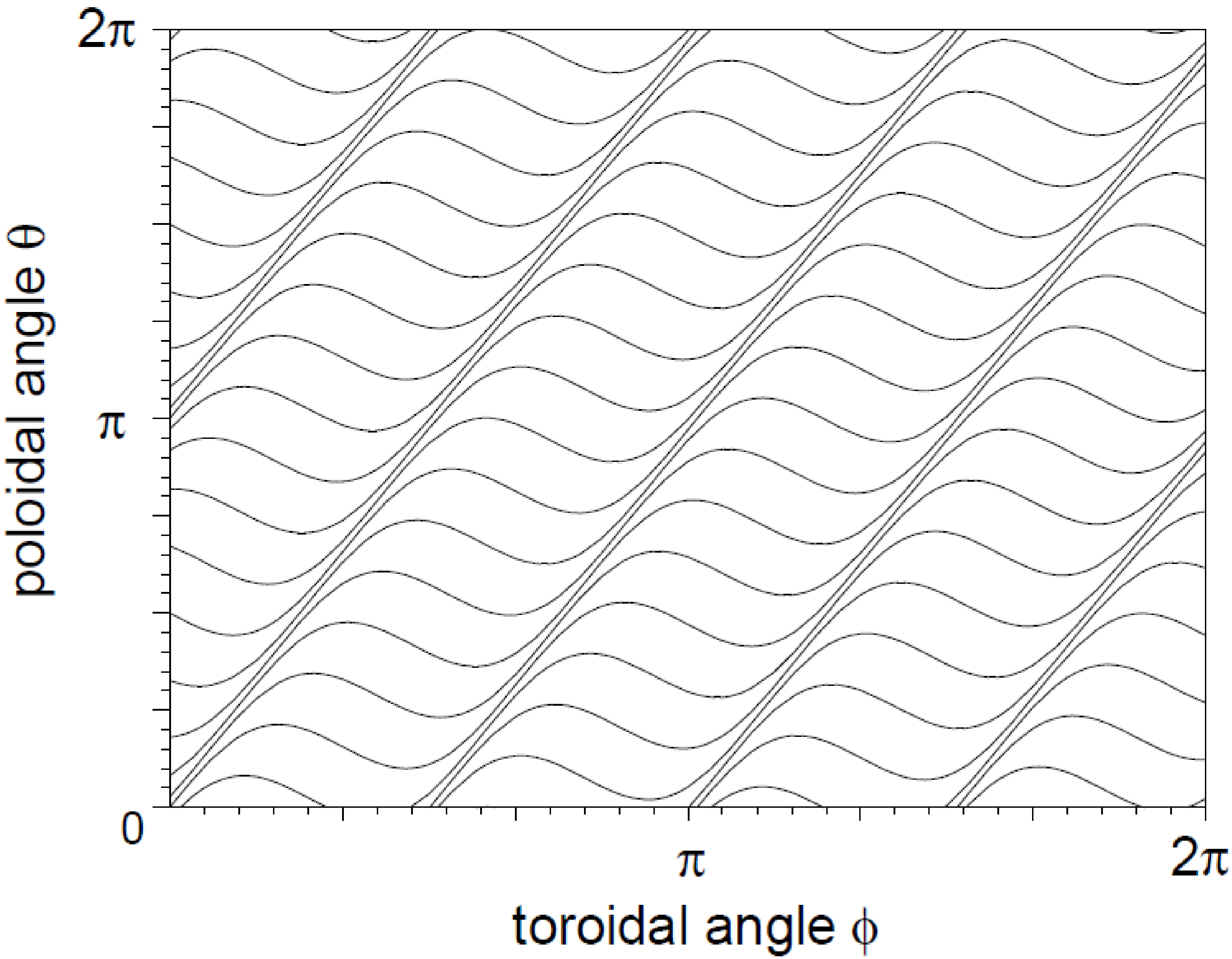
R. Wolf,

adapted by F. Warmer and H. Zohm

Tokamak: closed flux surfaces due to plasma current

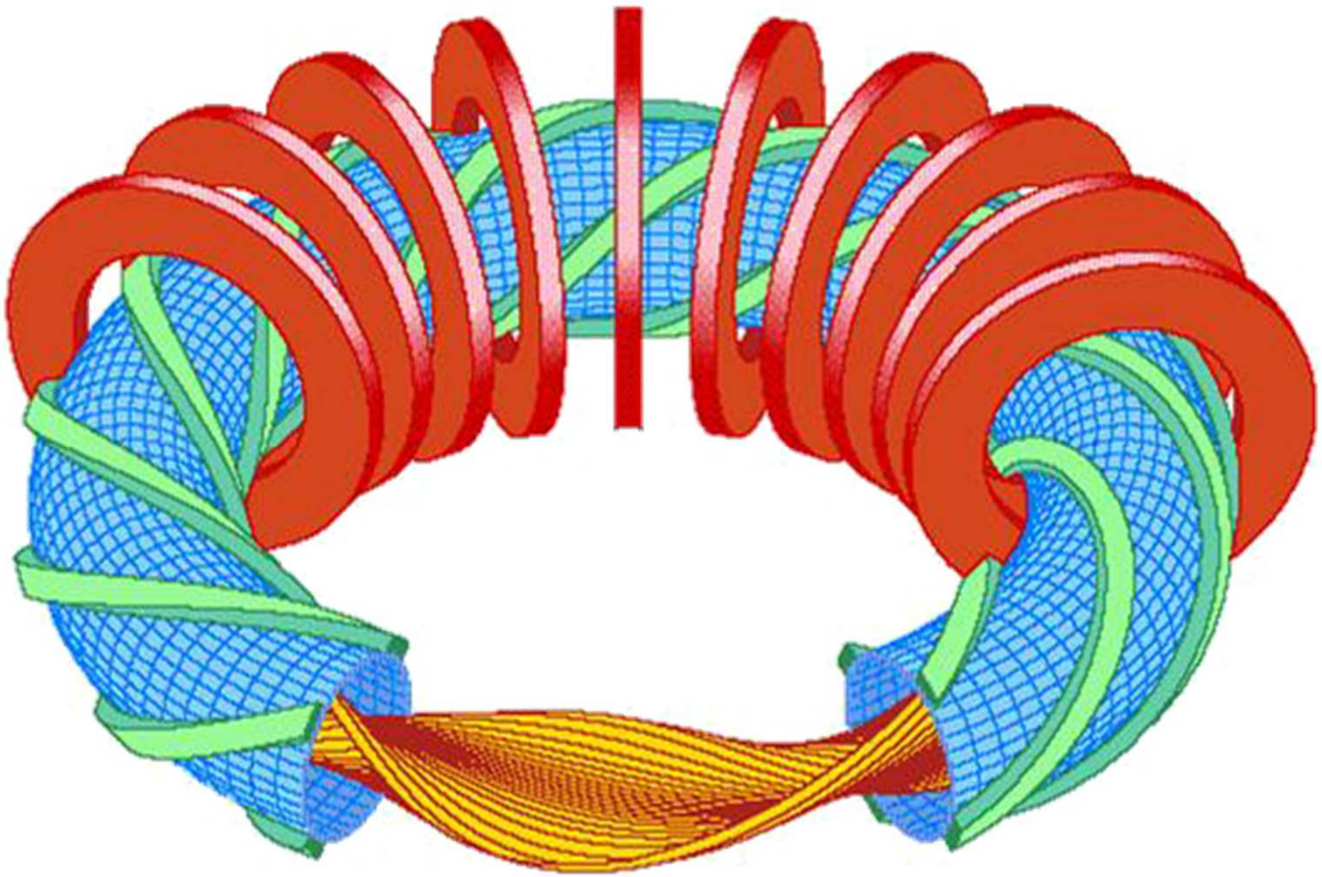


Stellarator: closed flux surfaces without a plasma current



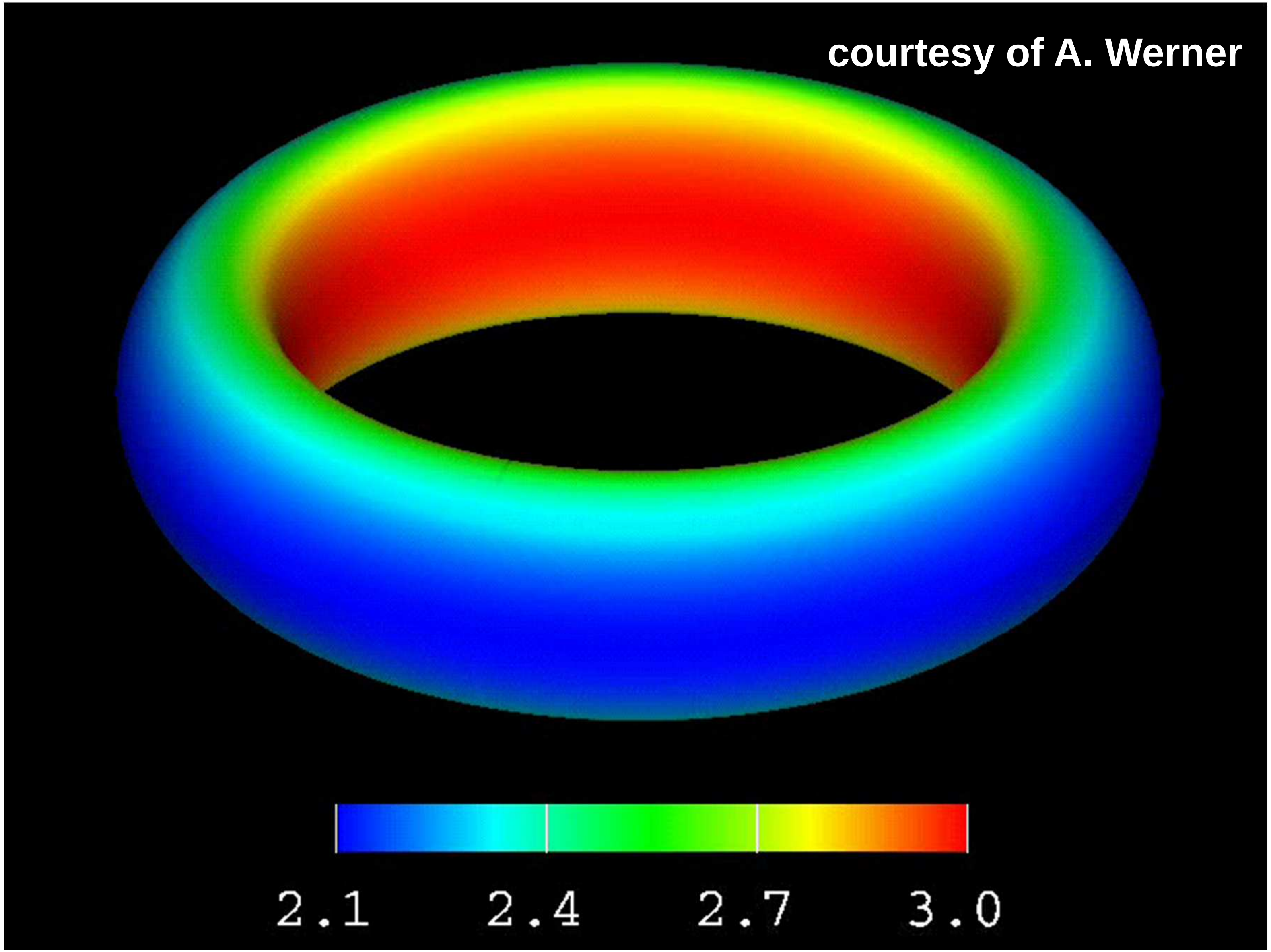
$$\int \mathbf{B} \cdot d\mathbf{s} = \mu_0 I_p = 0$$

Stellarator: closed flux surfaces without a plasma current



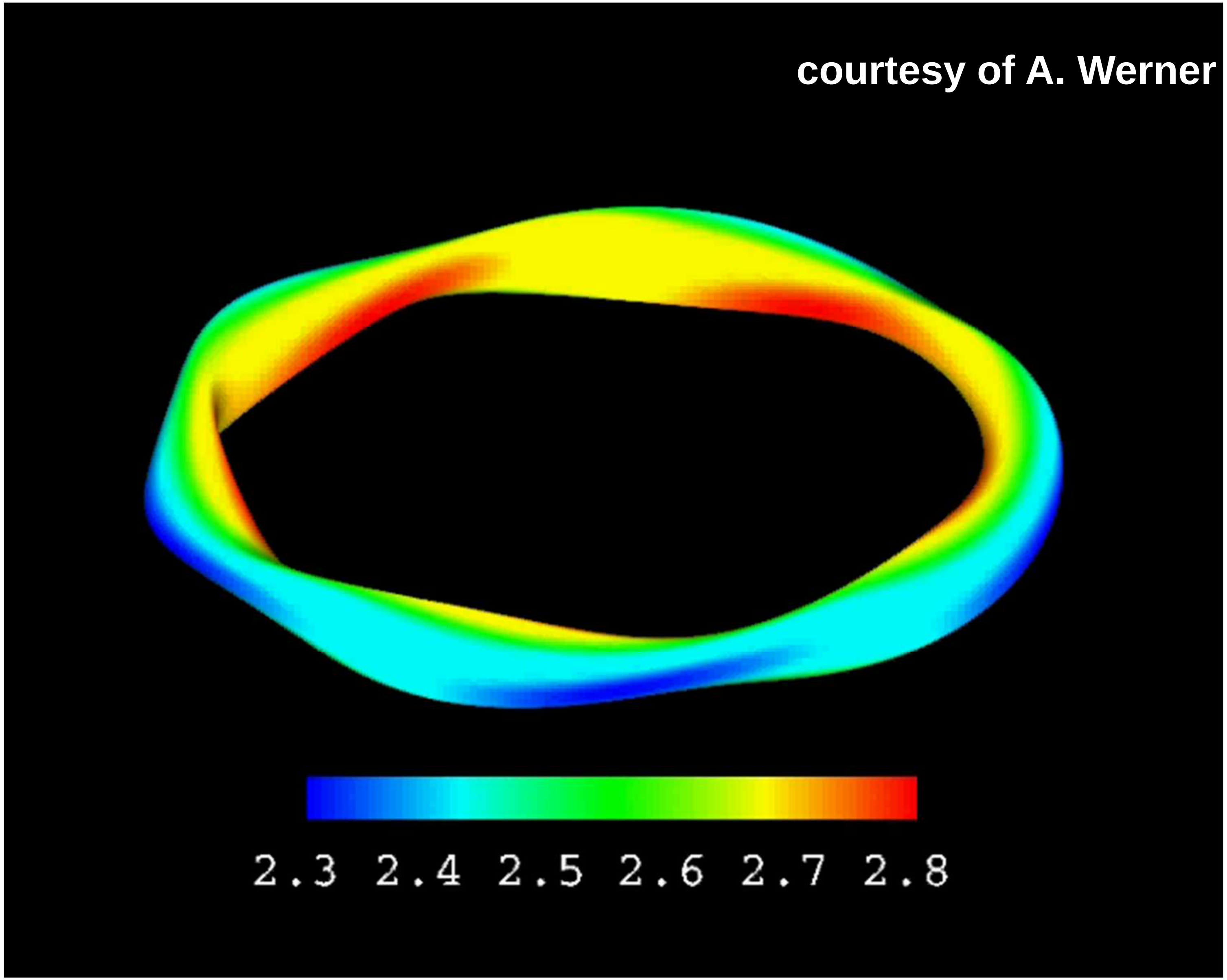
$\int \mathbf{B} \cdot d\mathbf{s} = \mu_0 I_p = 0$ ‚stellarator windings‘: $I(\theta) = I_0 \sin(\ell\theta - M\phi)$

Burning fusion plasma requires drift optimization



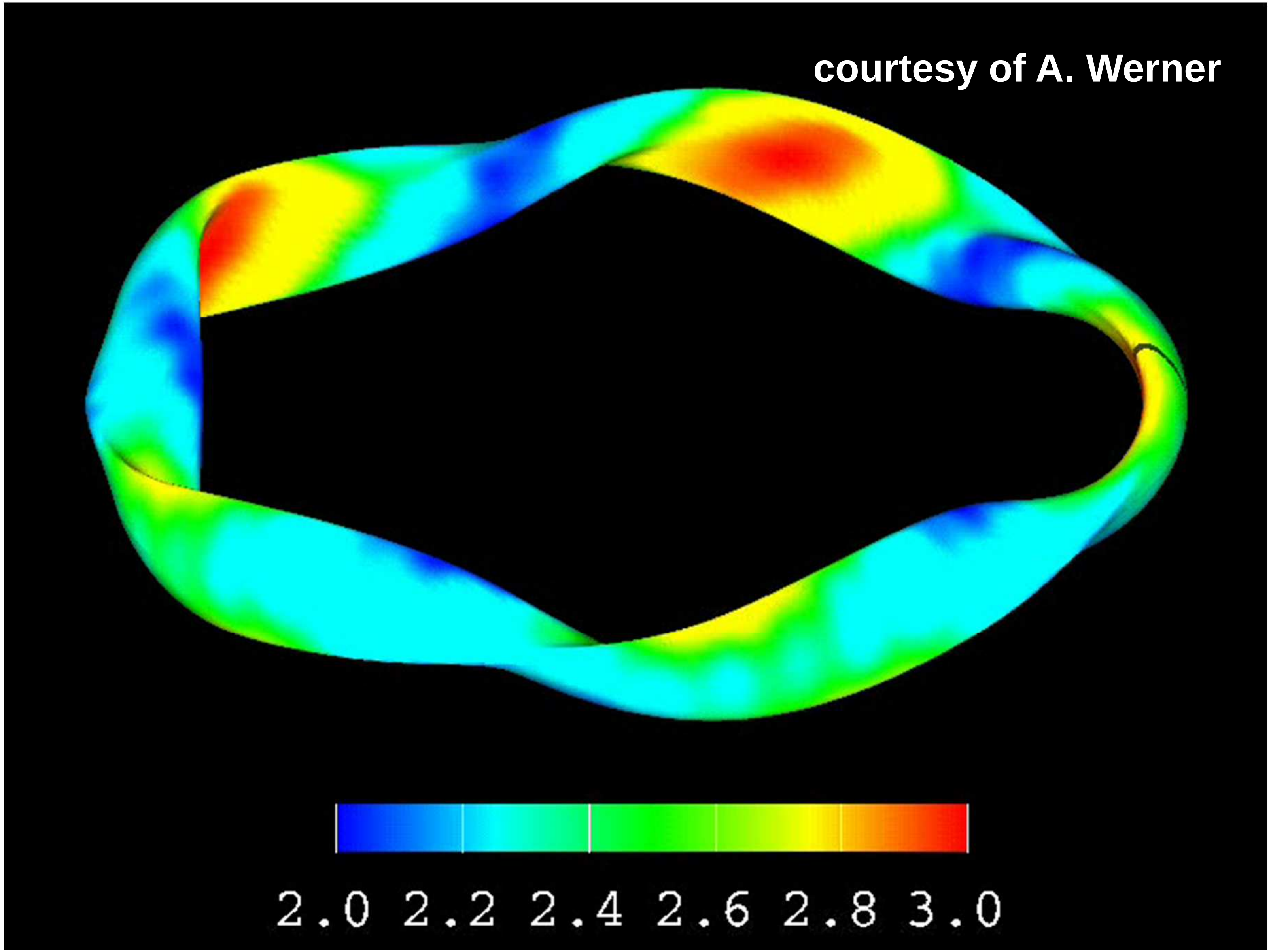
Confinement of fast particles in Tokamaks garantueed due to axisymmetry

Burning fusion plasma requires drift optimization



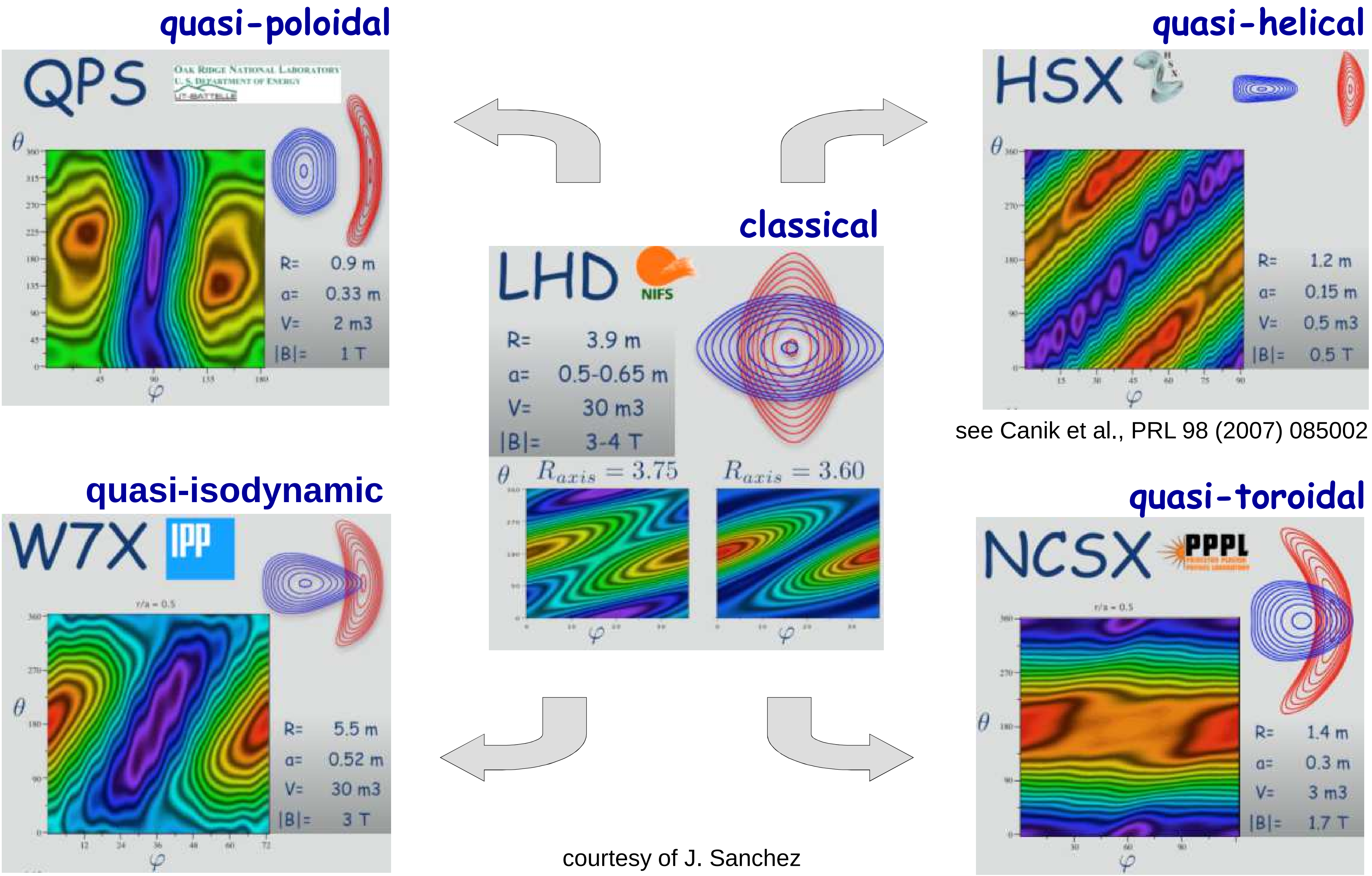
Classical stellarator exhibits bad confinement for fast particles (1/ν regime)

Burning fusion plasma requires drift optimization



Optimised stellarator confines fast particles due to poloidal drift of trapped orbits

Quasi-symmetries (just as a reminder)



see Canik et al., PRL 98 (2007) 085002

courtesy of J. Sanchez

The Large Helical Device (Toki, Japan)

LHD
R= 3.5-4.1 m, a= 0.6 m -> V= 28 m^3
(= twice the volume of AUG) superconducting coils

Helical coils and 3D shape of plasma

- → Good access to the plasma
- → Mechanical stresses can be treated well
- → Large superconducting helical coils
- → Basis for a Japanese reactor study

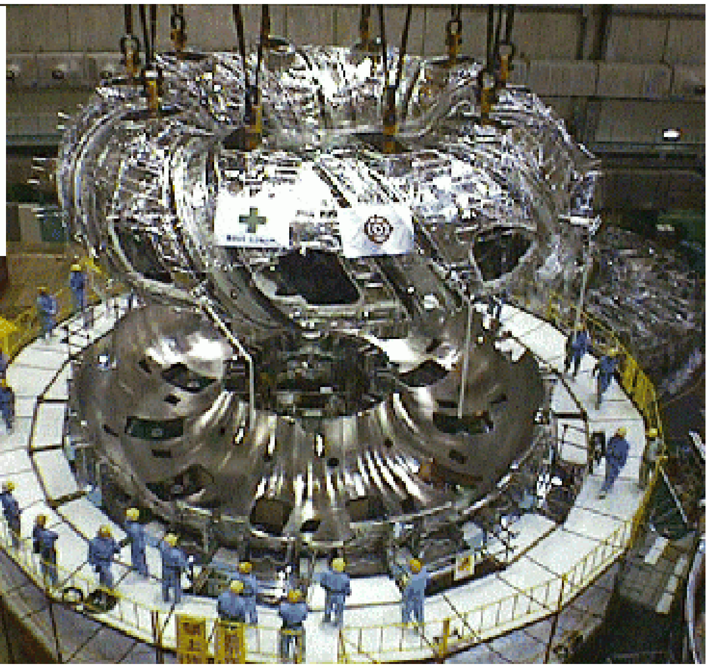
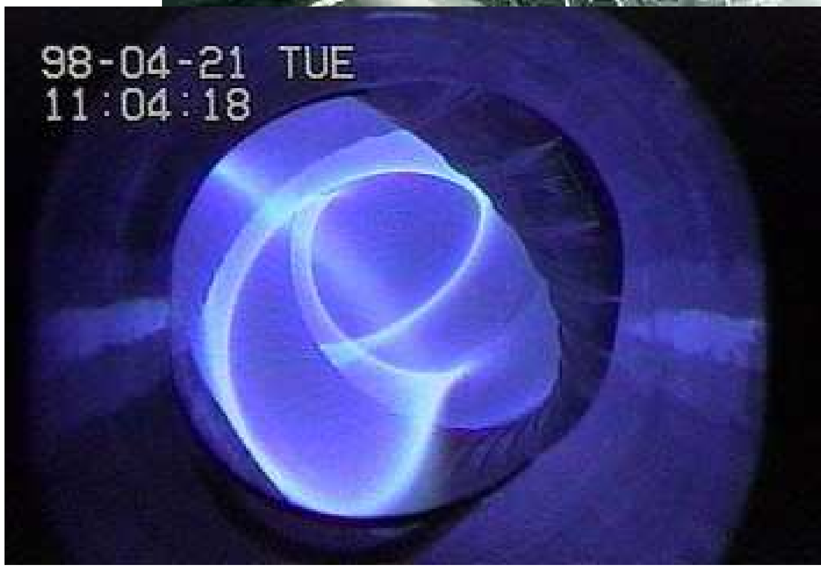
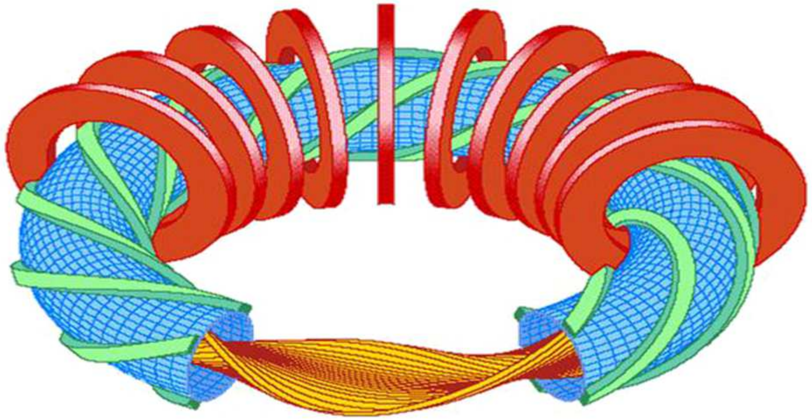


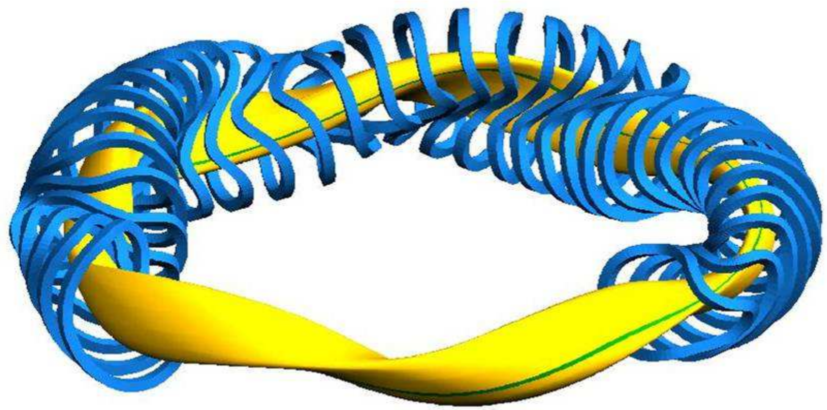
Fig. 2. The helical coil and can assembly is lowered onto the lower support structure on the machine.



The modular stellarator

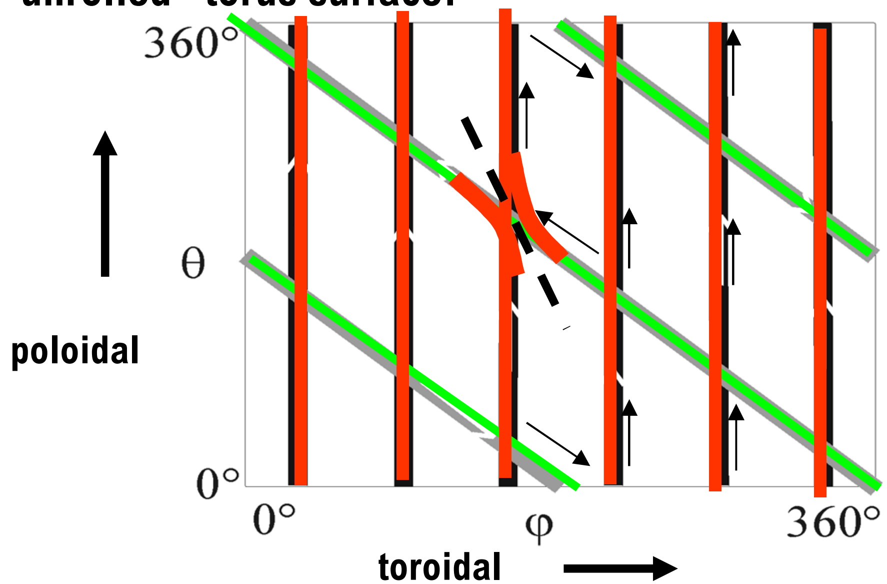


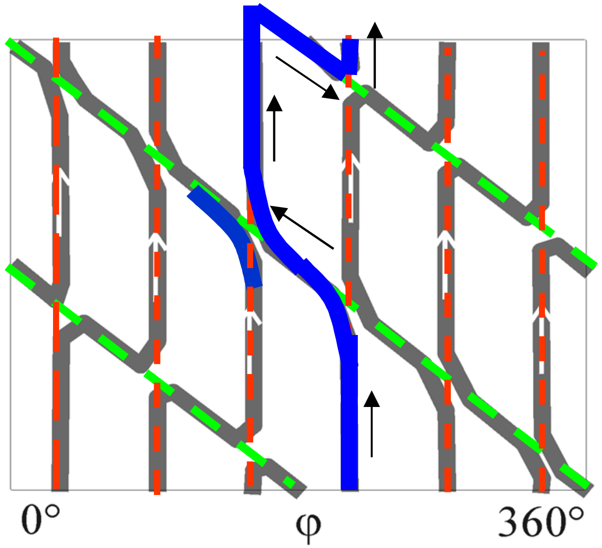
Toroidal field coils Helical field coils



Modular coils

"unrolled" torus surface:



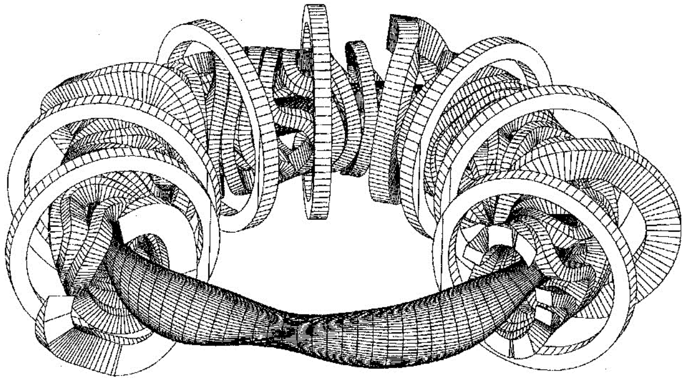


→ **No huge helical coils** / mechanical forces can be handled

→ **Magnetic fields can be tailored** by external current distribution

Modular stellarators

First modular concept by Rehker and Wobig 1972



Wendelstein 7-AS (Garching, Germany)

- First stellarator with modular coils
- Partially optimized w.r.t reduced equilibrium currents
- Predecessor of Wendelstein 7-X
- R = 2m, a $\leq$ 0,18m , V = 1 m^3, B = 2,5 T
- Shut down 2002

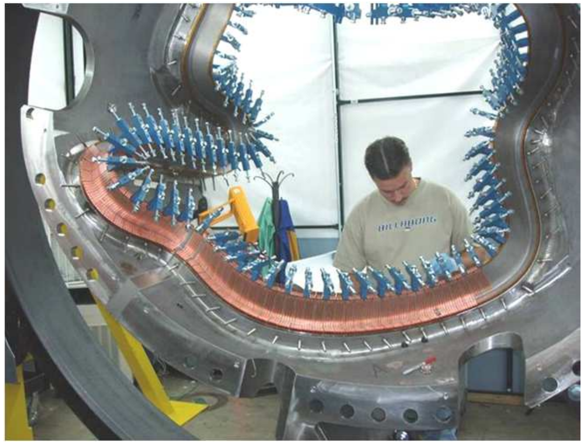
HSX (U Wisconsin, USA)

- Quasi-helical stellarator
- R = 1,2m, a = 0,15m , V = 0,44 m^3, B = 1,37 T

www.hsx.wisc.edu

NCSX (PPPL, Princeton, USA, mothballed)

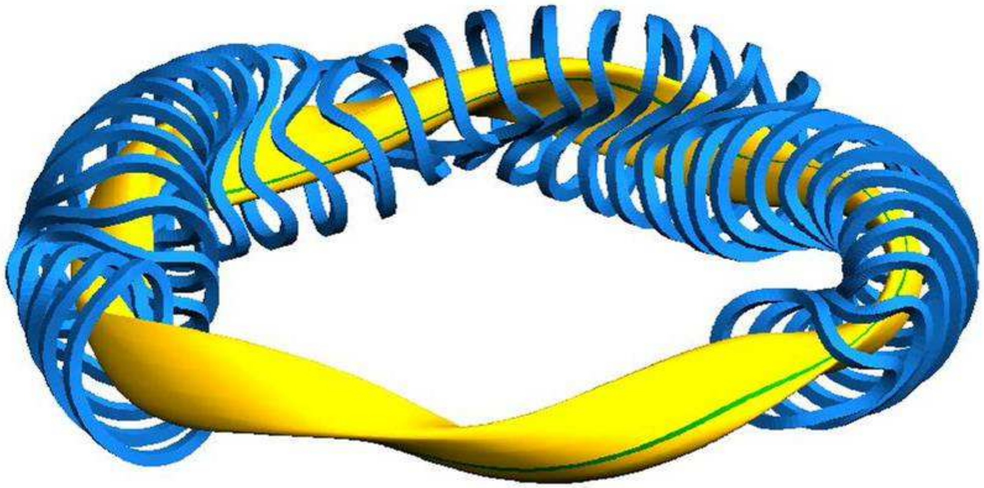
- Quasi-axissymmetric stellarator
- R = 1,42m, a = 0,33m ,
  V = 3 m^3, B = 1,2 – 2 T



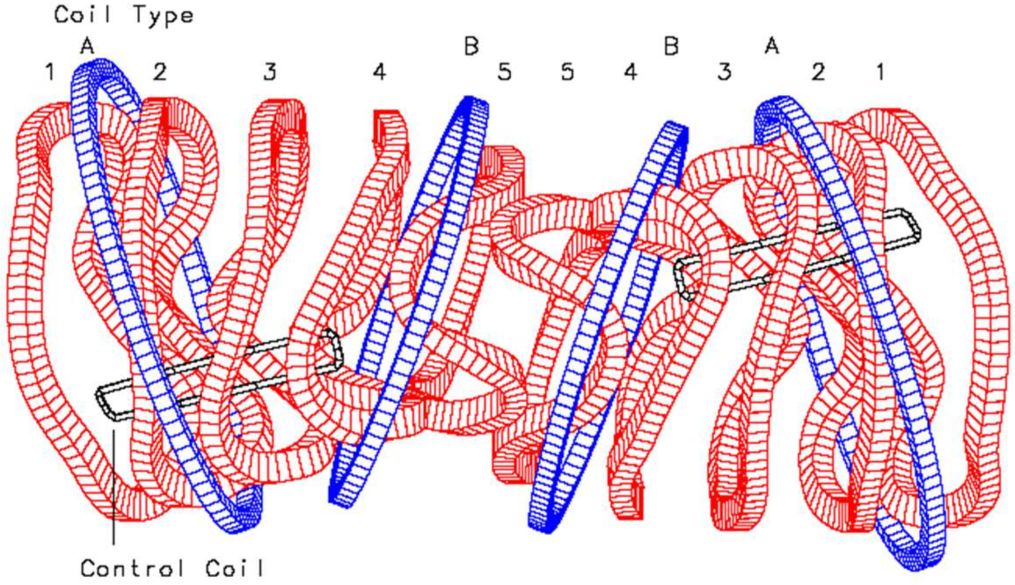
Modular stellarators

Wendelstein 7-X (Greifswald, Germany)

- **First "fully" optimized stellarator**
- **R = 5,5m, a = 0,55m , V = 30 m^3, B = 3 T**
- **Completion of assembly 2014, start of operation 2015**

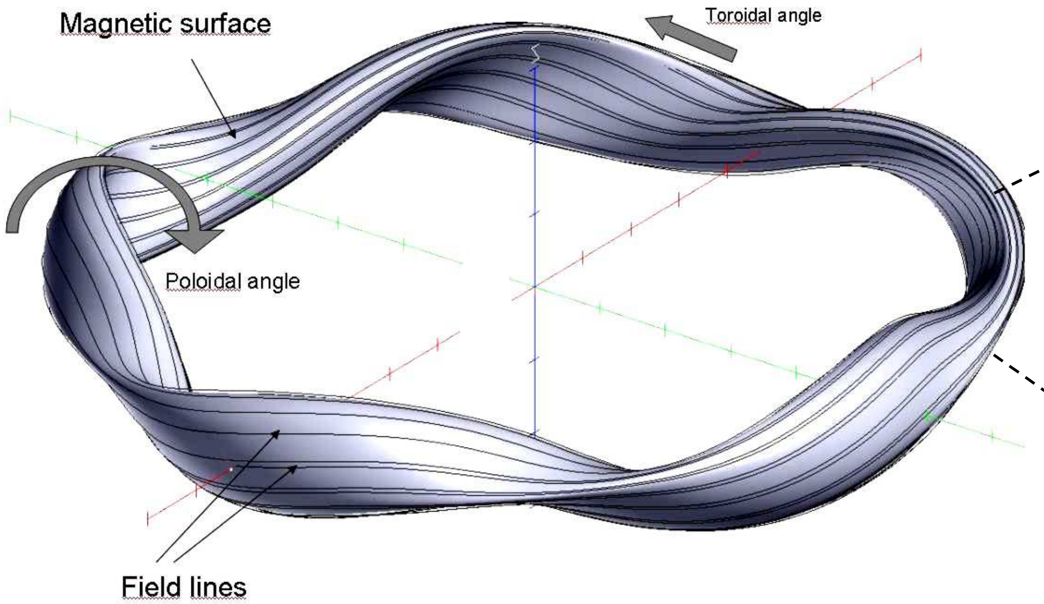


One module of the W7-X coil system

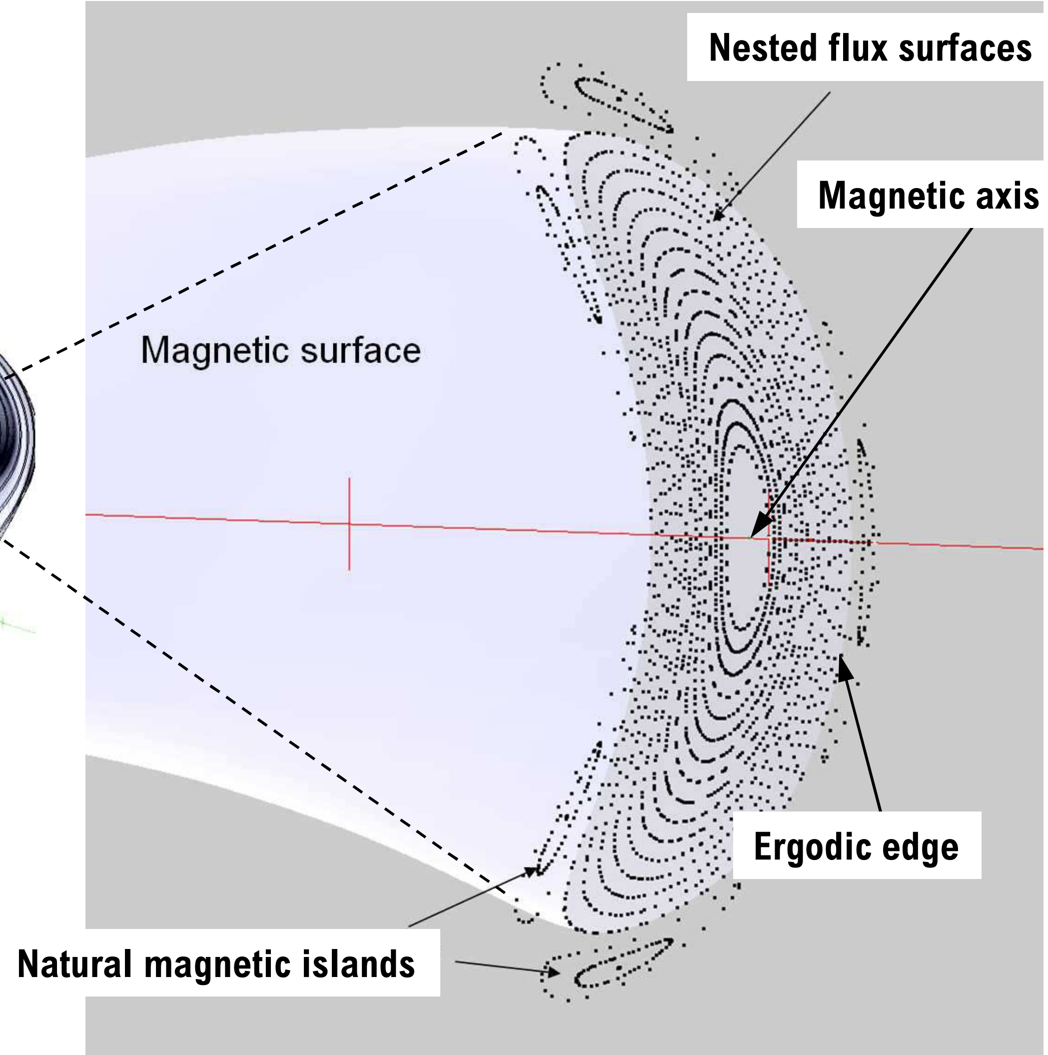


Flux surfaces

In a stellarator the measurement of the vacuum field gives direct information about the confinement properties (rotational transform ι)



Poincare plot



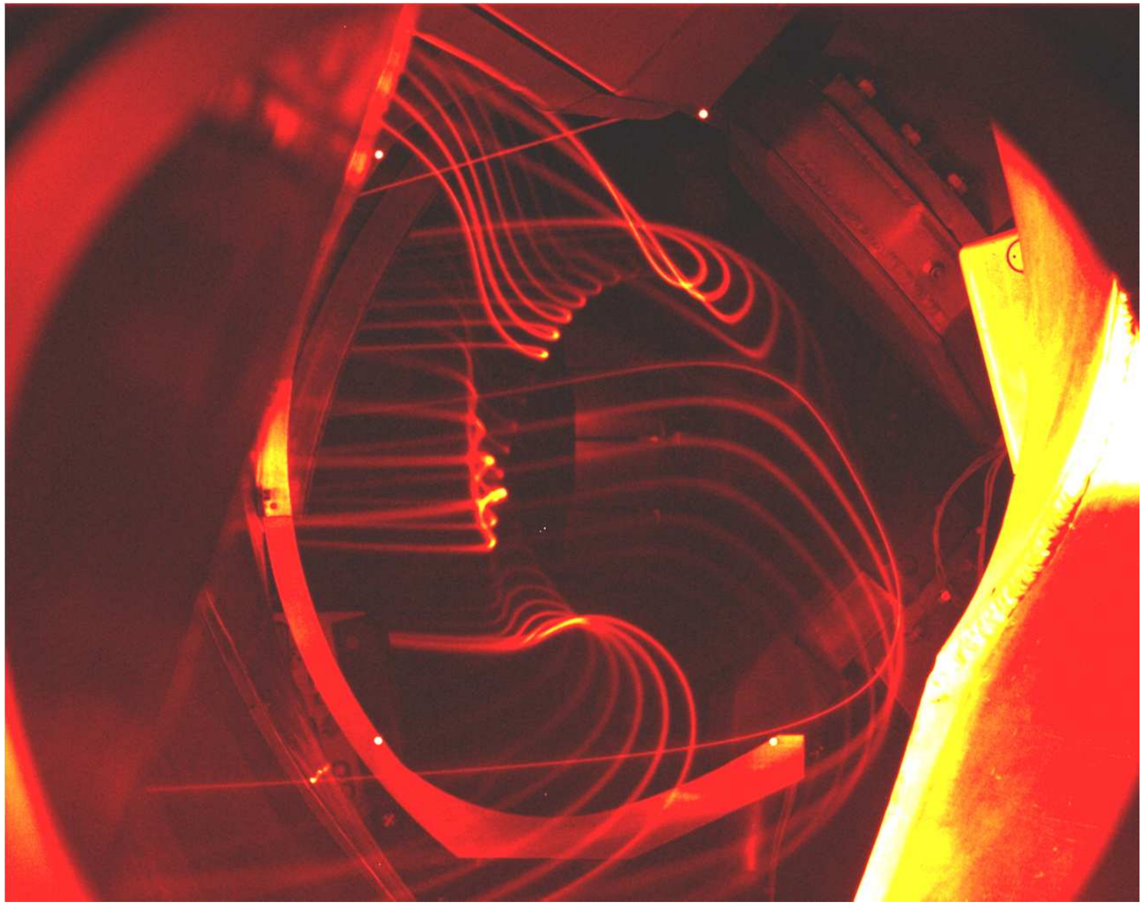
Rotational transform $\iota = R \langle B_\theta \rangle / r_{eff} \langle B_\phi \rangle$
(*local* pitch angle may vary strongly on flux surface)

- **Field lines close only at <u>rational values</u> of *m* toroidal and *n* poloidal transits $\iota/2\pi = m / n$**
- **Due to *m*-fold symmetry <u>natural magnetic islands</u> exist, breaking linear stellarator symmetry**

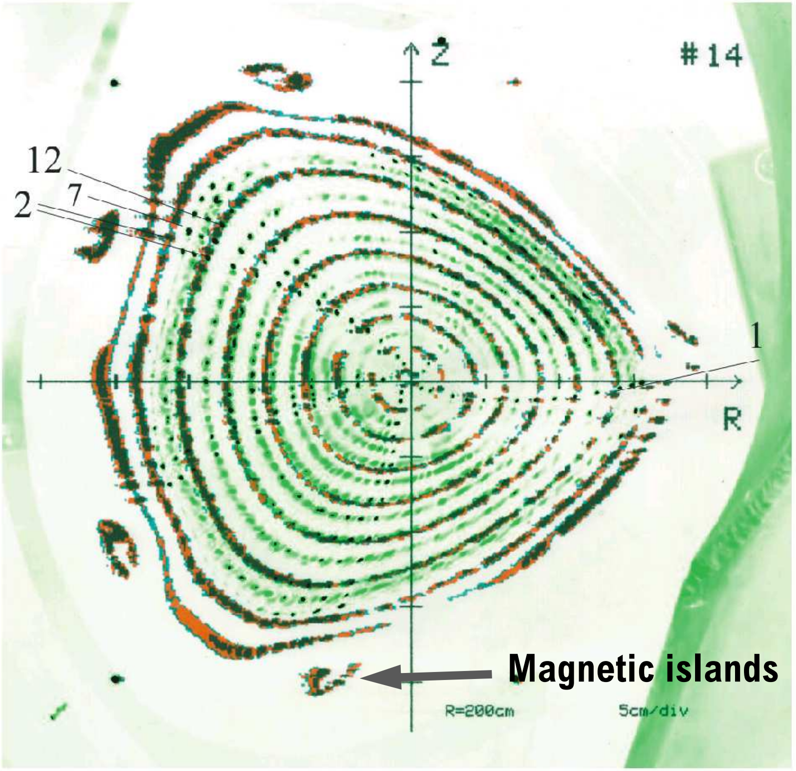
Visualizing field lines and flux surfaces

… in a tokamak they exist by symmetry considerations

Visualizing a flux surface:
The plasma shape in a stellarator is 3D



… extremely sensitive measurement



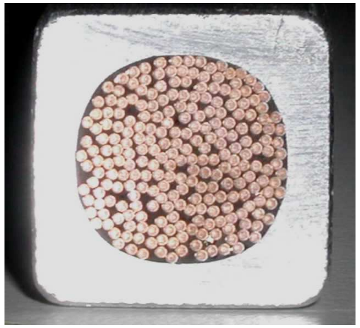
W7-AS:
Field-line tracing with an electron beam using fluorescence in Hydrogen gas (false colour).

W7-AS:
flux surface measurements before operation (dark) and after 56000 discharges (green)

M. Otte, R. Jaenicke, Stell. News (2006)

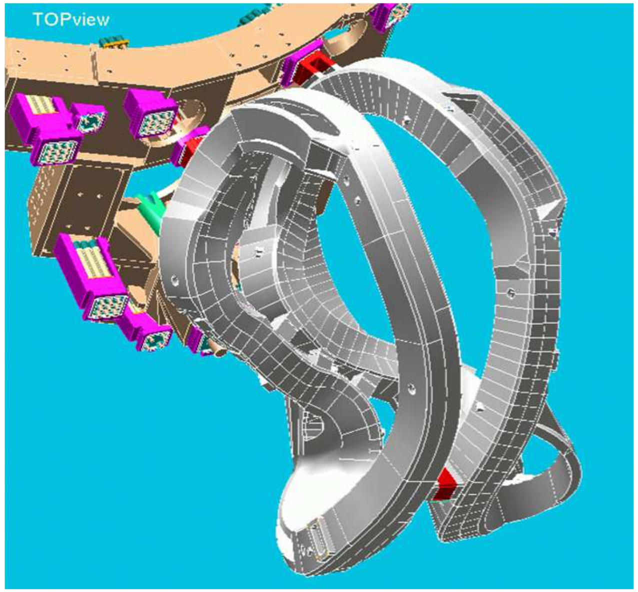
Some technical issues of (modular) coils

- Space between coils (also valid for the high filed side in a tokamak)
- In some areas strong bends required
  - influences choice of superconducting cable conduit
- Coils casings must be strong enough
  - support only in some positions
  - or more or less closed coil housing (NCSX)



Cable-in conduit conductor NbTi

W7-X coil

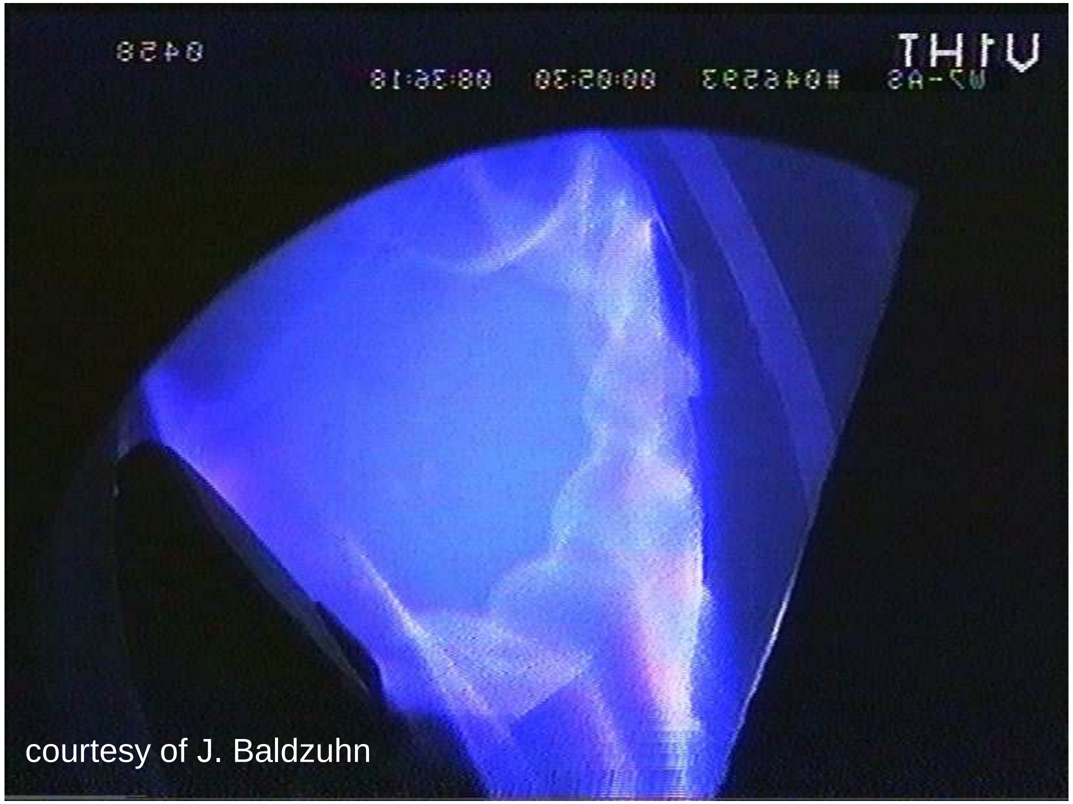


W7-X coil support

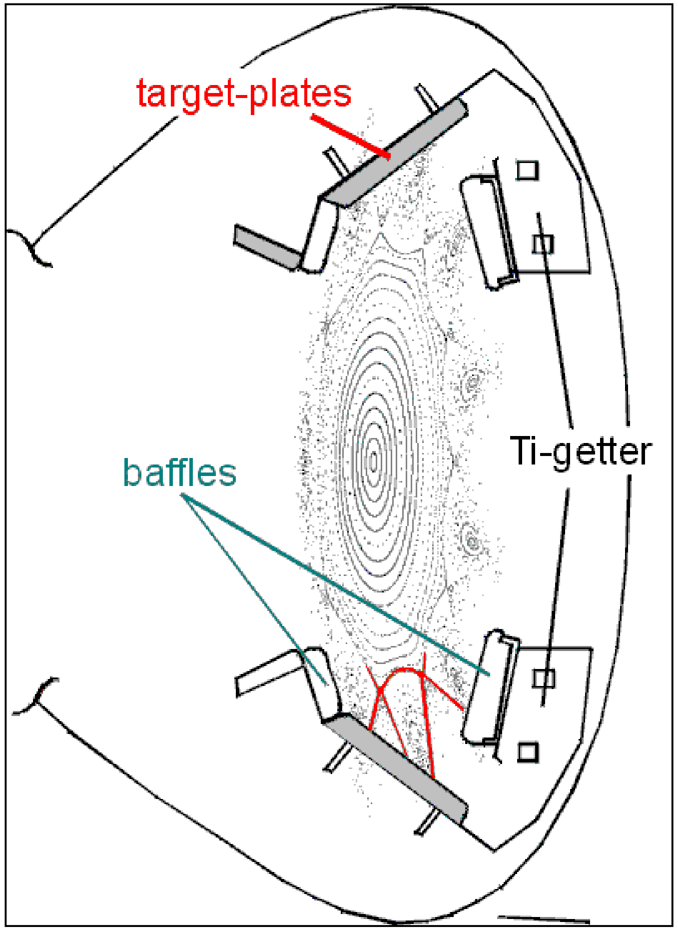
NCSX coil with support

Plasma exhaust requires a feasible divertor concept

Magnetic island divertor in Wendelstein 7-AS



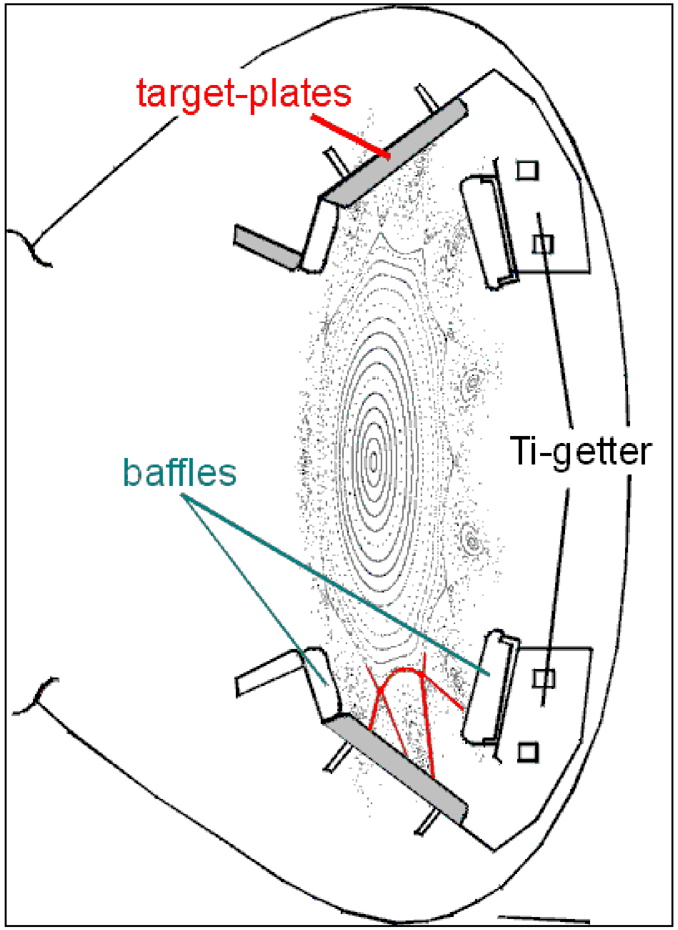
courtesy of J. Baldzuhn



Plasma exhaust requires a feasible divertor concept

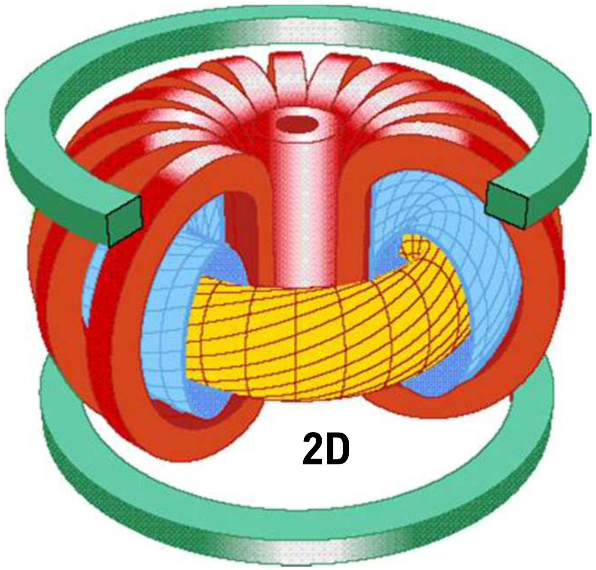
Magnetic island divertor in Wendelstein 7-AS

Divertor module



The stellarator concept

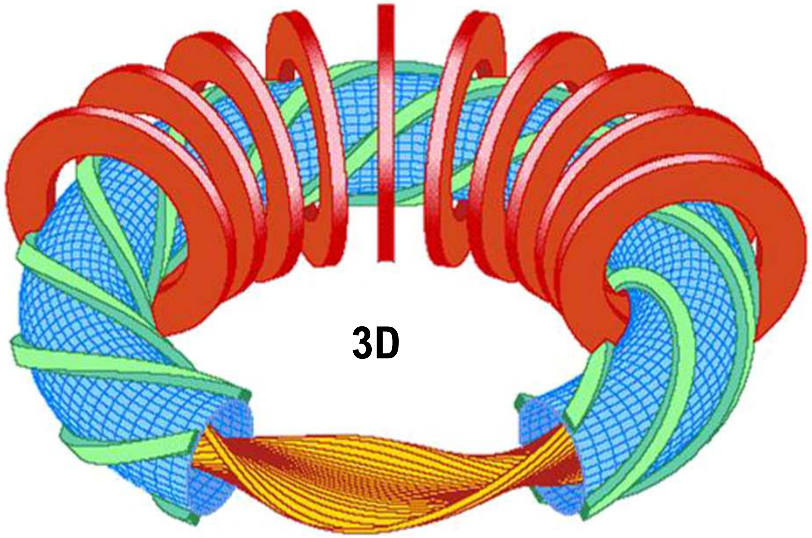
Tokamak



Significant part of the magnetic field generated by a plasma current

- Good confinement properties
- Concept further developed
- Pulsed operation
- Current driven instabilities / disruptions

Stellarator



Magnetic field essentially generated by external coils

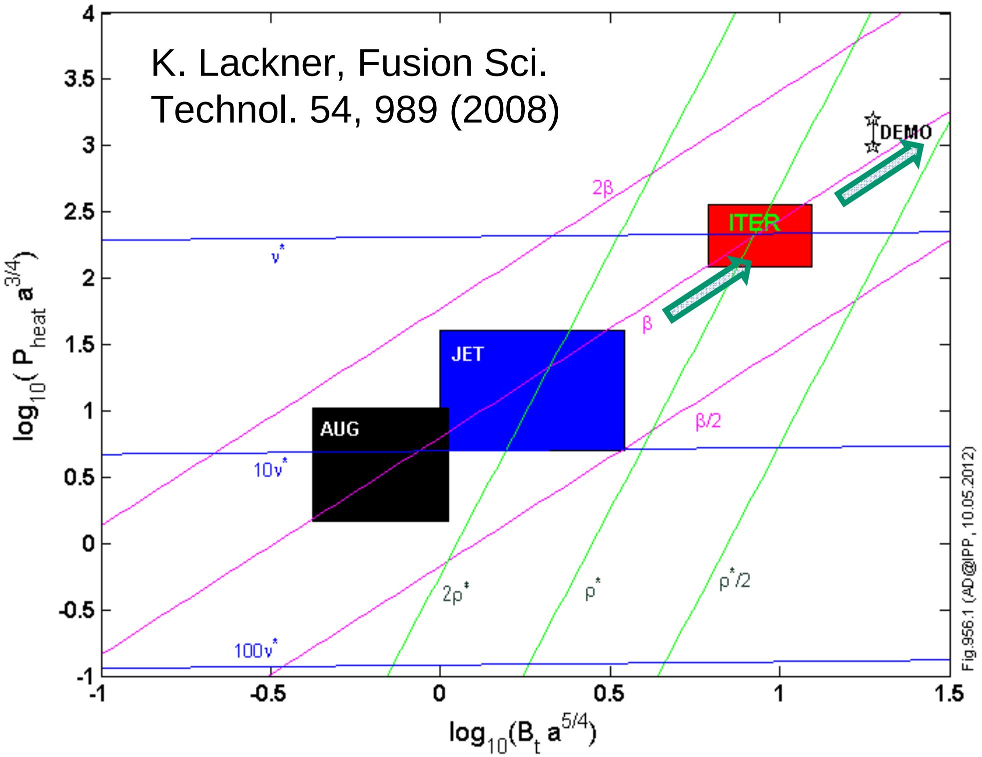
- Requires elaborate optimization to achieve necessary confinement
- Is ~1½ device generations behind
- Intrinsically steady state
- Soft operational boundaries (no disruptions!)

Contents

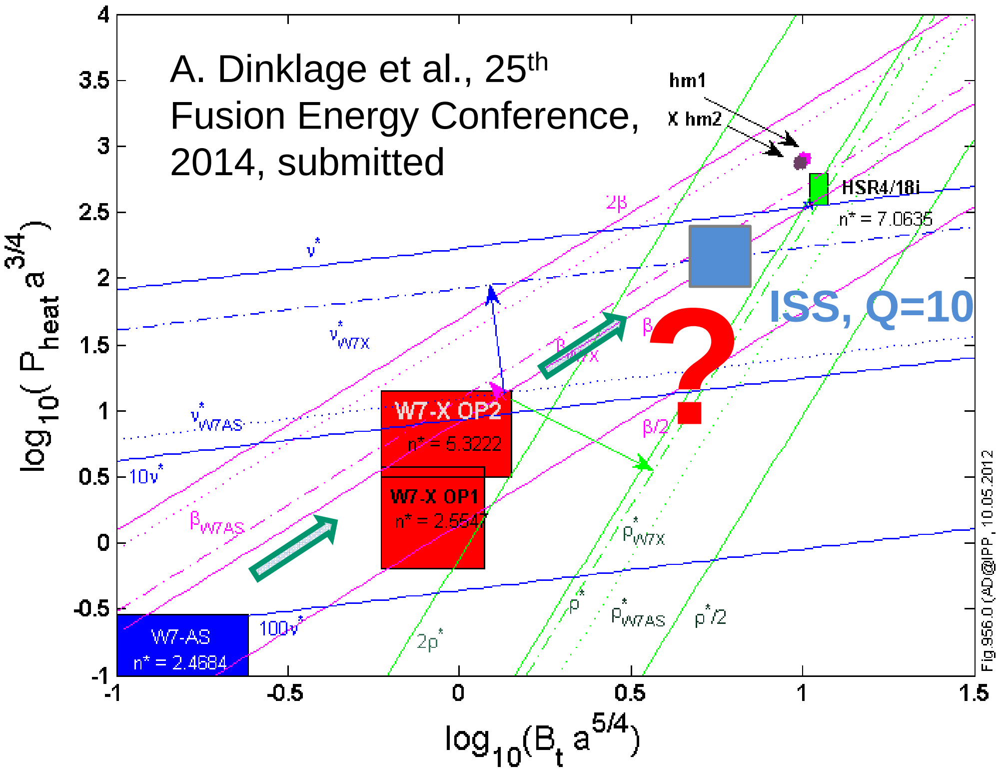
- The stellarator concept
- Issues
  - Sufficient confinement of thermal plasma and fast ions (α-particles in a fusion reactor)
  - Steady state magnetic field
  - Reliable operation at high plasma densities, high plasma pressure (β)
  - Wall materials compatible with heat and particle fluxes (neutron fluxes) and plasma operation, feasible exhaust concept
  - Bringing everything together: The optimized stellarator
- **Extrapolation to a stellarator reactor**
- Summary

Stepladder to Fusion Power Plants

Tokamak



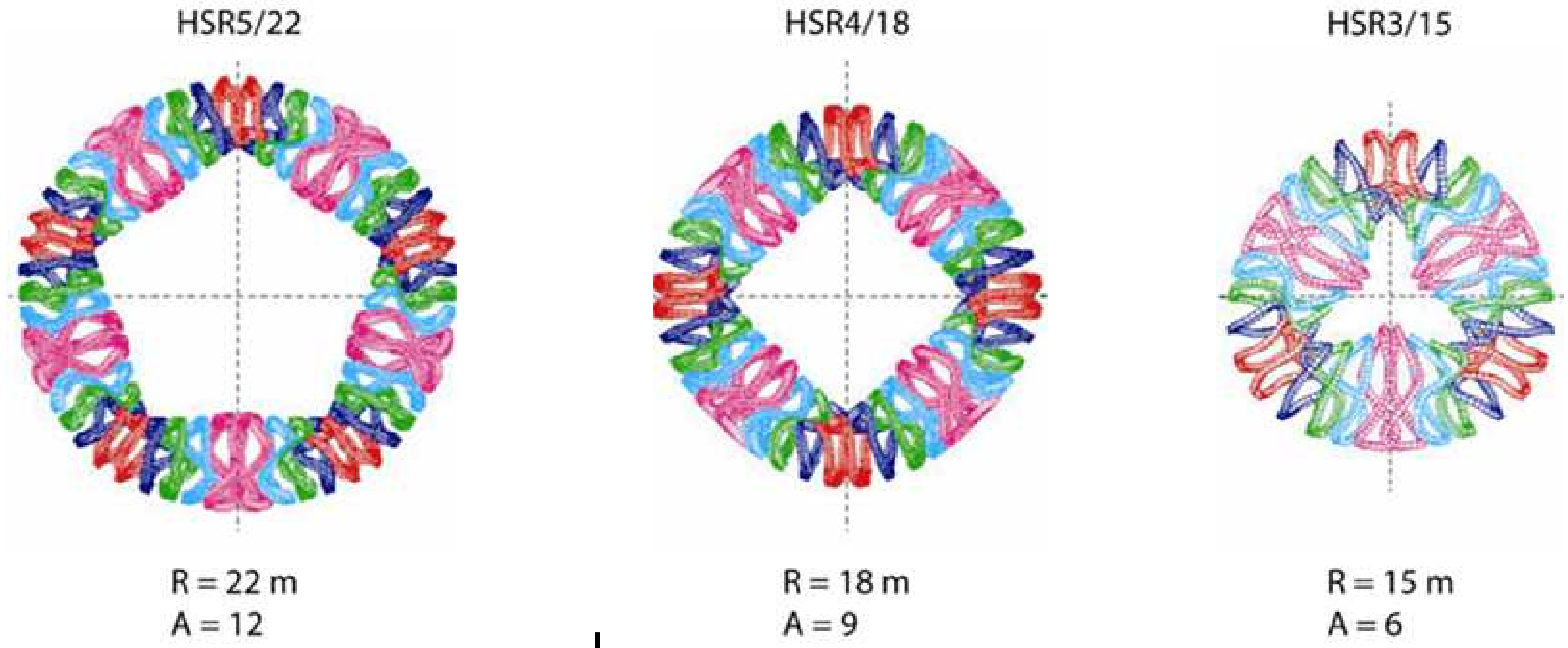
Stellarator



$$\beta = 2\mu_0 \frac{p}{B_t^2} \qquad \nu^* = \frac{R_0\,\nu}{v\,\tau} \qquad \rho^* = \frac{v_i\,m_i}{e\,B\,a}$$

<u>gaps for Stellarator large, burning plasma physics (3D)?</u>

The Helical Advanced Stellarator (HELIAS) line



Good properties

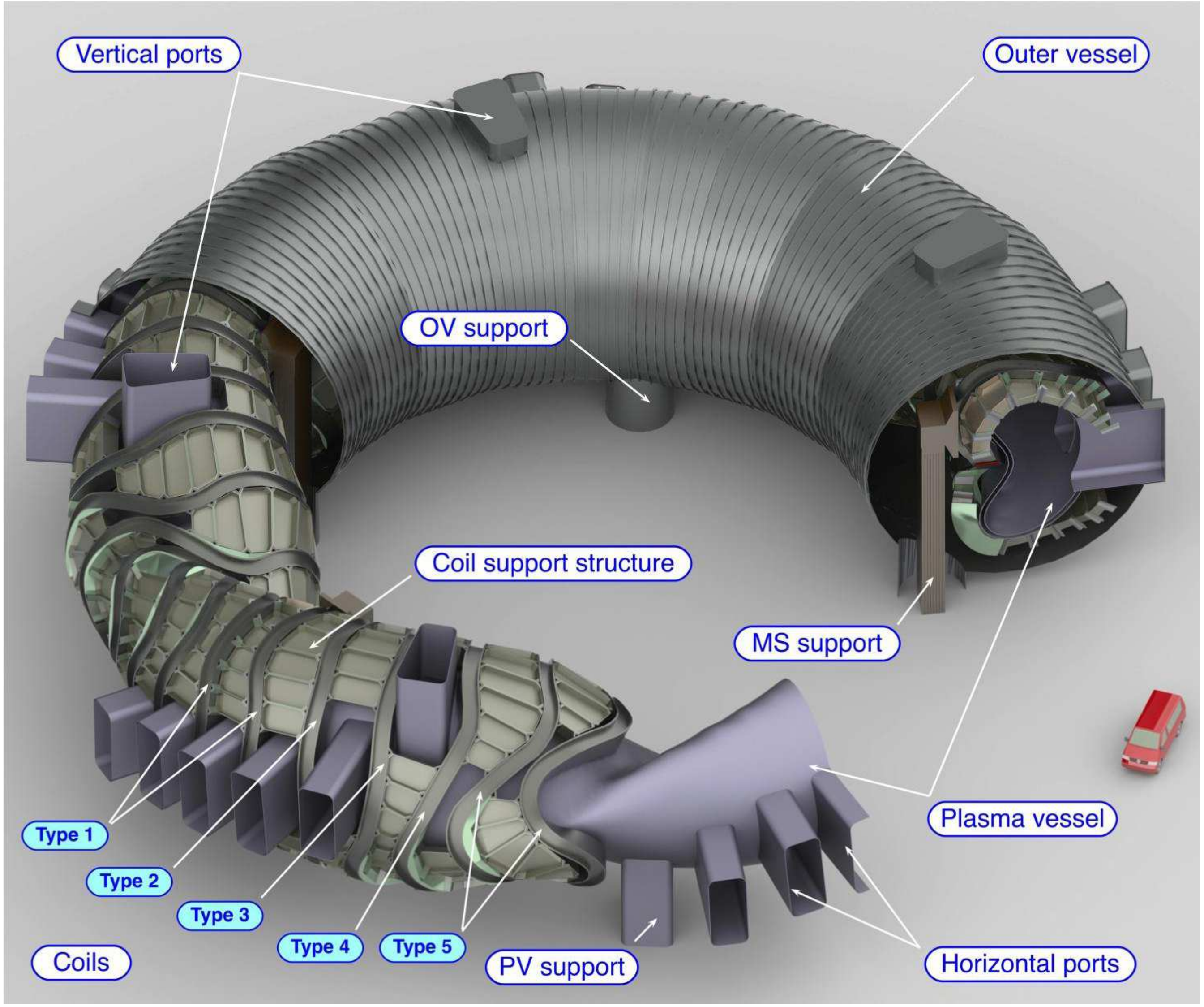
But high aspect ratio

α-confinement ?

Bootstrap current ?

Yu. Igitkanov et al., Fusion Engineering and Design 81 (2006) 2696

Engineering Concept: HELIAS 5-B

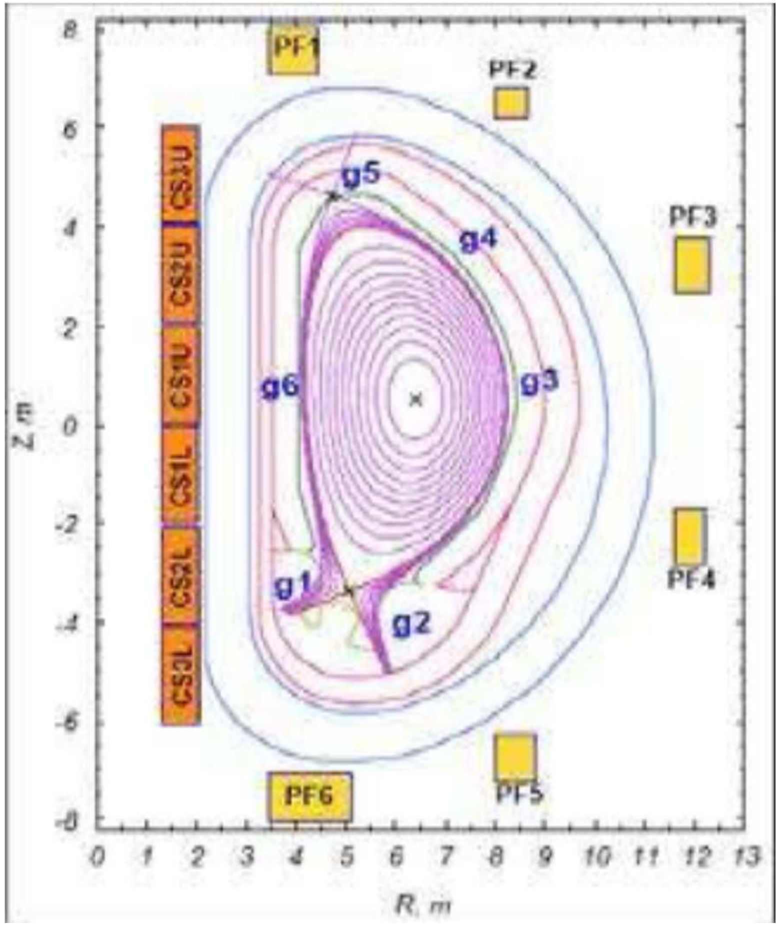


Symmetry	5 Periods
Coil number	50
Major radius	22 m
Overall diameter	60 m
$B_{avg.}$ on axis	5.9 T
B on coil	12.5 T
Coil current	13.65 MA
Magnetic energy	160 GJ (44.4 MWh)

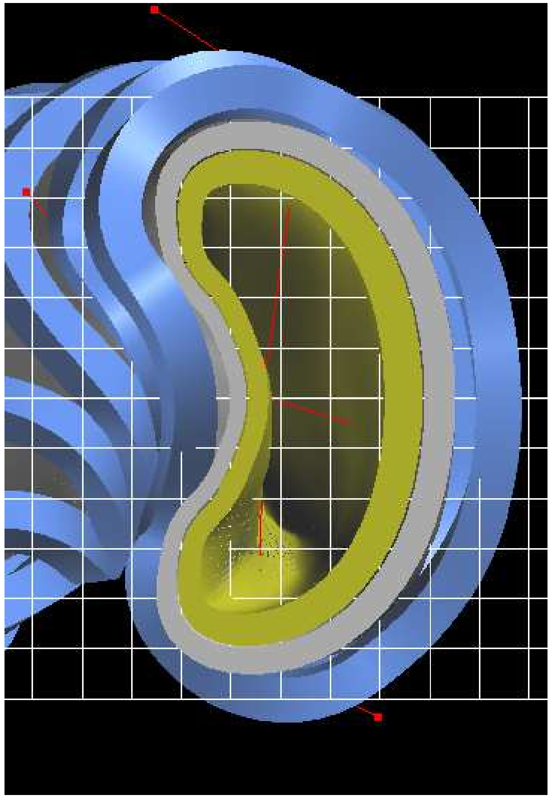
F. Schauer et al., Fusion Engineering and Design 88 (2013) 1619

Coil study based on ITER superconductor technology

Comparison of ITER and HELIAS 5-B coils
(same scale)



ITER toroidal field (TF) coil

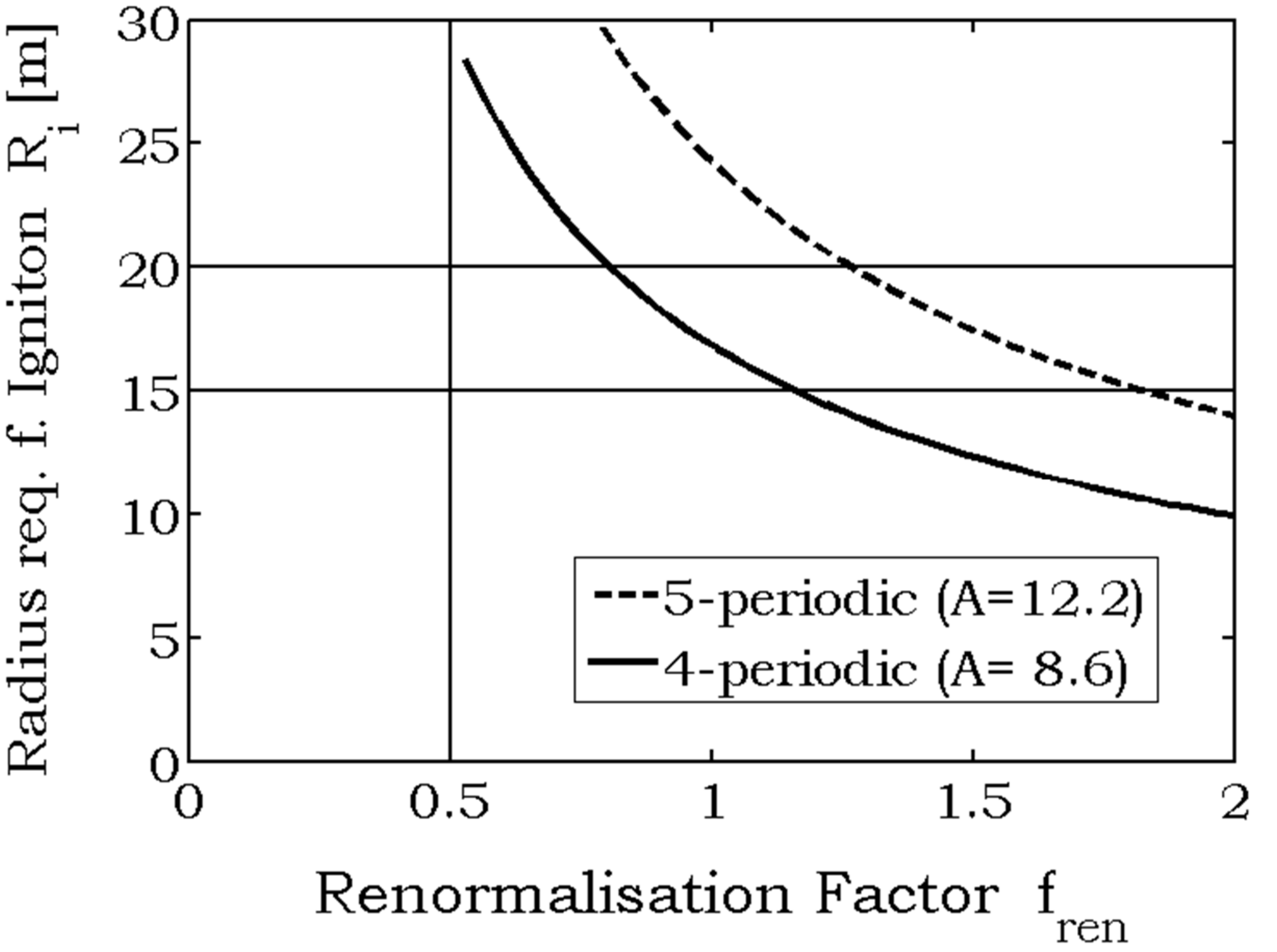


HELIAS 5-B coil #1

Similar size!

F. Schauer et al., Contrib. Plasma Phys. 50 (2010) 750

Can the HELIAS be build smaller?

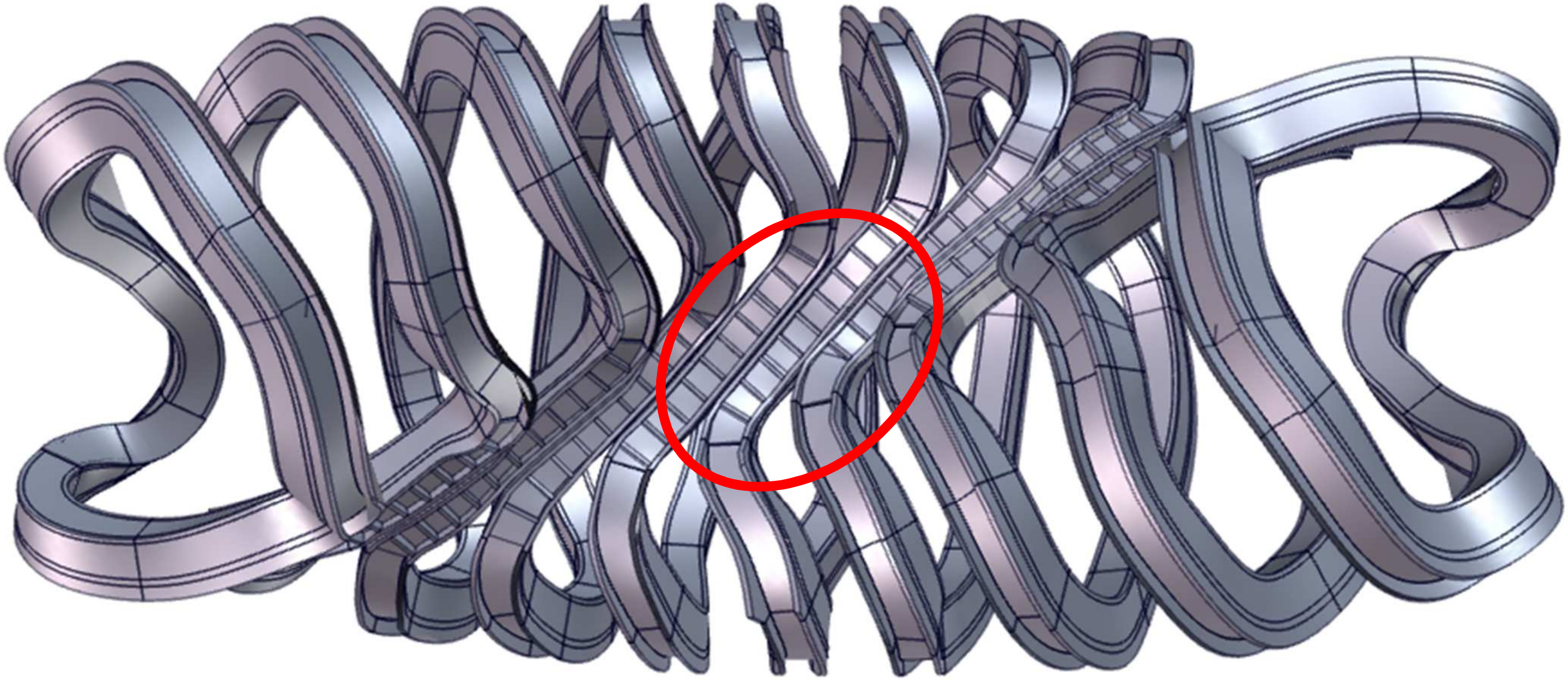


Theoretically YES!

= confinement improvement w.r.t. Empirical scaling law

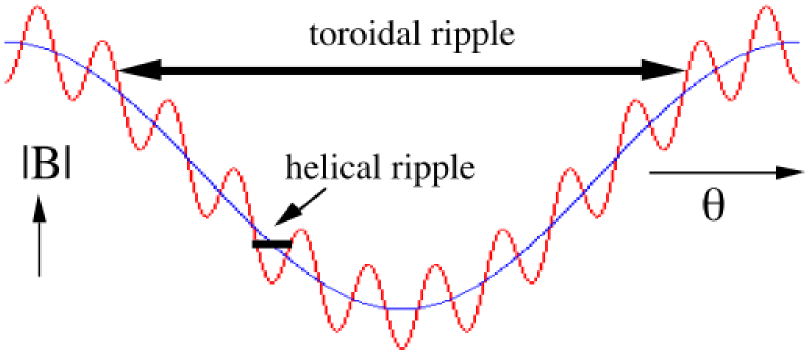
Why not build smaller?

“Extrapolated” HELIAS 5-B Coil Set

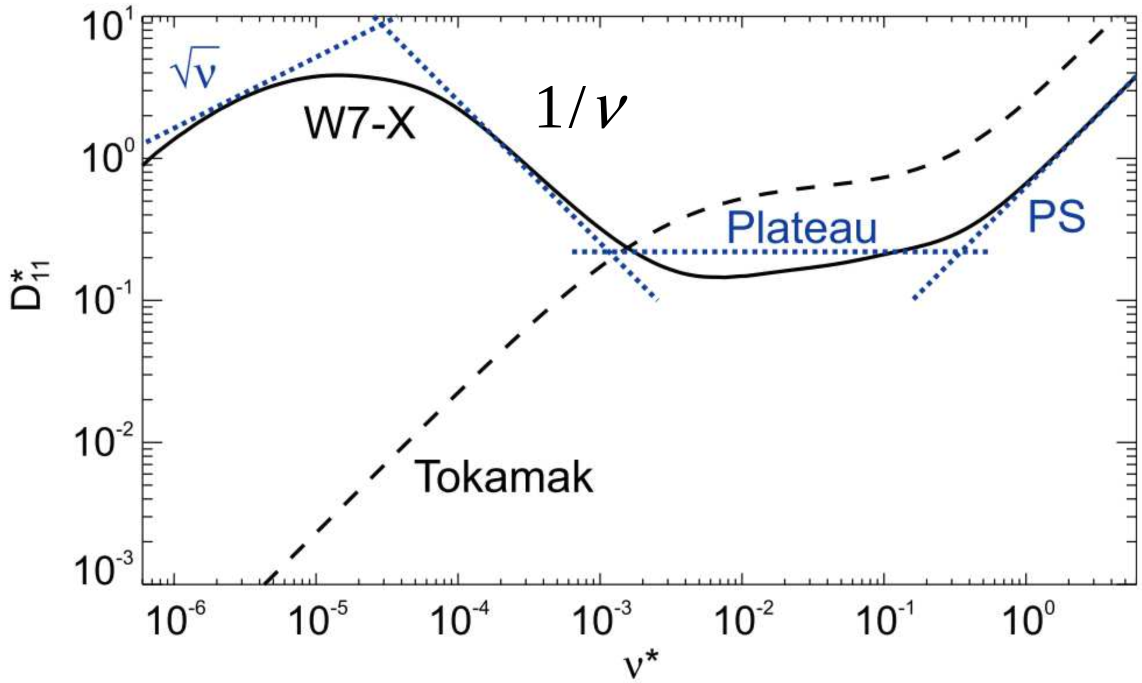


Strong effort required to design coils for more advanced magnetic configurations

Optimisation of Neoclassical Transport



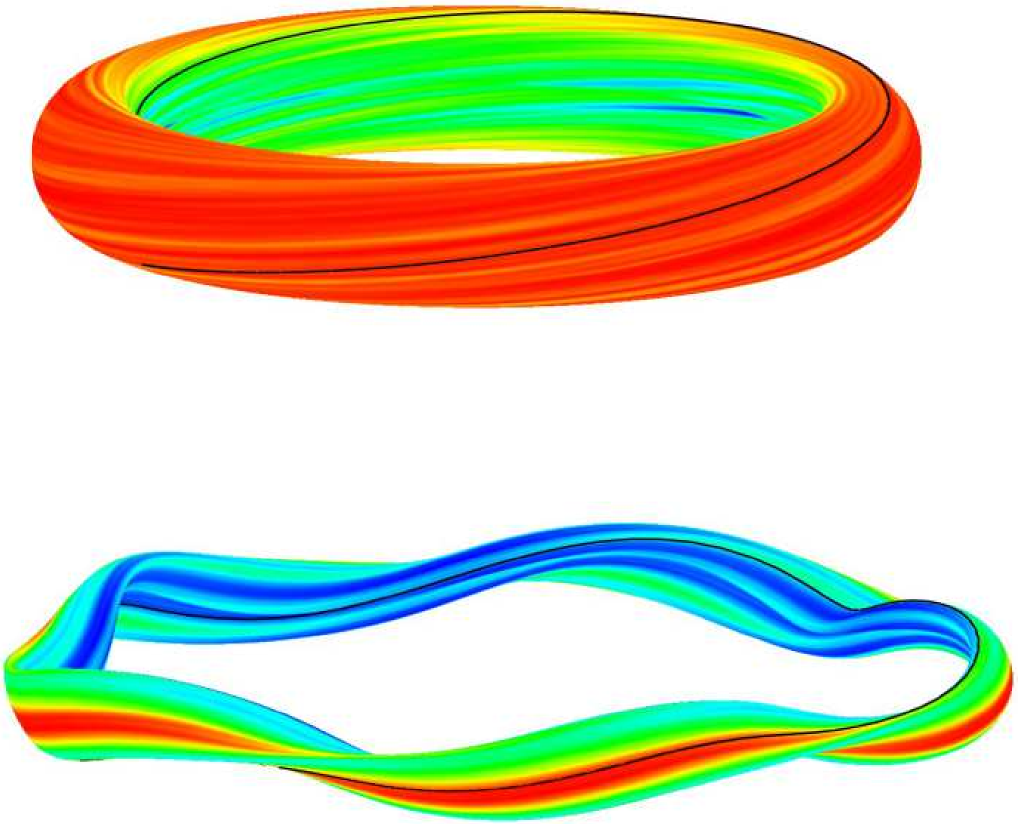
- **Localised, trapped particles**
  - → **Drift losses, i.e. "neoclassical" transport**
- **well-validated and established models existing**
- **has been minimised in advanced configurations**
  - → **e.g. Maximum-J configurations**



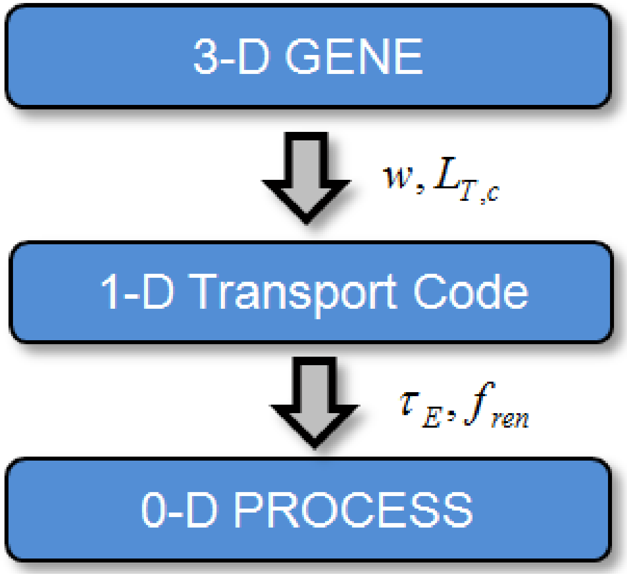
$$D_{1/\nu} \sim \varepsilon_{eff} T^{7/2}$$

see e.g. C. Beidler et al., Nucl. Fusion 51 (2011) 076001
or P. Helander et al., Plasma Phys. Control. Fusion 54 (2012) 124009

ITG Turbulence



- Optimisation of neoclassical transport
  → turbulence becomes important!
- TEMs possibly stabilised in Helias configurations
- ITG turbulence qualitative different

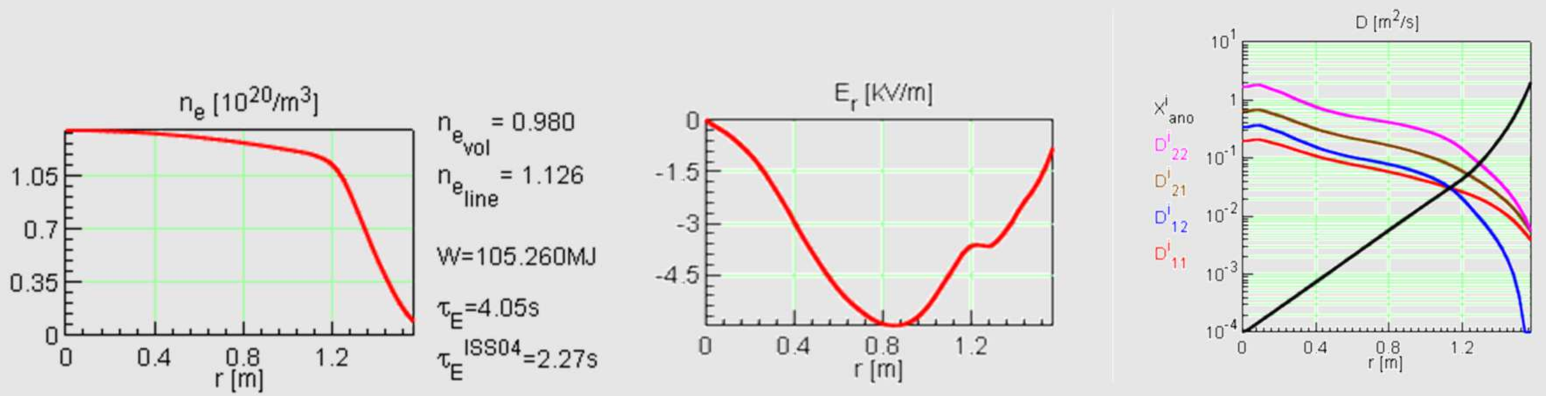


see e.g. P. Helander et al., Plasma Phys. Control. Fusion 54 (2012) 124009

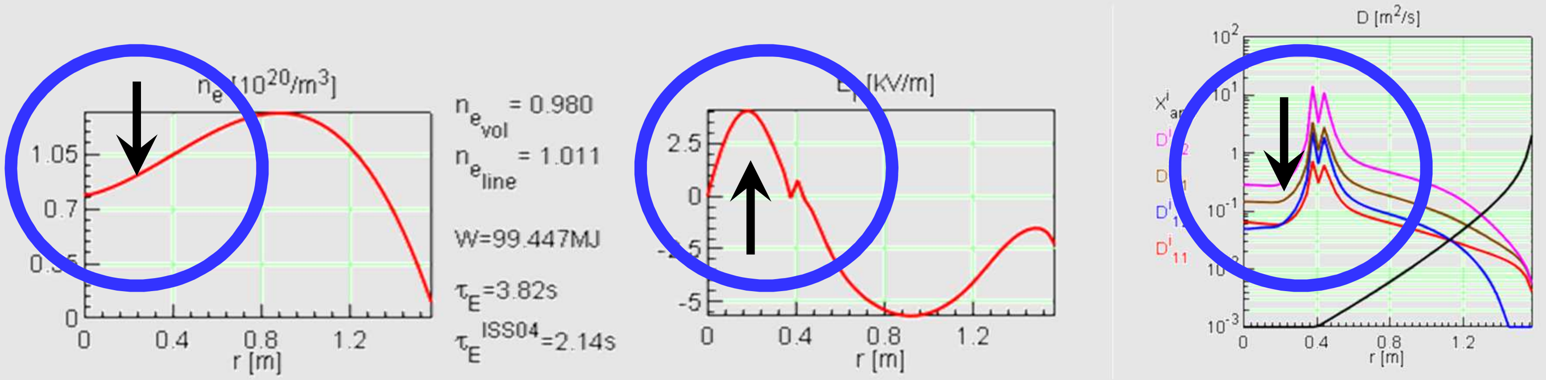
Arrive at integrated predictive modelling!

Effect of profile shaping (W7-X * 3 predictive simulations)

flat



hollow



Warmer, Dinklage, Beidler, Turkin, Wolf (2012, unpublished)

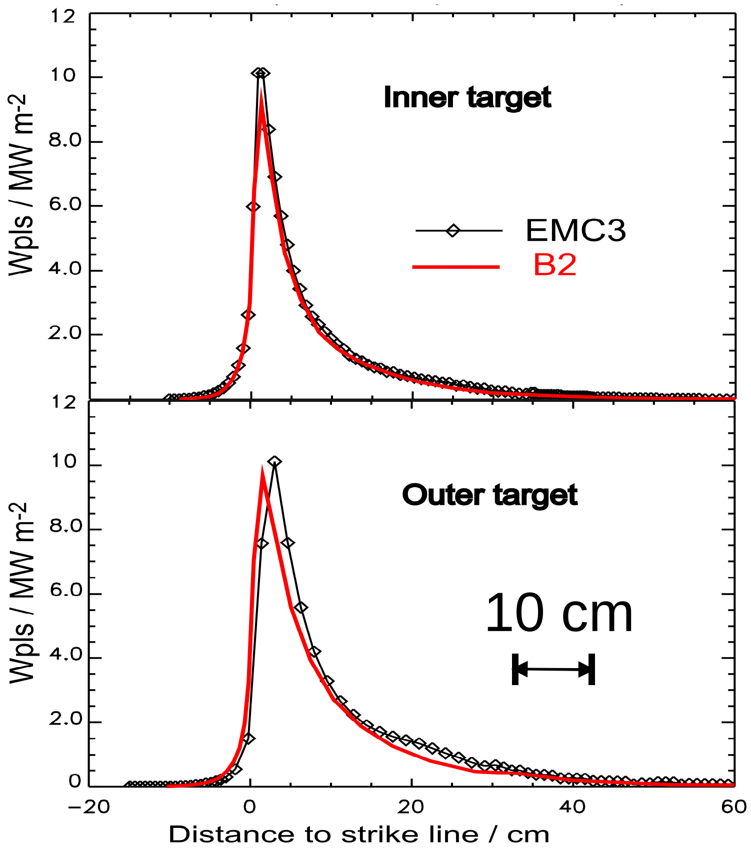
Neoclassical theory: positive E_r for n´/n > 0: impact on αs and impurities in the core?

(scenario development is done parallel to systems code tasks)

Power Exhaust

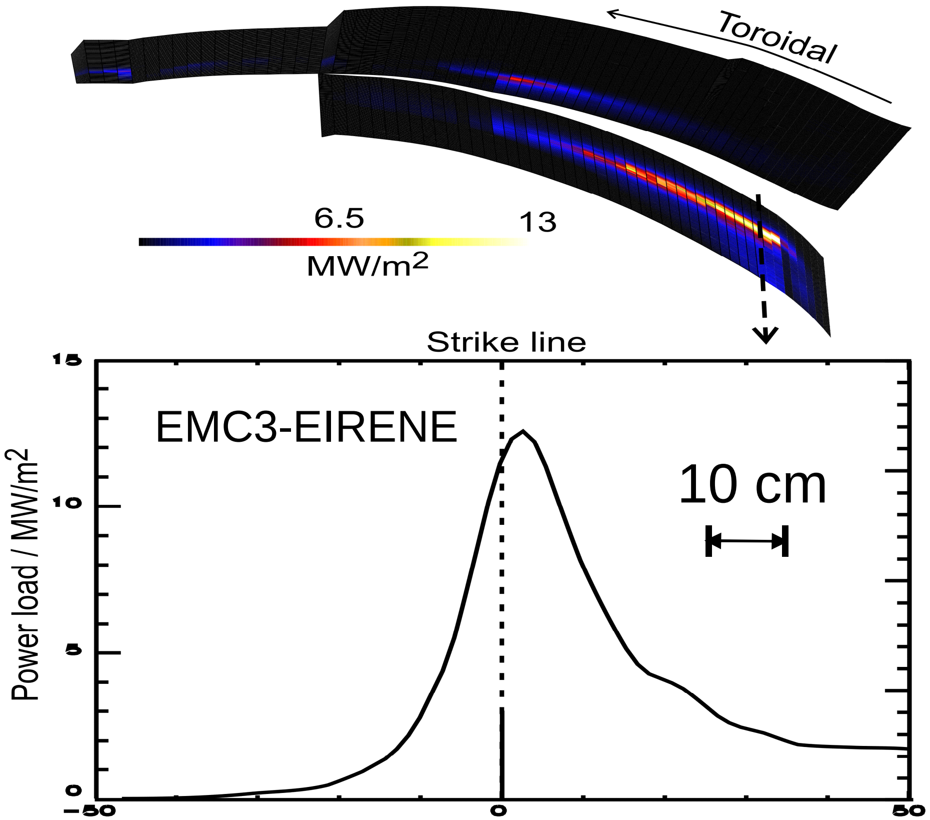
ITER

P_{SOL}=60 MW, D=0.3m²/s, $\chi_e = \chi_i$ =1m²/s



W7-R

P_{SOL}=200 MW, D=0.5m²/s, $\chi_e = \chi_i$ =1.5m²/s

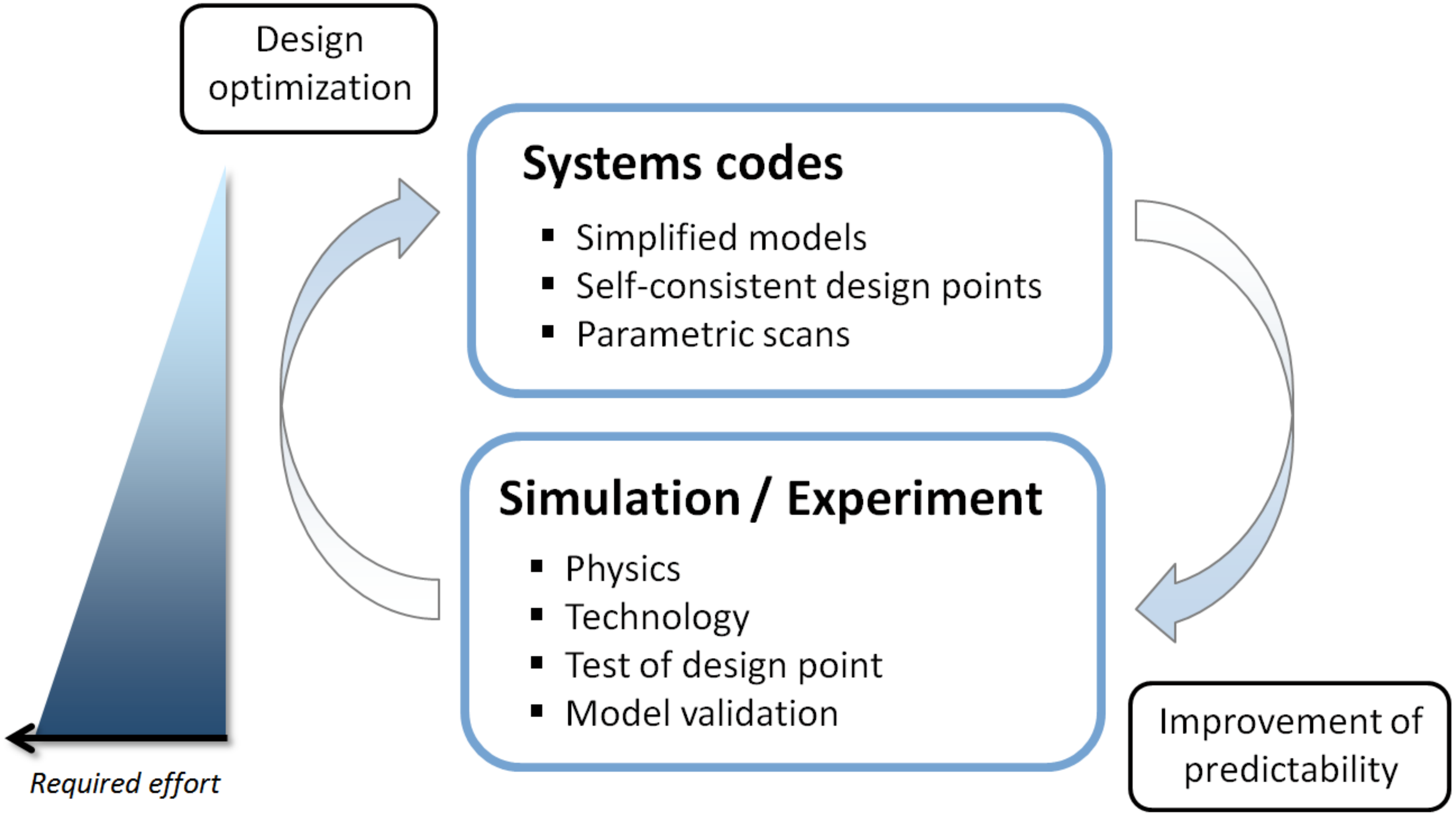


Assume PSOL = 300 MW and a margin of 100% accounting for asymmetries in power load

Radiation fraction for a P_{peak} = 5 MW/m²: **Tokamak: 0.95** **Stellarator: 0.86**

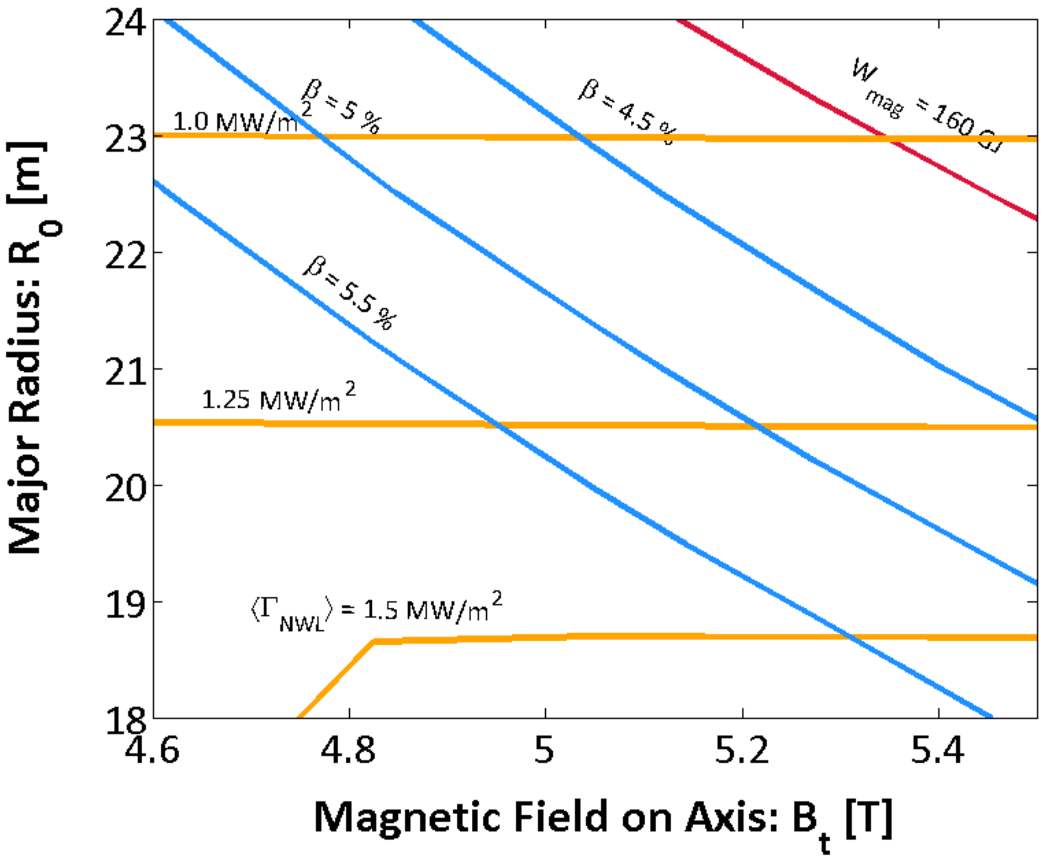
Y. Feng et al., Plasma Phys. Control. Fusion 53 (2011) 024009

Systems Code Approach



Iterative procedure for design point optimization and validation

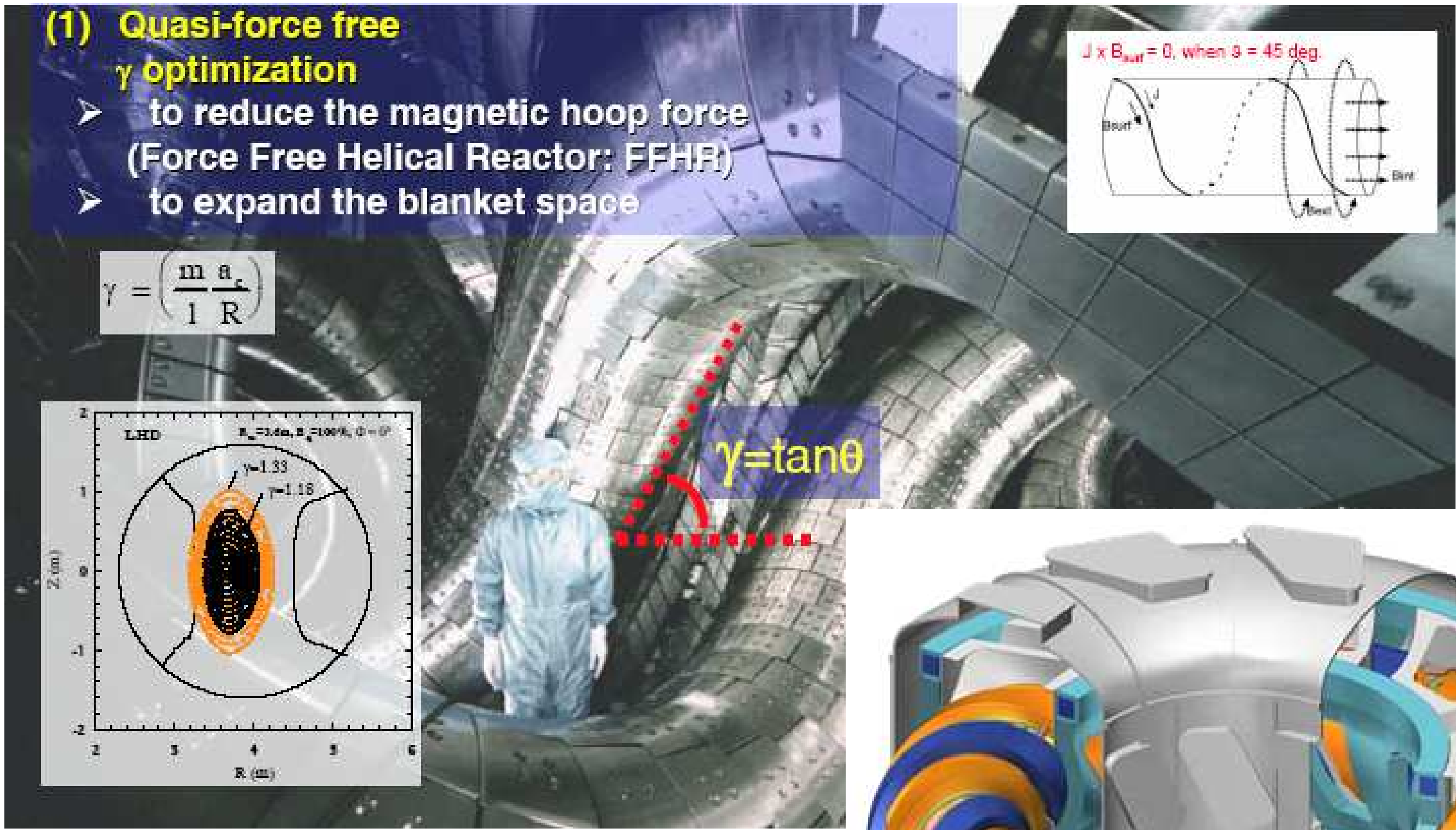
HELIAS



Requirements

- **Sufficiently good confinement to provide ignition**
- **Nb3Sn superconductor technology**
- **Sufficient space for blanket (~1.3 m)**
- **$\langle\beta\rangle$ = 4 – 5 %**
- **Fusion power ~ 3GW**
- **Advantage of large aspect ratio**
  → reduced neutron flux to the wall (average 1 MW/m², peak 1.6 MW/m²)

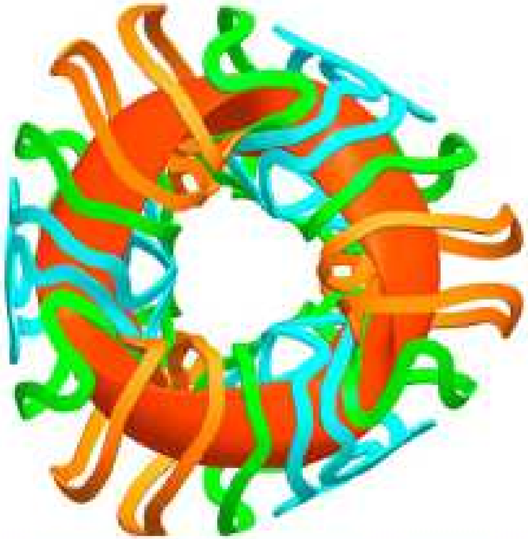
Force Free Helical Reactor (FFHR) – LHD type



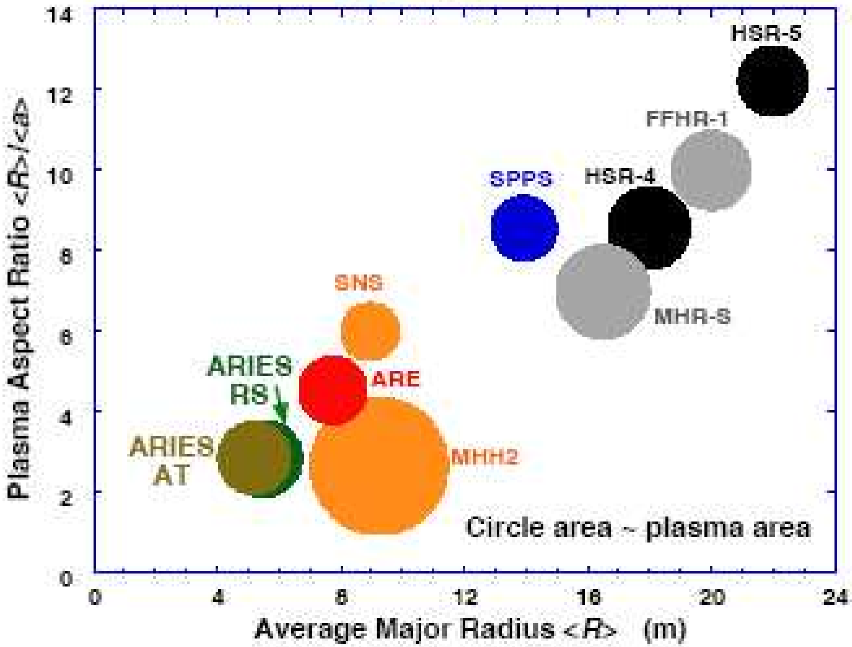
FFHR
1 GW
6 Tesla
25,000 tons

O. Motojima et al., Fusion Eng. Design 83 (2007) 983

ARIES – Compact Stellarator (CS) – NCSX type



Min. coil-plasma distance (m)	1.3
Major radius (m)	7.75
Minor radius (m)	1.7
Aspect ratio	4.5
β (%)	5.0
Number of coils	18
B_o (T)	5.7
B_{max} (T)	15.1
Fusion power (GW)	2.4
Avg./max. wall load (MW/m²)	2.6/5.3
Avg./max. plasma q'' (MW/m²)	0.58/0.76
Alpha loss (%)	~5



A. R. Raffray et al., Fusion Sci. Techn. 54 (2008) 725

UC San Diego

Thank you for your Attention!

If you are interested:

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