Operational Limits in Stellarators

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Contents

- The stellarator concept
  - Some stellarators which are discussed in this presentation
- Operational limits
  - **Transport effects**
  - **Equilibrium effects**
  - **Stability limits**
  - **Density limits**
- Summary

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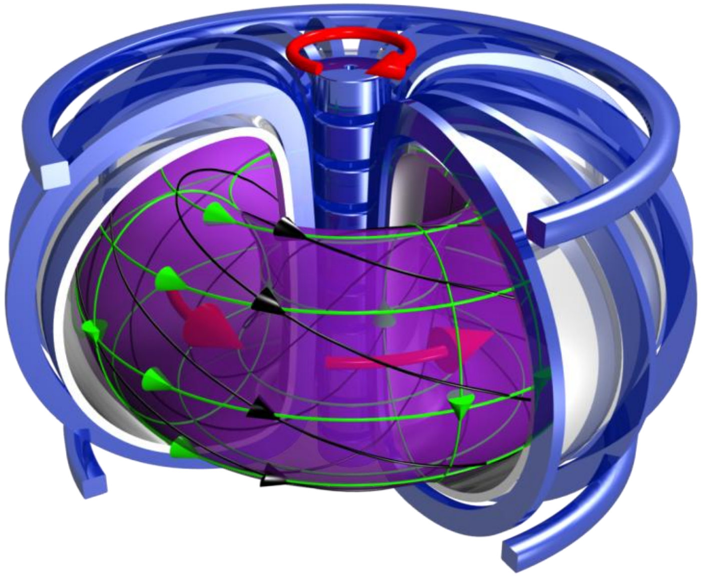
The stellarator concept

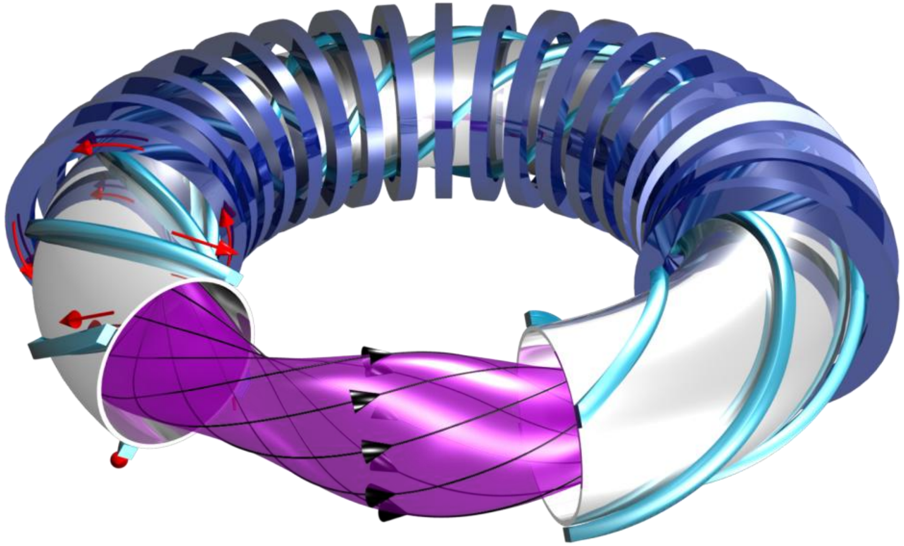
Tokamak (2D)



Significant part of the magnetic field generated by a plasma current

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

Stellarator (3D)



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

The stellarator concept: advantages

- **Intrinsically steady state magnetic field (no current drive)**
  - current drive requirements limited to small adjustments of the rotational transform
    (one to two orders of magnitude smaller than in tokamaks)
  - intrinsically lower re-circulating power (could operate ignited)
  - quiescent steady state (at high β)
- **No current driven instabilities**
  - no need to control profiles (?)
  - no need for feedback or rotation to control instabilities, or nearby conducting structure
- **No disruptions**
  - eases design of plasma facing components (breeding blanket)
  - disruption avoidance or mitigation schemes not required
- **Very high density limit (no Greenwald limit)**
  - easier plasma solutions for divertor
  - reduced fast-ion instability drive

The stellarator concept: disadvantages

- **3D magnetic field configuration**
  - generally poor neoclassical confinement
  - generally poor fast particle confinement
  - tendency for impurity accumulation
  - more complex divertor (and other plasma facing components)
  - more complex coil configuration

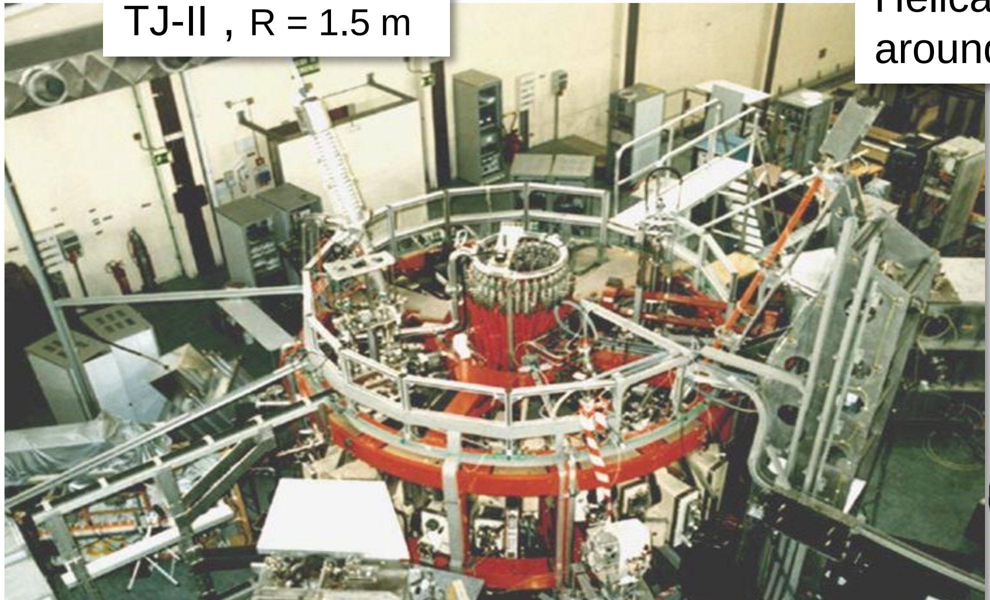
Physics issues addressed by stellarator optimization

Physics issue addressed by finding a suitable confinement / operating regime

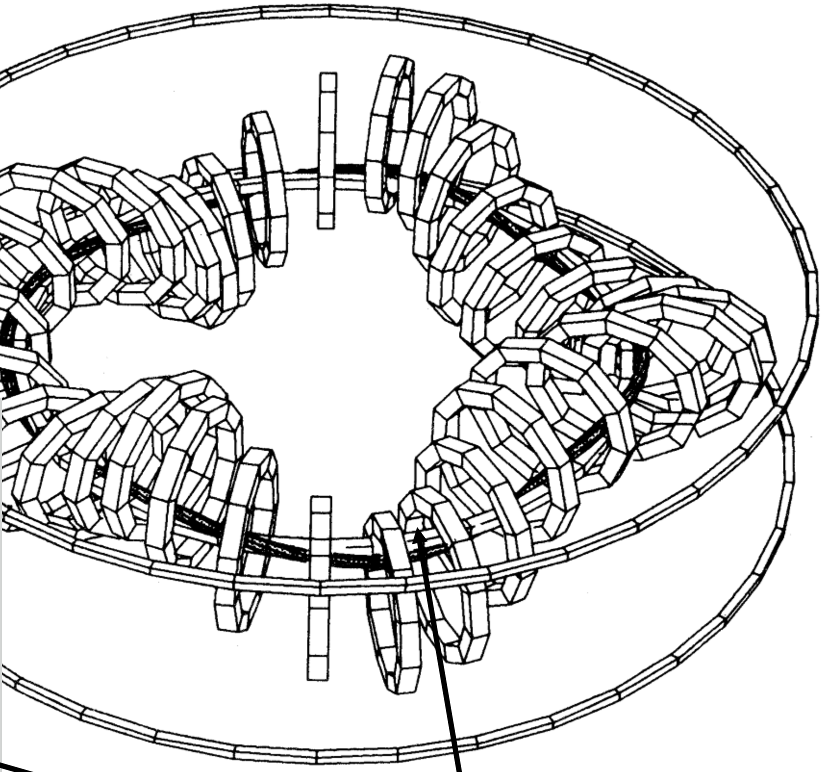
- Engineering issues addressed when designing and building new devices
- Development of feasible concepts will become important reactor design
- Here issues are maintenance and remote handling

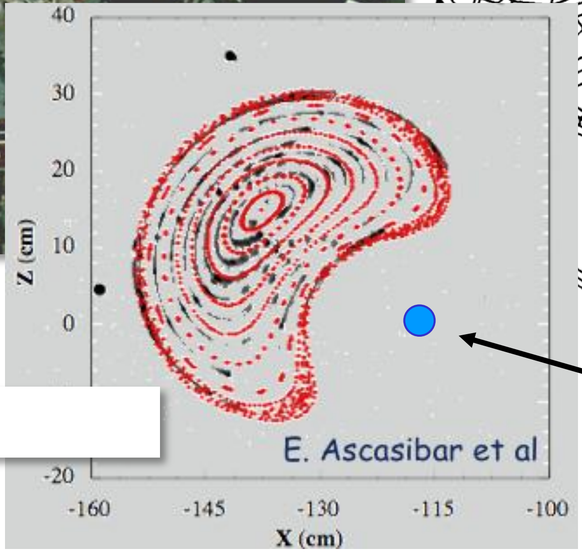
The Heliac TJ-II (Madrid, Spain)

TJ-II , R = 1.5 m



Helical component achieved by winding the plasma around a single central conductor (helical axis)





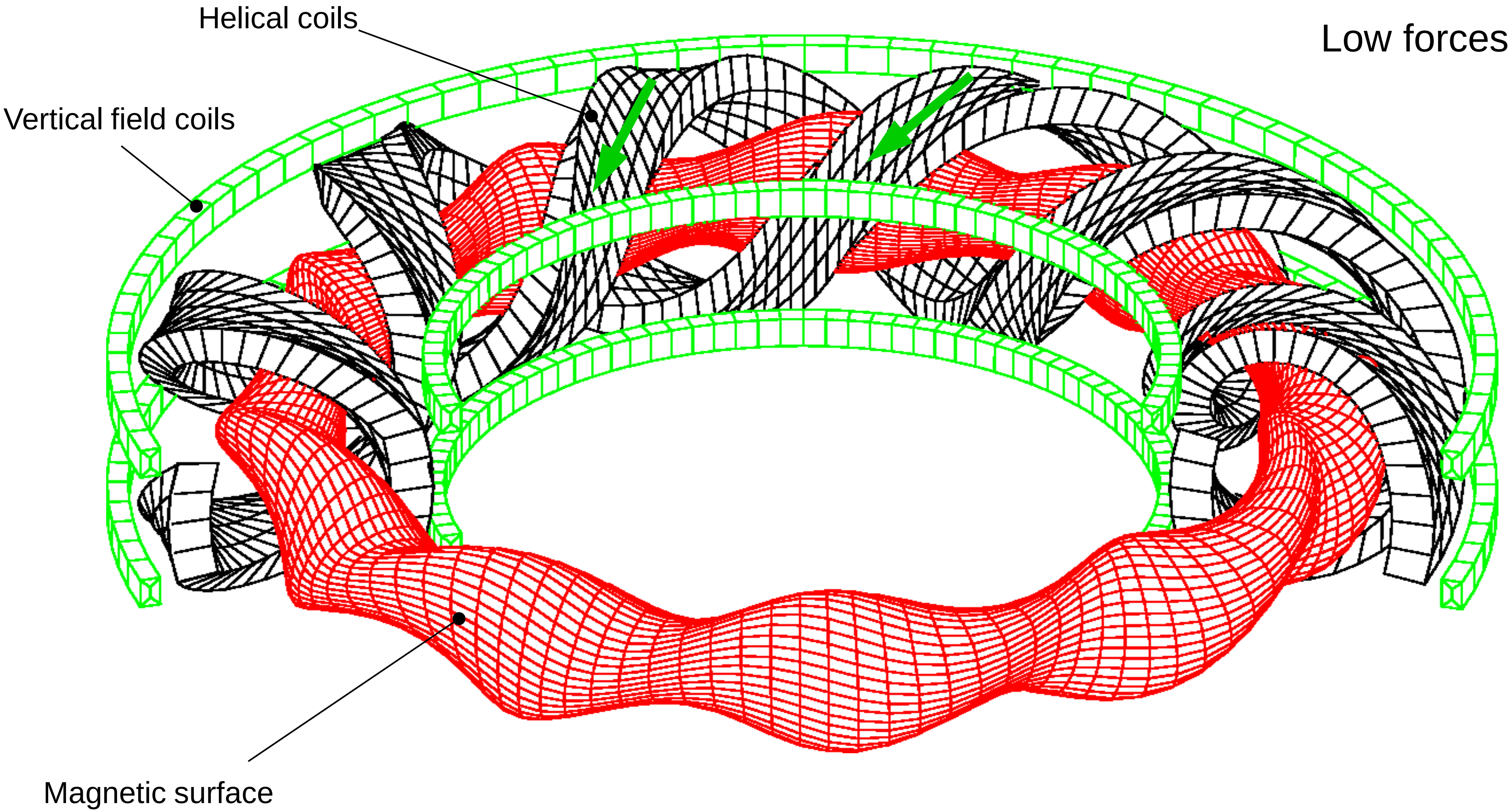
E. Ascasibar et al

Constant plasma cross section

Central conductor

The torsatron / heliotron

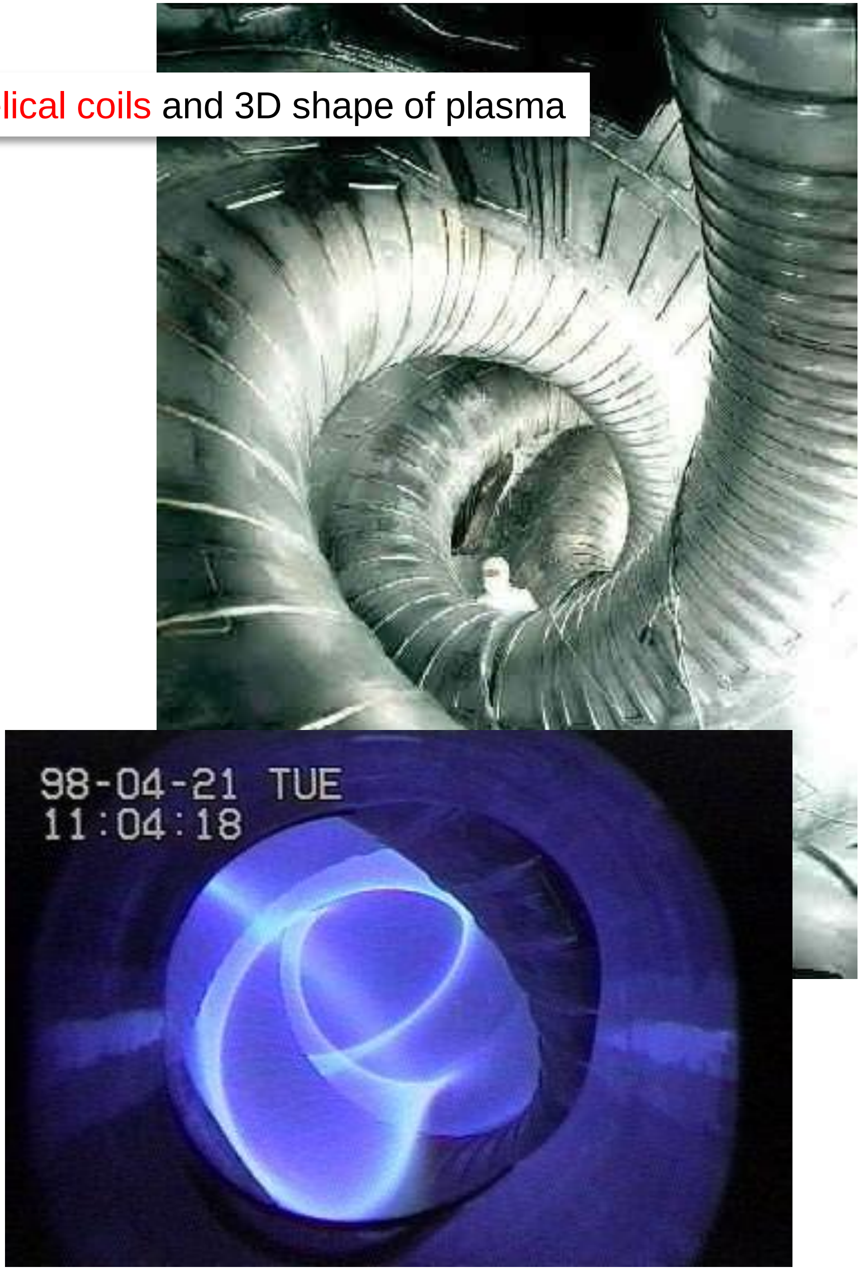
No toroidal field coils at all



Currents in conductors flow in same direction
→ vertical B-field must be balanced

The Large Helical Device (Toki, Japan)

LHD
R= 3.5-4.1 m, a= 0.6 m -> V= 28 m^3
superconducting coils



Helical coils and 3D shape of plasma

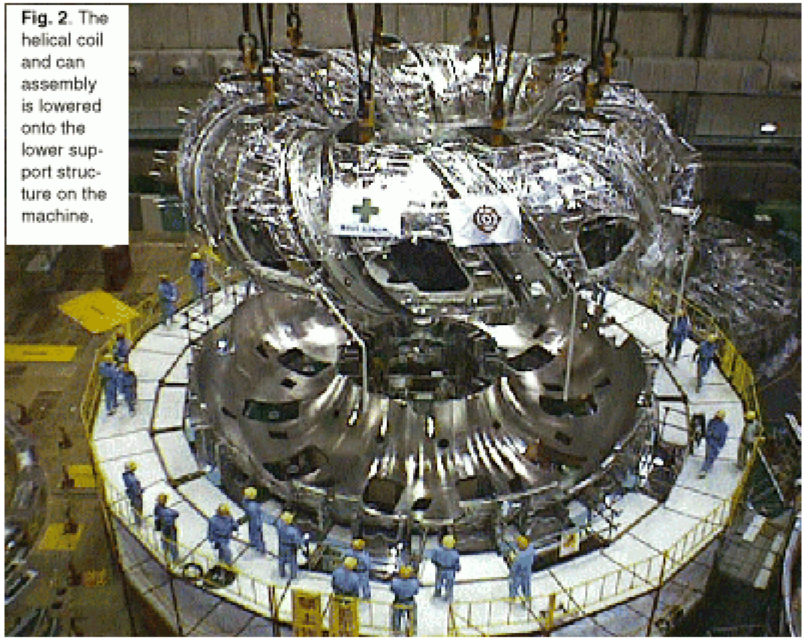
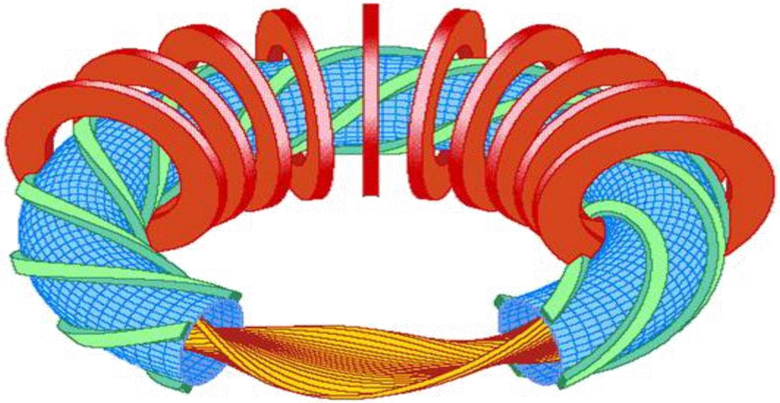


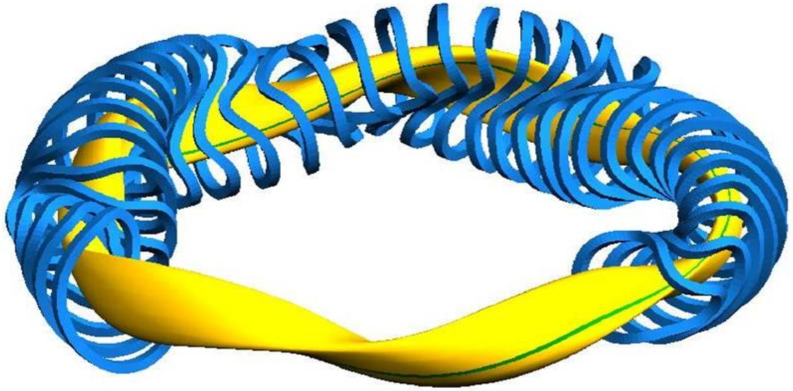
Fig. 2. The helical coil and can assembly is lowered onto the lower support structure on the machine.

The modular stellarator



Toroidal field coils

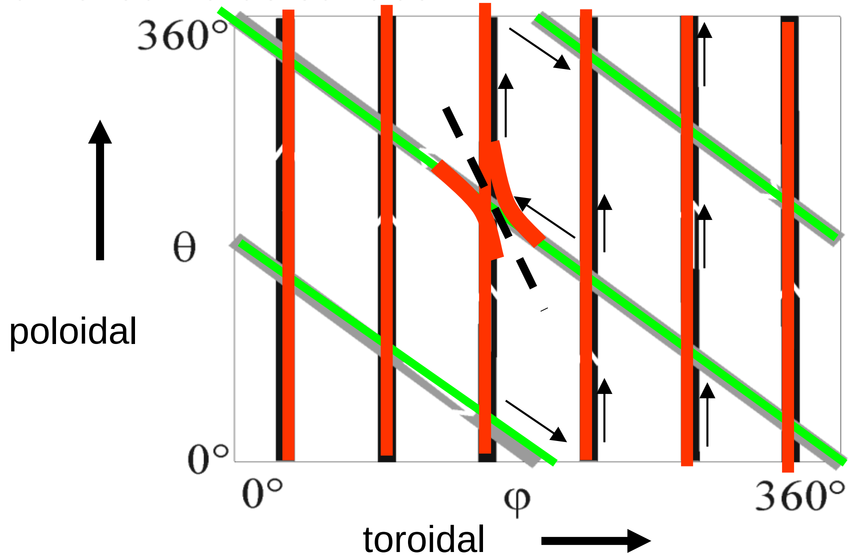
Helical field coils

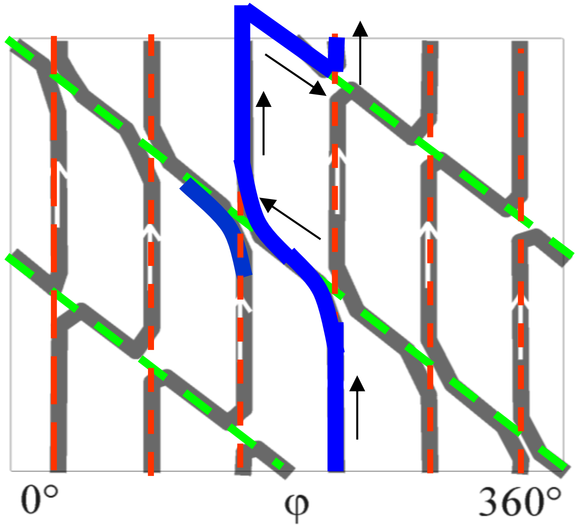


W7-X

Modular coils

“unrolled” torus surface:



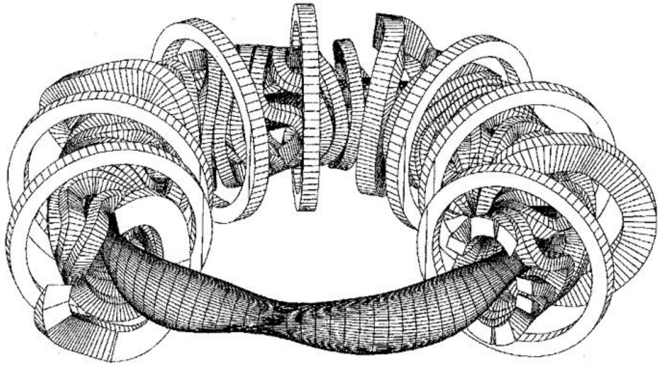


→ No huge helical coils / mechanical forces can be handled

→ Magnetic fields can be tailored by external current distribution

Modular stellarators

First modular concept by Rehker and Wobig 1972

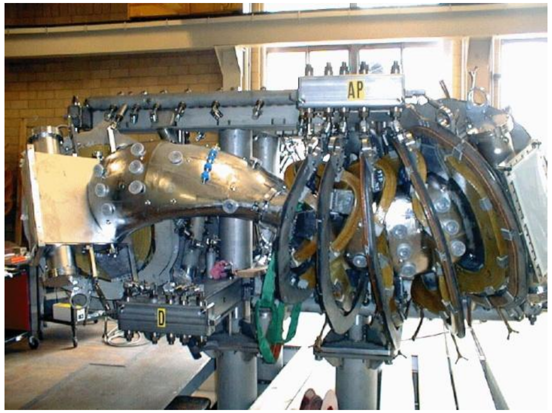


Wendelstein 7-AS (Garching, Germany)

- 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

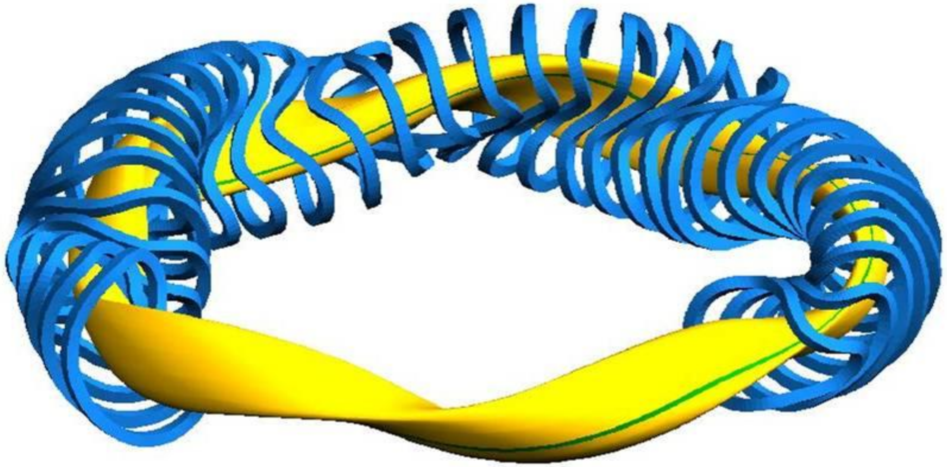
NCSX (PPPL, Princeton, USA, mothballed)

- Quasi-axissymmetric stellarator
- $R = 1{,}42m, a = 0{,}33m ,$ $V = 3\ m^3, B = 1{,}2 – 2\ T$
- Mothballed during construction

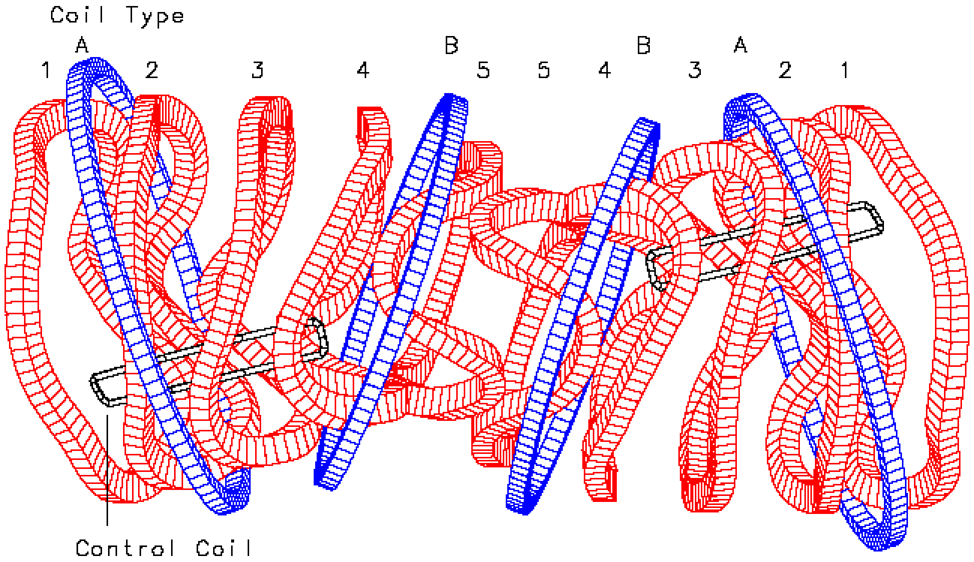
Modular stellarators

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, first plasma 2015

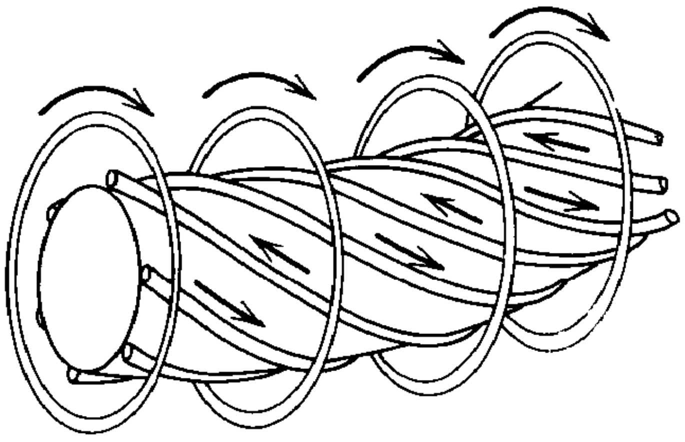


One module of the W7-X coil system

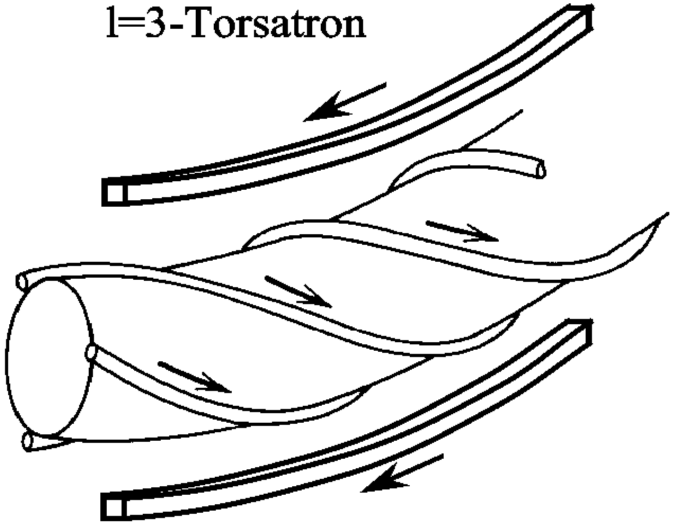


Summary – the stellarator concept

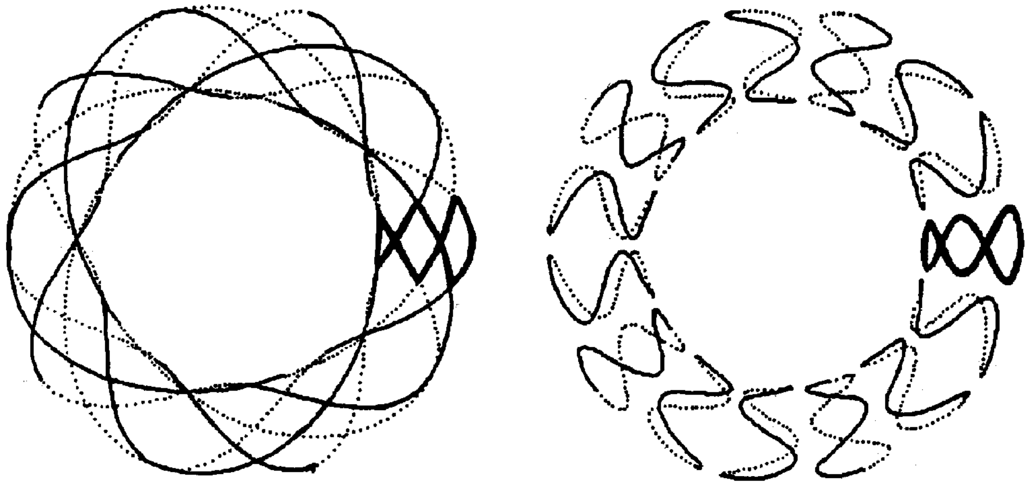
l=3-Stellarator



l=3-Torsatron



Modular



Essential for stellarator optimization

… but large forces require sufficiently strong coil casings and coil support structure

www.ipf.uni-stuttgart.de/lehre/plasmaphys

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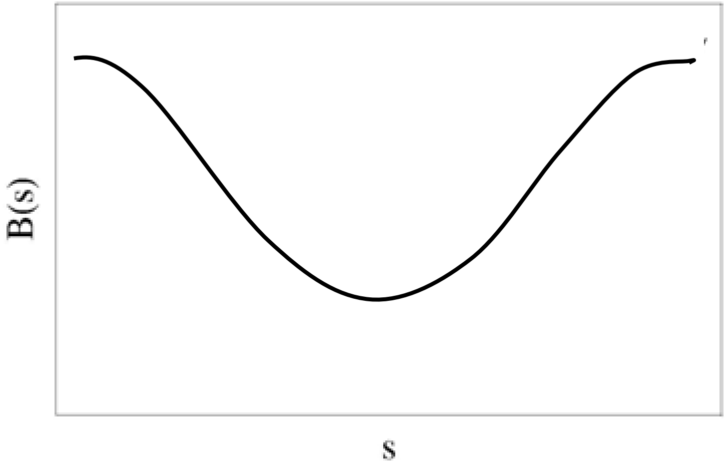
Plasma confinement

In a torus: Modulation of the magnetic field strength along the magnetic field lines

magnetic field gradients; field line curvature

Diffusion of the thermal plasma: $D \sim \varepsilon_{eff}^{3/2} \cdot T^{7/2}$

Generally, because of large mean free path, radial drift of fast ions



Tokamak
→ toroidal trapping
(toroidal ripple not shown)

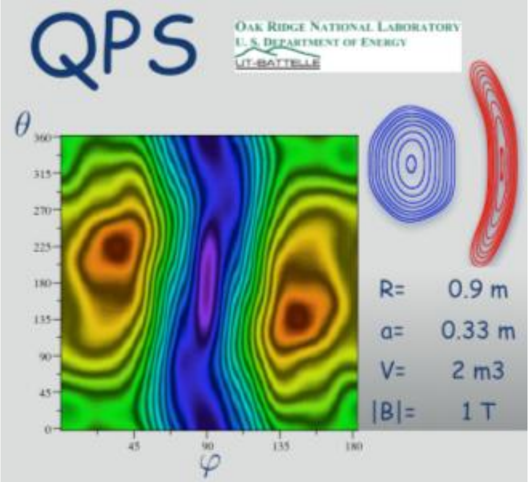
Stellarator
→ toroidal trapping
→ helical trapping

Coordinate along field line, one toroidal circumference

→ Quasi-symmetry acts on ε_{eff}

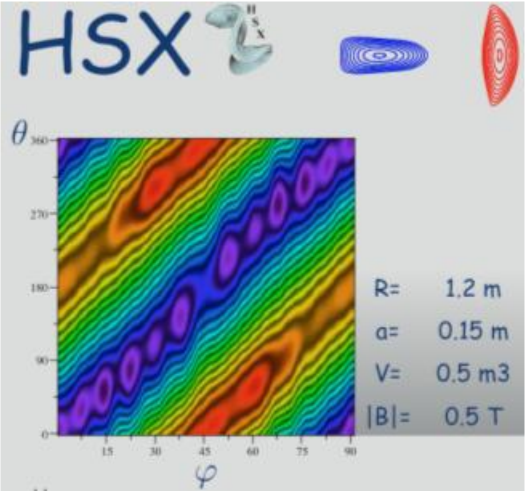
Quasi-symmetries

quasi-poloidal



R= 0.9 m
a= 0.33 m
V= 2 m3
|B|= 1 T

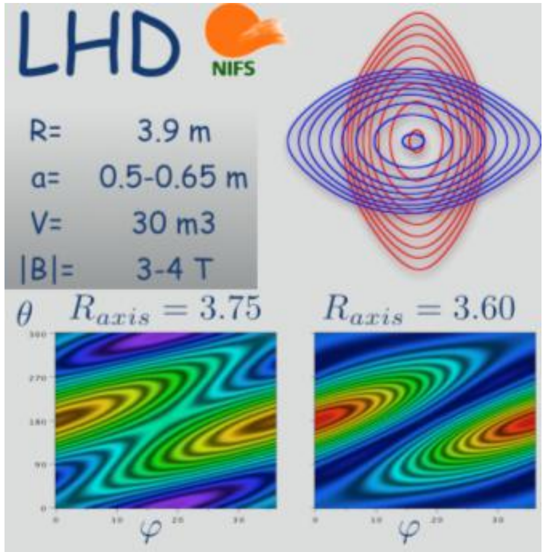
quasi-helical



R= 1.2 m
a= 0.15 m
V= 0.5 m3
|B|= 0.5 T

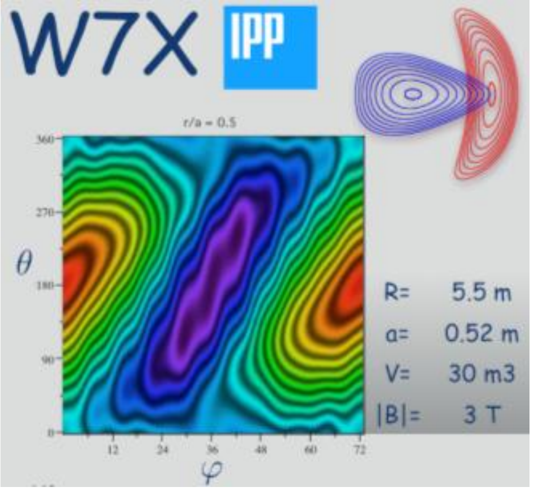
see Canik et al., PRL 98 (2007) 085002

classical



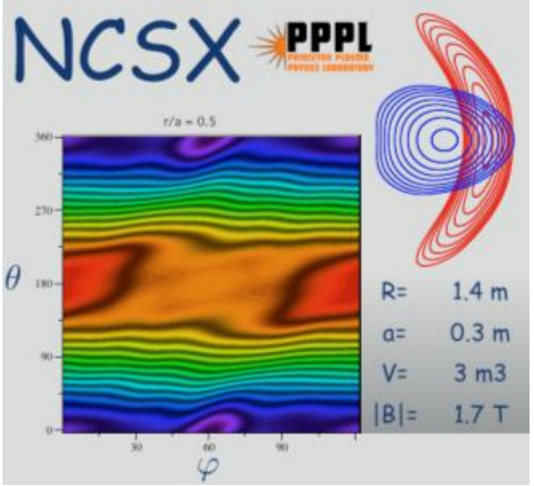
R= 3.9 m
a= 0.5-0.65 m
V= 30 m3
|B|= 3-4 T

quasi-isodynamic



R= 5.5 m
a= 0.52 m
V= 30 m3
|B|= 3 T

quasi-toroidal

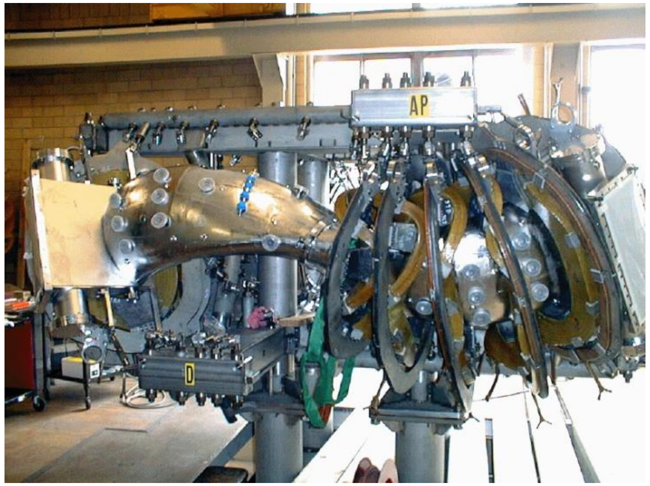


R= 1.4 m
a= 0.3 m
V= 3 m3
|B|= 1.7 T

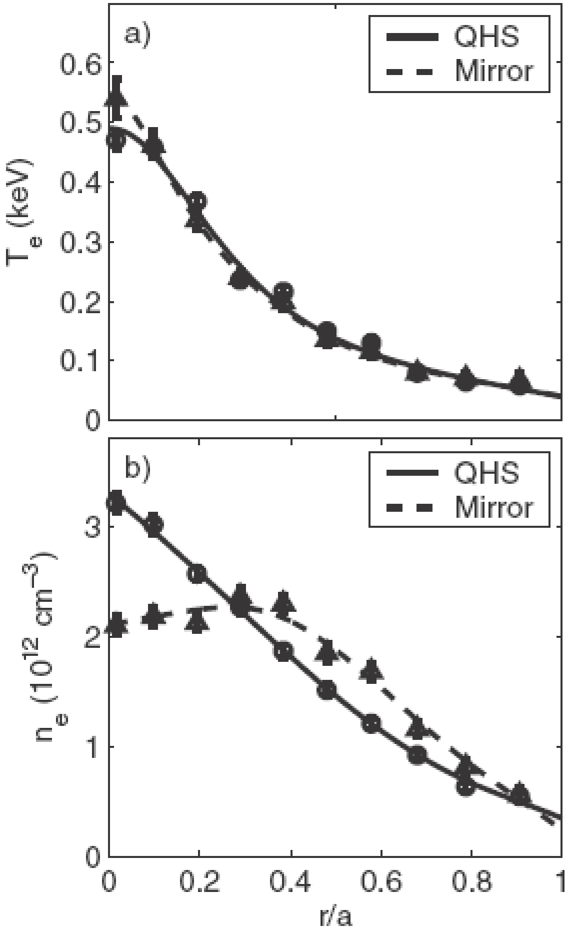
courtesy of J. Sanchez

Effect of quasi-symmetry on confinement (minimize ε_{eff})

HSX (Madison, WI, USA)
26 kW – quasi-helical symmetry (QHS)
67 kW – mirror configuration



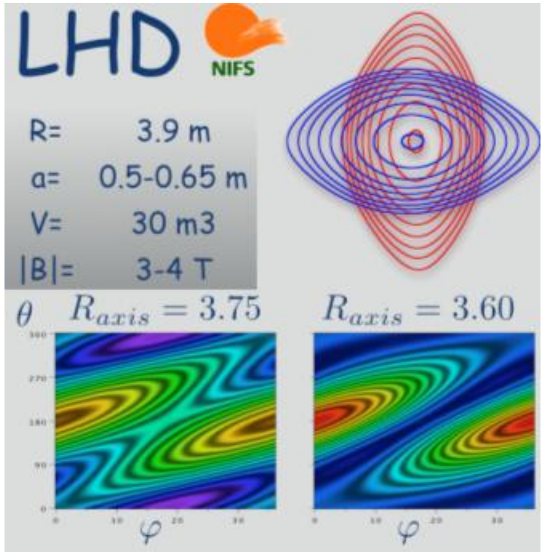
www.hsx.wisc.edu

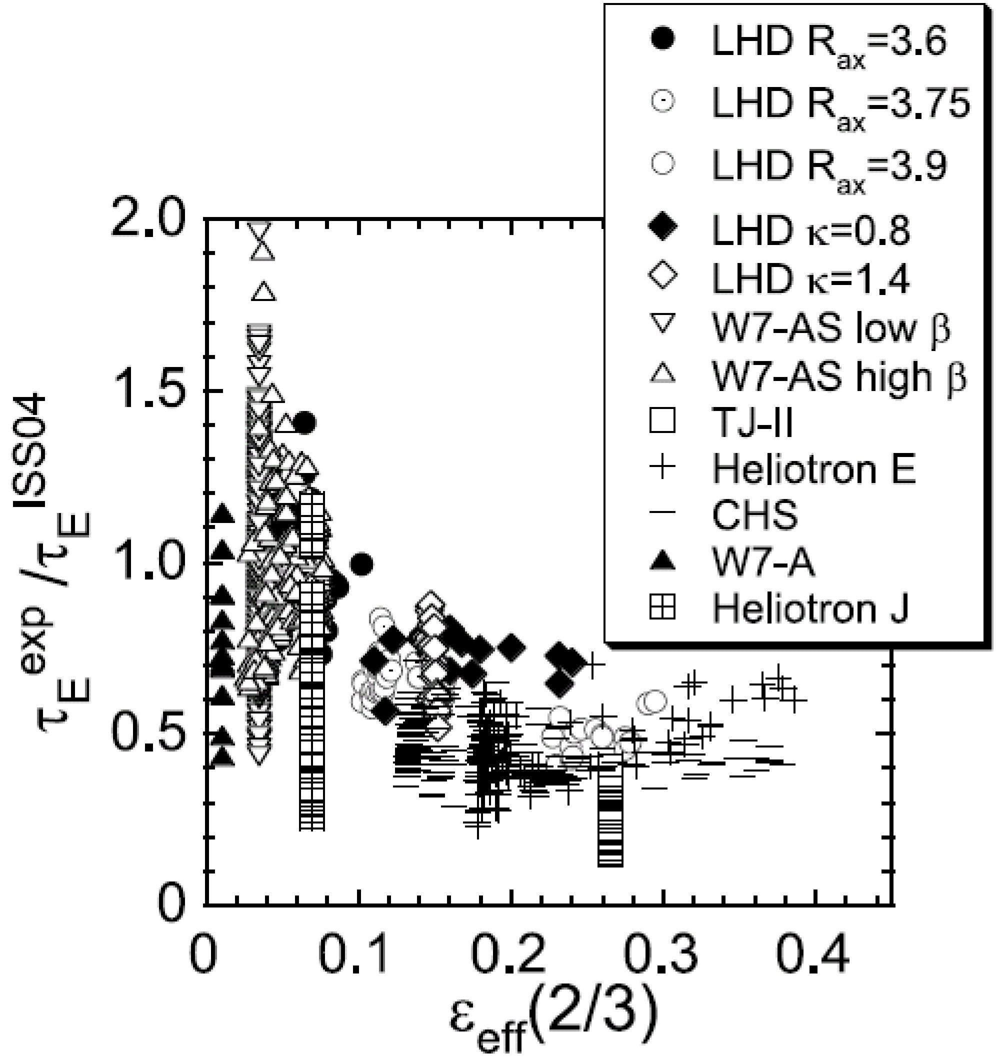


from Canik et al., PRL 98 (2007) 085002

Effect of quasi-symmetry on confinement (minimize ε_{eff})

classical





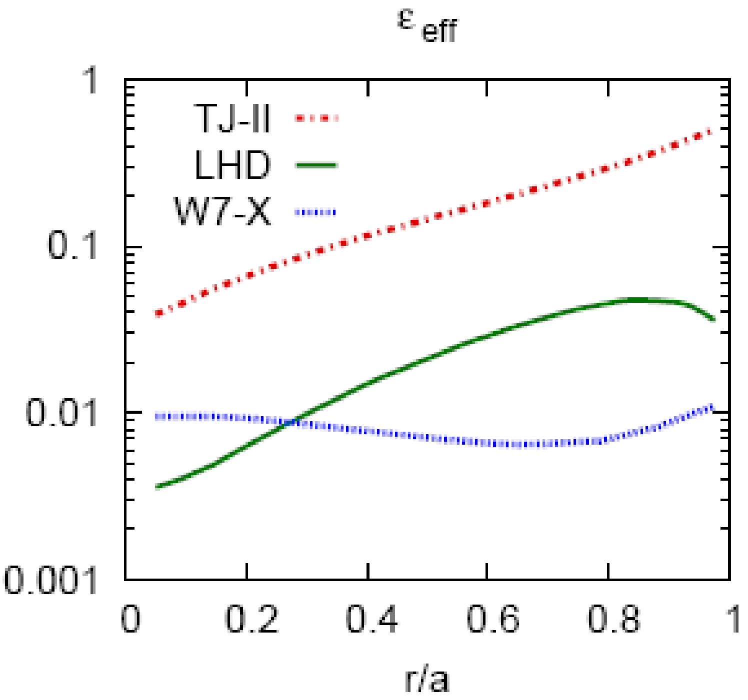
ε_{eff} becomes smaller with

- reducing R
- increasing κ

from Dinklage et al FST 51 (2007) 1

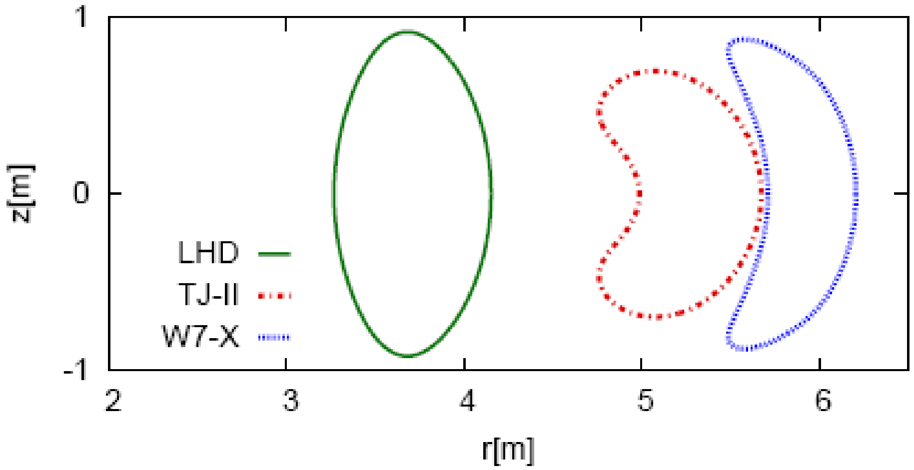
Effect of neoclassical optimization on performance

Comparison of TJ-II, LHD, W7-X



Assuming

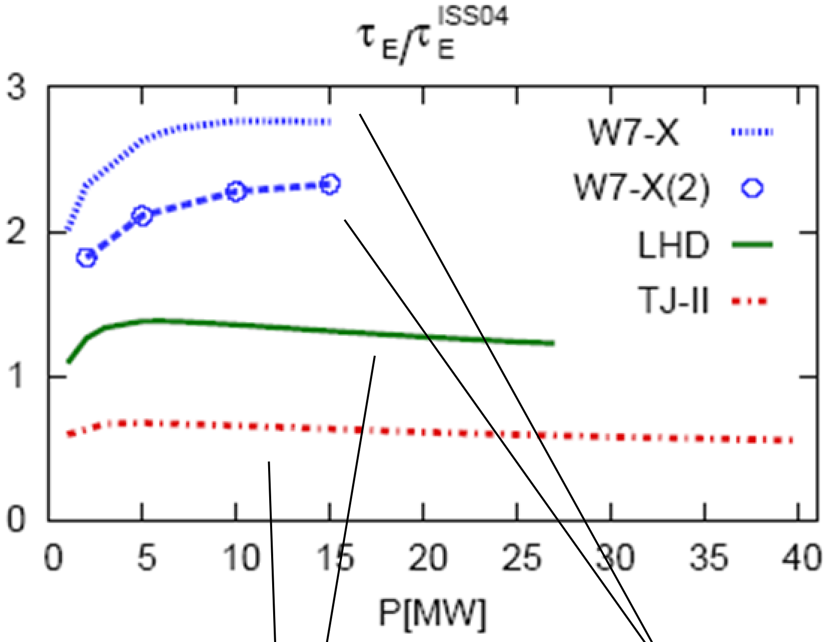
- Neoclassical transport
- Similar heating (ECRH)
- Same volume
- B = 2.5 T, n(0) = 10^{20} m^{-3}
- Simple model for turbulent transport (at the edge)



Y. Turkin et al., PoP, 2011

Effect of neoclassical optimization on performance

Effect on confinement

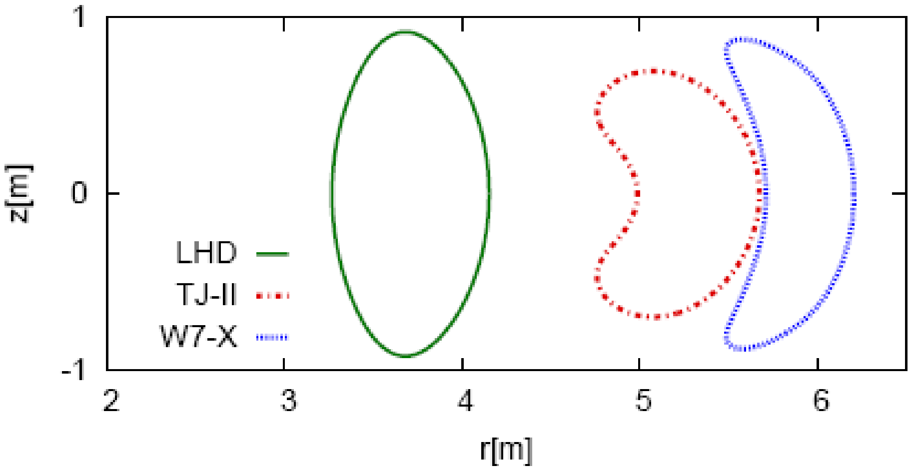


Variation of turbulent transport

$T^{7/2}$ dependence prevails

Assuming

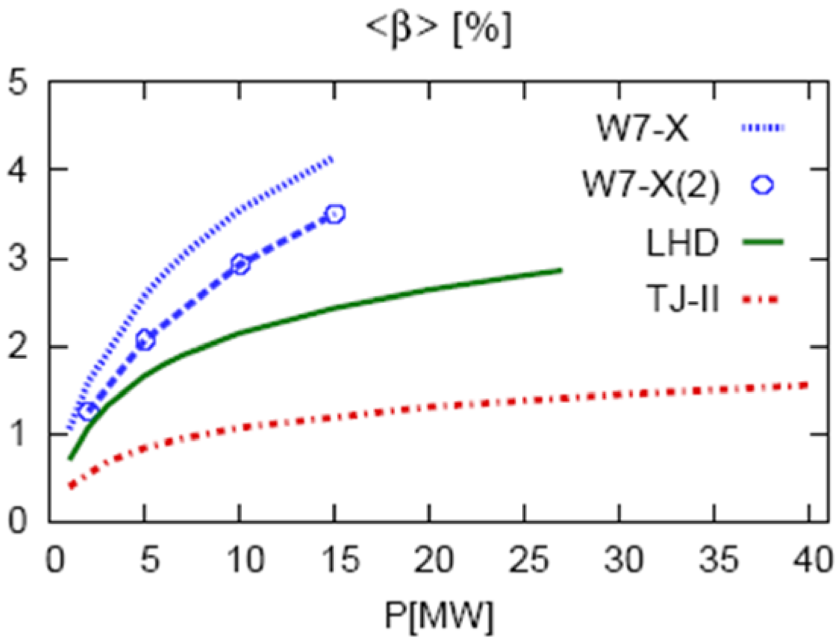
- Neoclassical transport
- Similar hearing (ECRH)
- Same volume
- $B = 2.5$ T, $n(0) = 10^{20}$ m^{-3}
- Simple model for turbulent transport (at the edge)



Y. Turkin et al., PoP, 2011

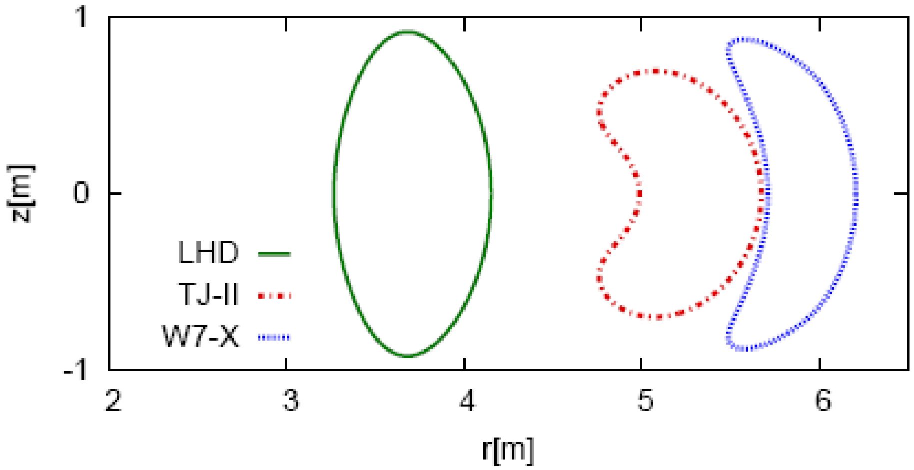
Effect of neoclassical optimization on performance

Effect on achievable β



Assuming

- Neoclassical transport
- Similar hearing (ECRH)
- Same volume
- B = 2.5 T, n(0) = 10^{20} m^{-3}
- Simple model for turbulent transport (at the edge)



Y. Turkin et al., PoP, 2011

Summary – transport effects

- Plasma core dominated by neoclassical transport
  - depending on temperature and configuration ($D \sim \varepsilon_{eff}^{\,3/2} T^{7/2}$)
  - interesting question: what happens to the relation between turbulent and neoclassical transport if ε_{eff} is very small (optimized stellarator)

- Too large effective ripple prohibits large β-values (at reasonable heating power)
  - stellarator optimization aims at minimizing ε_{eff}

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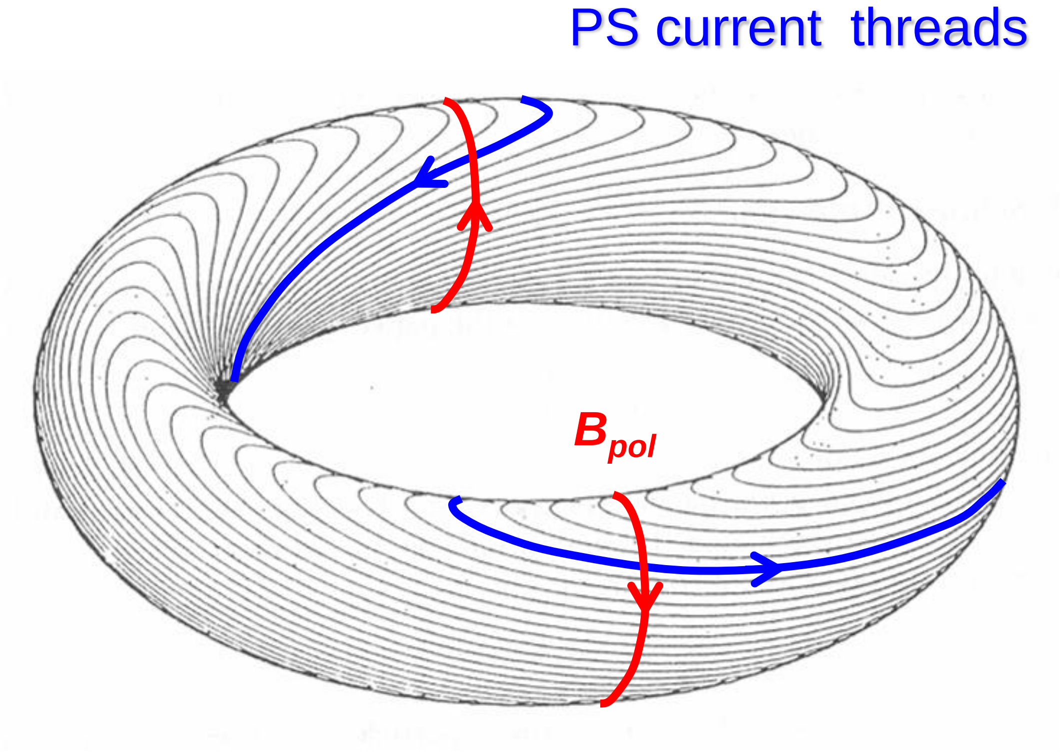
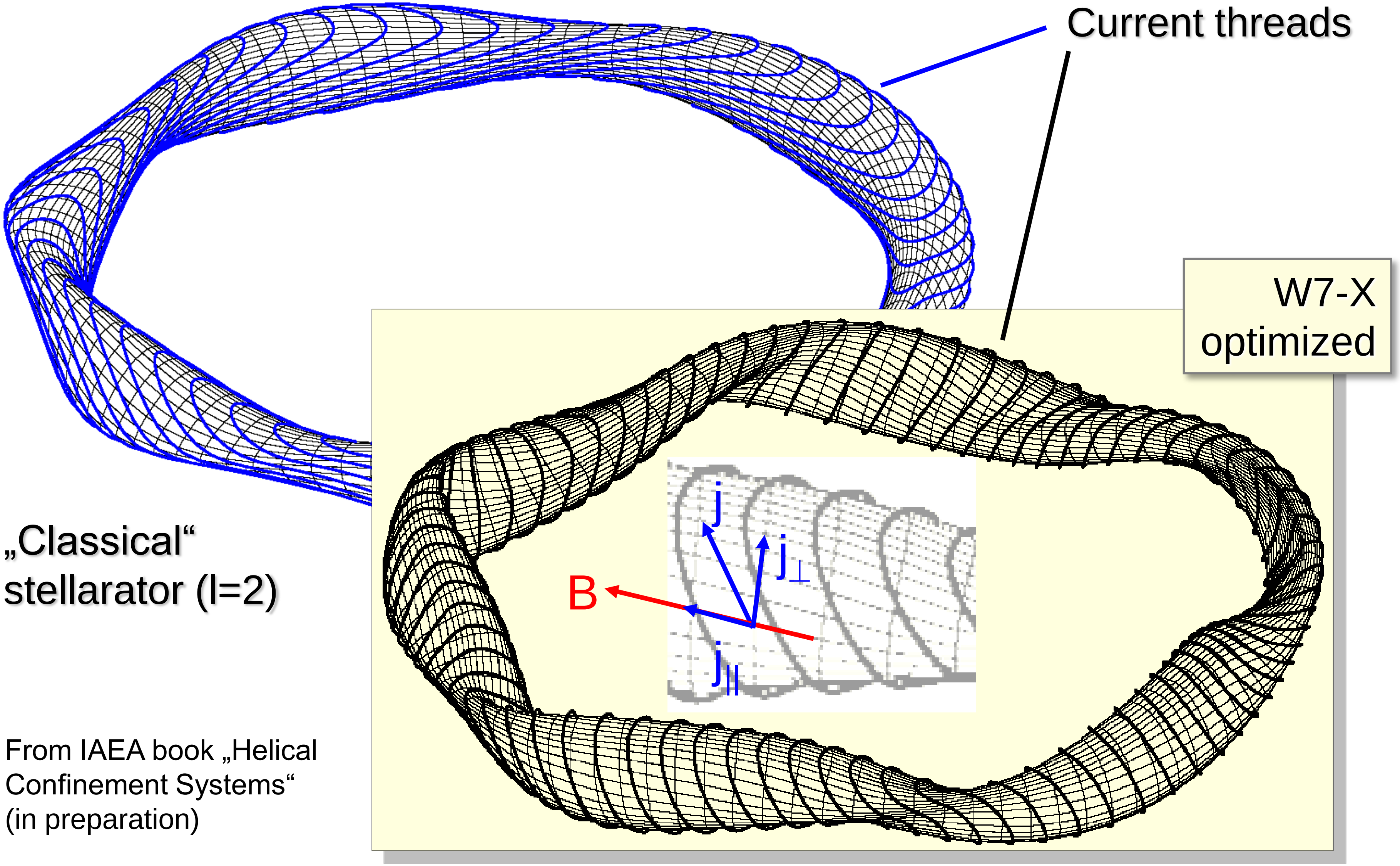
Pfirsch-Schlüter currents in tokamaks



Pfirsch-Schlüter currents weaken the poloidal field at the inner side of the torus and strengthen it at the outer side

→ Shafranov shift

Pfirsch-Schlüter currents in stellarators



Current threads

W7-X
optimized

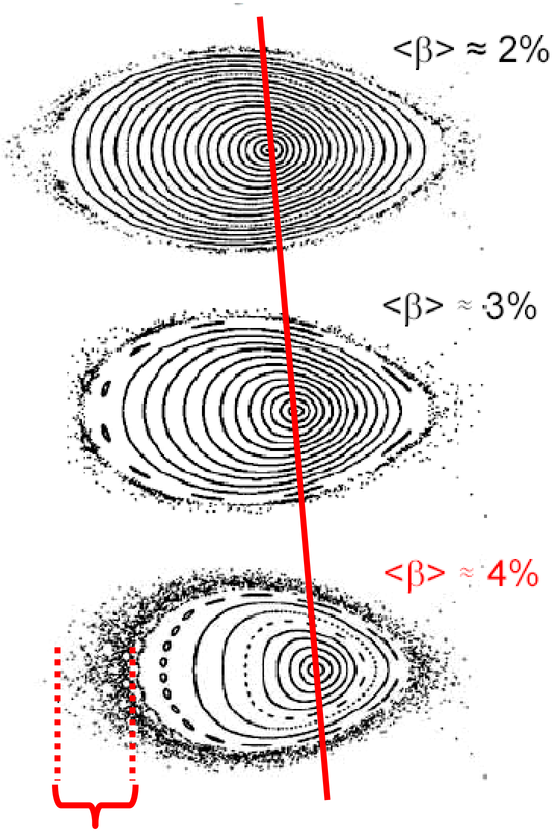
„Classical“
stellarator (l=2)

From IAEA book „Helical
Confinement Systems“
(in preparation)

Equilibrium effects in LHD and comparisons to W7-X

LHD

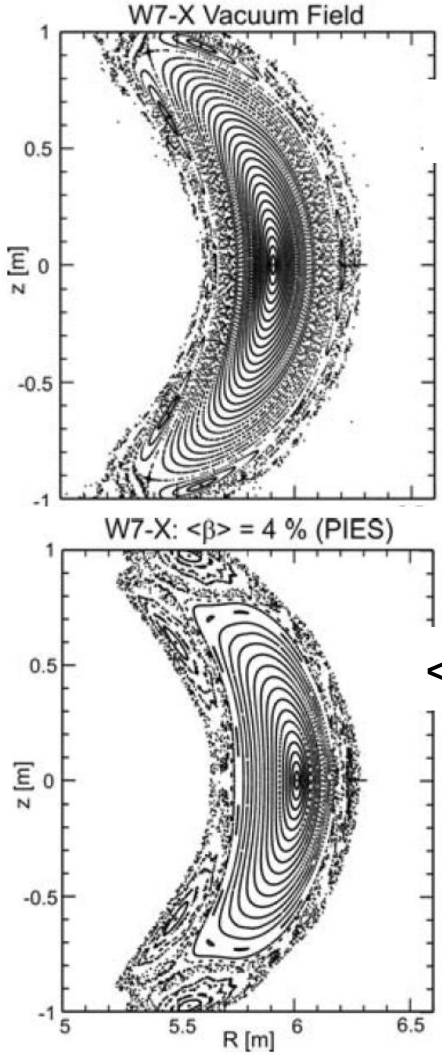
Shift of magnetic axis



Increasing ergodization of the plasma boundary
Reduction of the confining volume

from Suzuki et al.

W7-X



<β> = 0

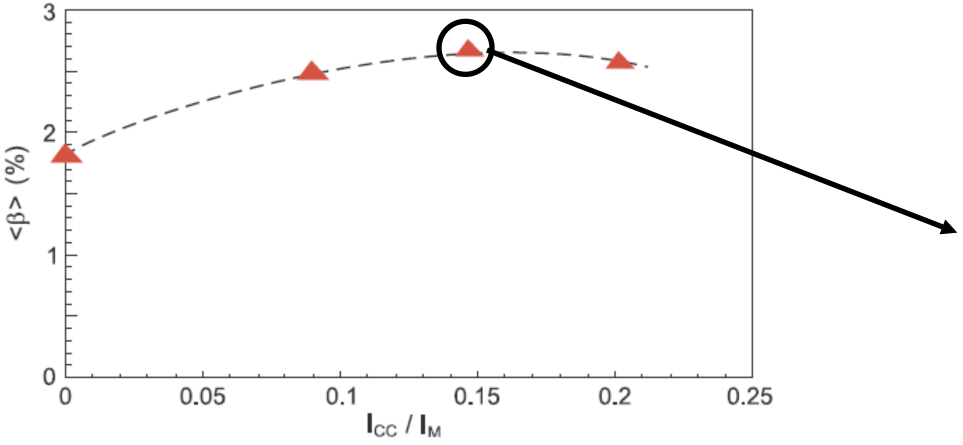
<β>=4%

Loss of confinement volume ~ 20%

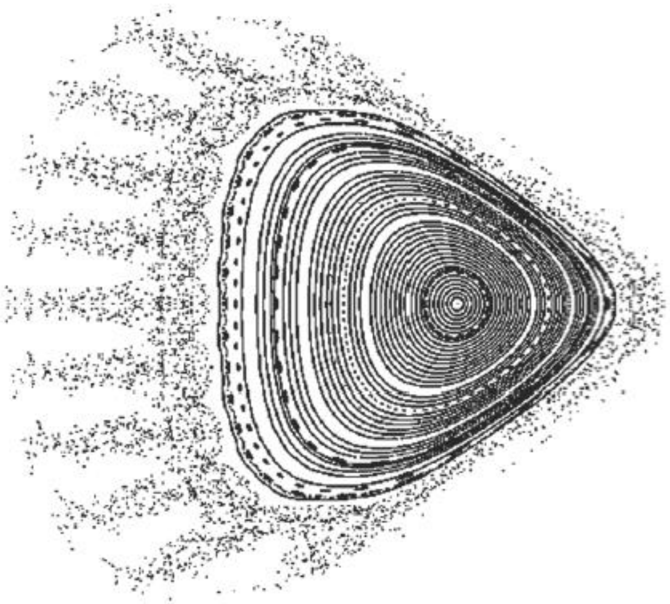
M. Drevlak et al., NF 45 (2005) 731

Equilibrium effects observed in W7-AS

Evidence that in W7-AS high β (low B) regimes β-limit is in fact equilibrium limitation



Achievable <β> versus divertor coil current normalized to modular coil current (W7-AS)



PIES simulation of W7-AS

Perturbation by divertor coils improves achievable <β>

→ Field line diffusion coefficient determines β-limit

from Reiman et al., NF 47 (2007) 572

Summary – equilibrium effects

- In stellarators the equilibrium pressure limit is more important than in tokamaks
  - rather high ballooning limit possible (evidence from LHD, design basis of W7-X, $\langle\beta\rangle = 5\%$)
  - loss of effective confinement volume due to tendency to form stochastic region at the plasma edge at high β

- Pfirsch-Schlüter (PS) currents and resulting Shafranov shift cause stochastization
  - minimizing PS currents in W7-X
  - alignment of PS with the diamagnetic current

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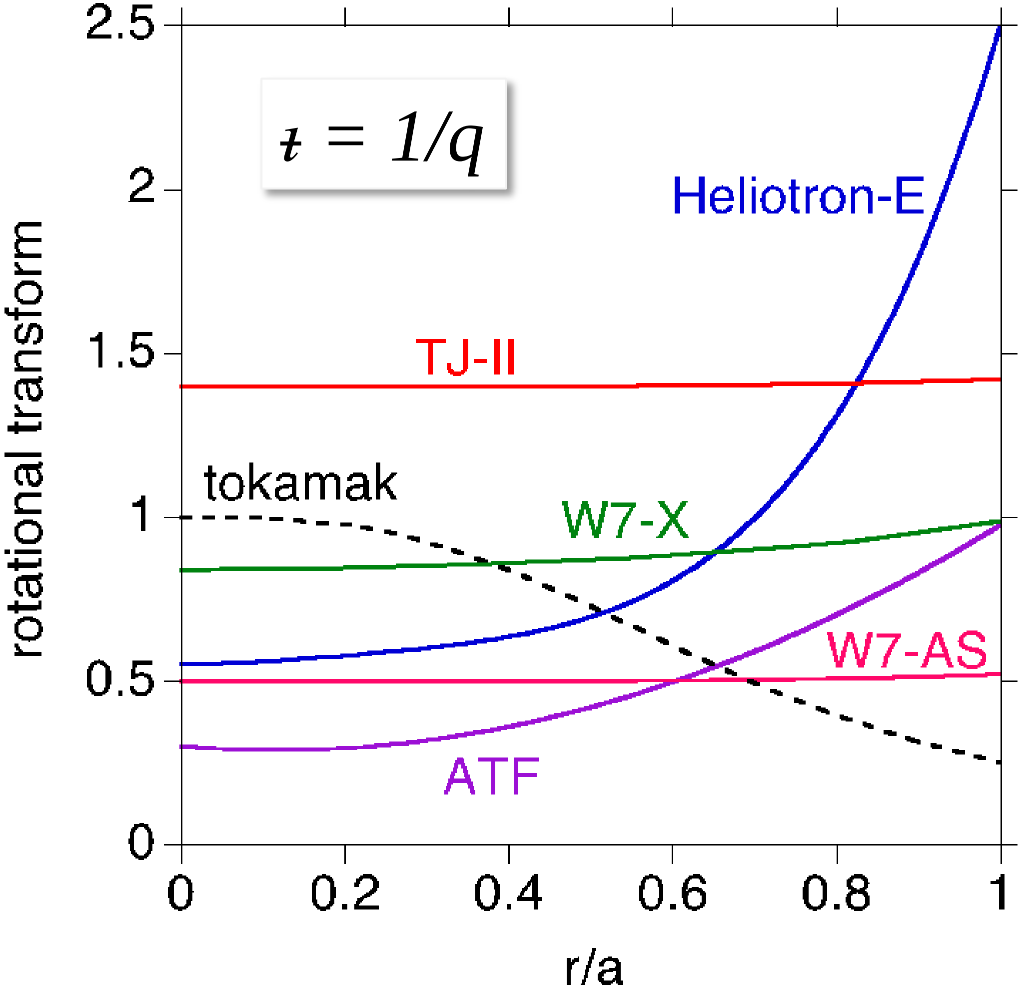
Profiles of rotational transport



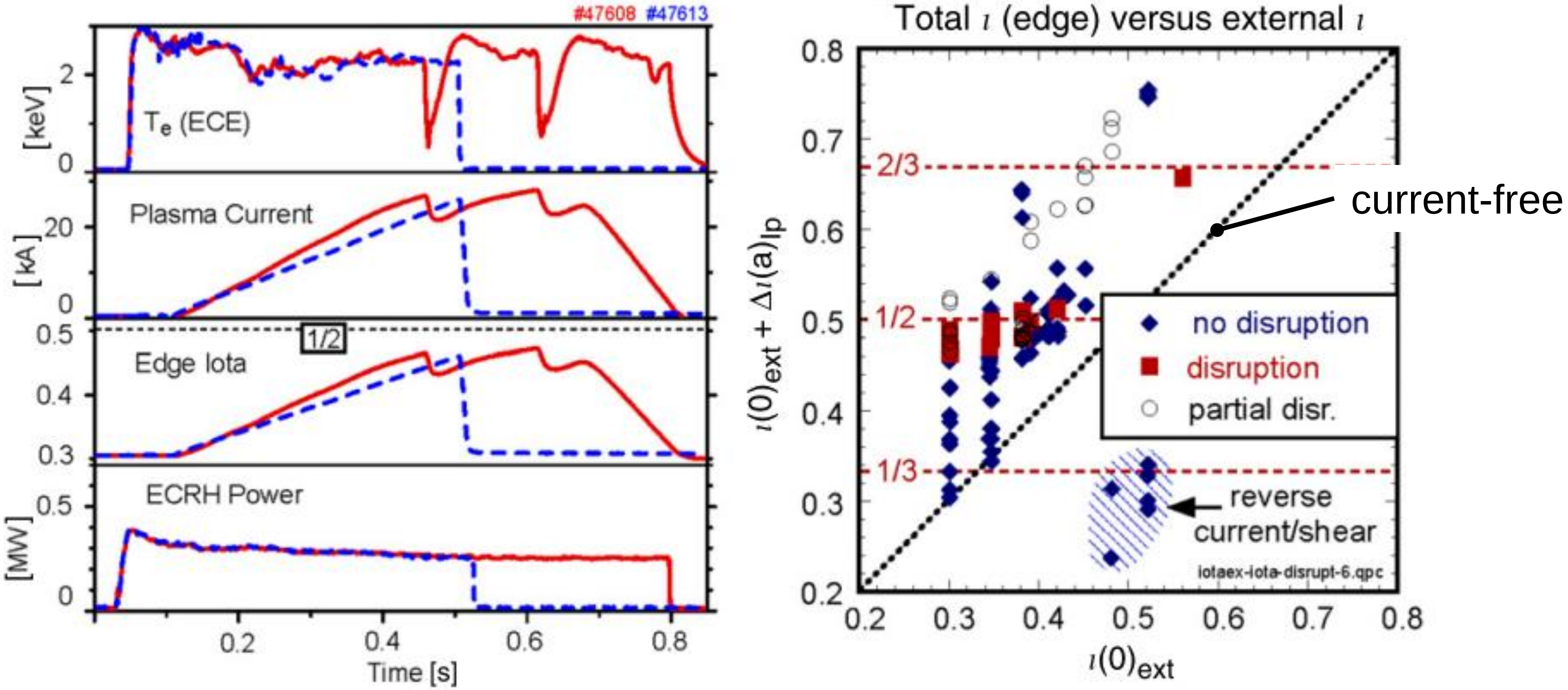
High shear
(LHD,
Heliotron-E, ...)

Low shear (W7-X,
TJ-II, ...)

Compared to tokamaks, stellarators almost “always” have reversed magnetic shear

→ Stabilizing for neoclassical tearing modes

Tearing modes associated with auxiliary plasma currents



W7-AS experiments motivated by emergence of compact current carrying quasi-axis-symmetric stellarators (NCSX)

– Current drive provides contribution $\Delta\iota(a)$ to total rotational transform

– Most unstable at $\iota(a) = \iota_{ext} + \Delta\iota(a) = ½$, $q(a) = 2$ and $\iota_{ext} < 0.5$ ($\Delta\iota(a)/\iota_{ext} > 0.15$)

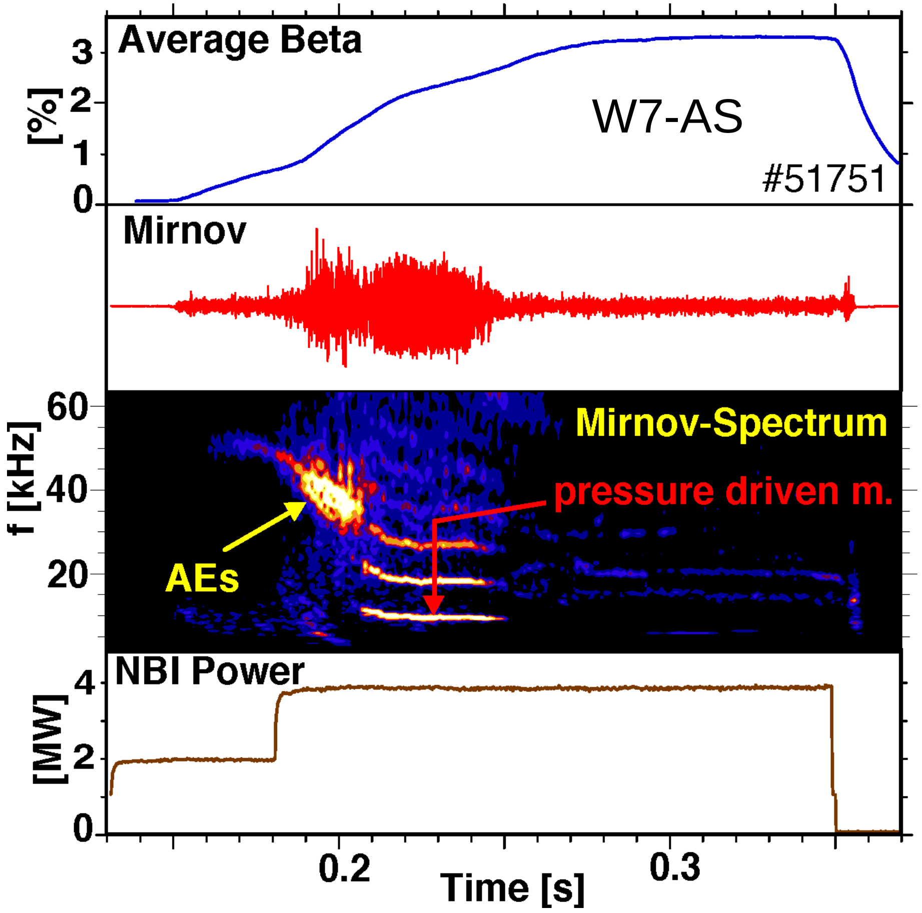
– Disruption events provoked by tearing modes (m=2 etc.)

Hirsch et al., PPCF 50 (2008) 053001

Quiescent behaviour at stability limit

Stability limits

- Often observed as a saturation of achievable plasma pressure



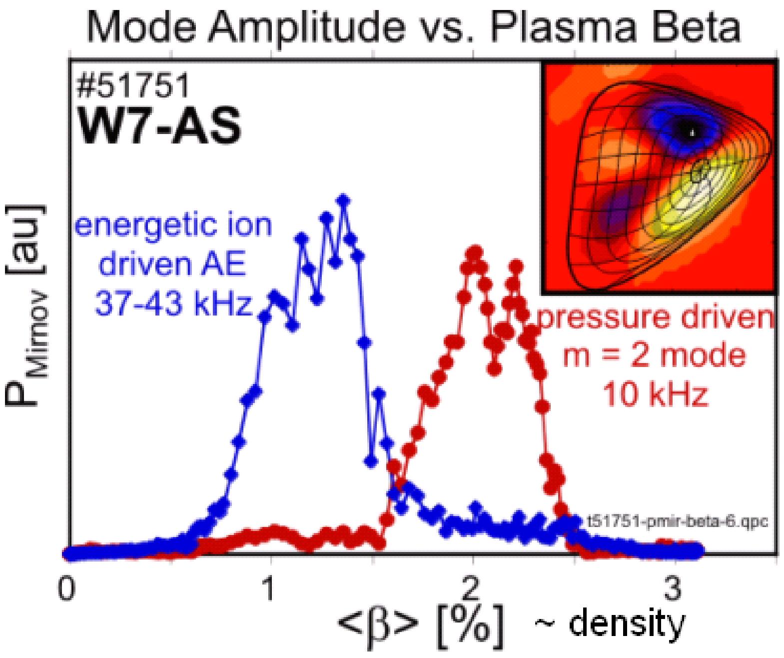
- Only weak effect of low-n pressure driven modes below $\langle\beta\rangle = 2.5$ %
- Alfvén Instabilities restricted to lower density (higher fraction of fast ions)
- Formation of magnetic well stabilizes pressure driven modes (self-healing of instabilities)

from A. Weller et al.

Quiescent behaviour at stability limit

Stability limits

- Often observed as a saturation of achievable plasma pressure



- Only weak effect of low-n pressure driven modes below <β> =2.5 %
- Alfvén Instabilities restricted to lower density (higher fraction of fast ions)
- Formation of magnetic well stabilizes pressure driven modes (self-healing of instabilities)

from A. Weller et al.

Summary – stability limits

- No disruptions (unless large ohmic, bootstrap or driven currents exist)

- Most important in stellarators
  - interchange type modes (∇p driven)
  - stabilized by magnetic shear and magnetic well
  - favourable „reversed“ shear by external field

- Often observed as a saturation of achievable plasma pressure

- Energetic particle driven MHD instabilities, potentially dangerous (α-losses in reactor), wall damage

Contents

- The stellarator concept
  - Some stellarators which are shown in this presentation
- Operational limits
  - **Transport effects**
  - **Equilibrium effects**
  - **Stability limits**
  - Density limits
- Summary

The stellarator can operate beyond the Greenwald limit

Importance of high density

– maximizing fusion power

$P_f = \int n^2 <\sigma v> E_f \, dV \sim p^2 <\sigma v>/T^2 \sim \beta^2 \cdot B^2$

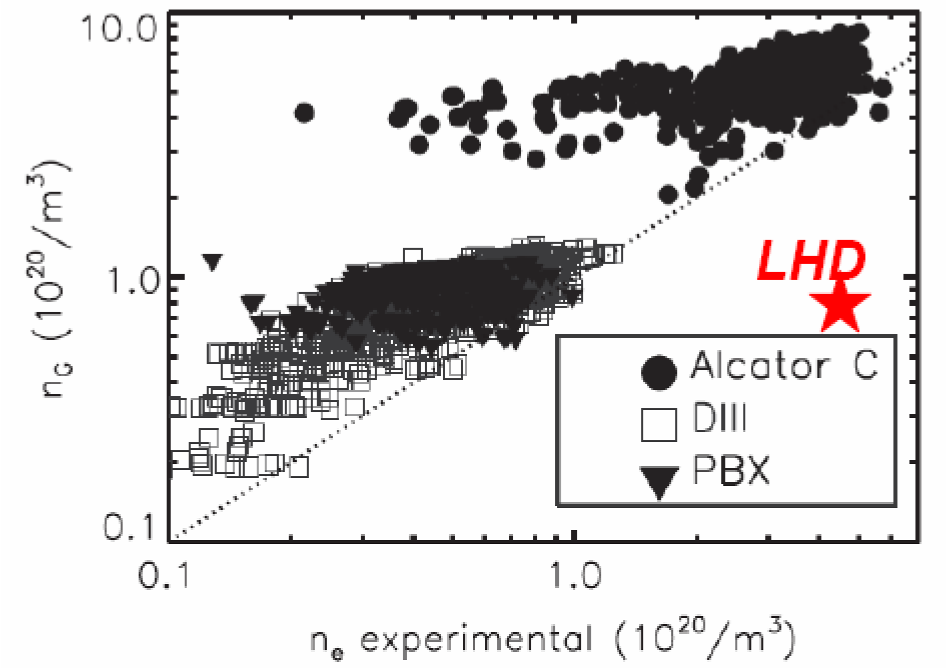
$<\sigma v>/T^2 \approx$ const. at 10 keV

– stay in optimum fusion reaction range even at higher β

– current drive efficiency is no issue

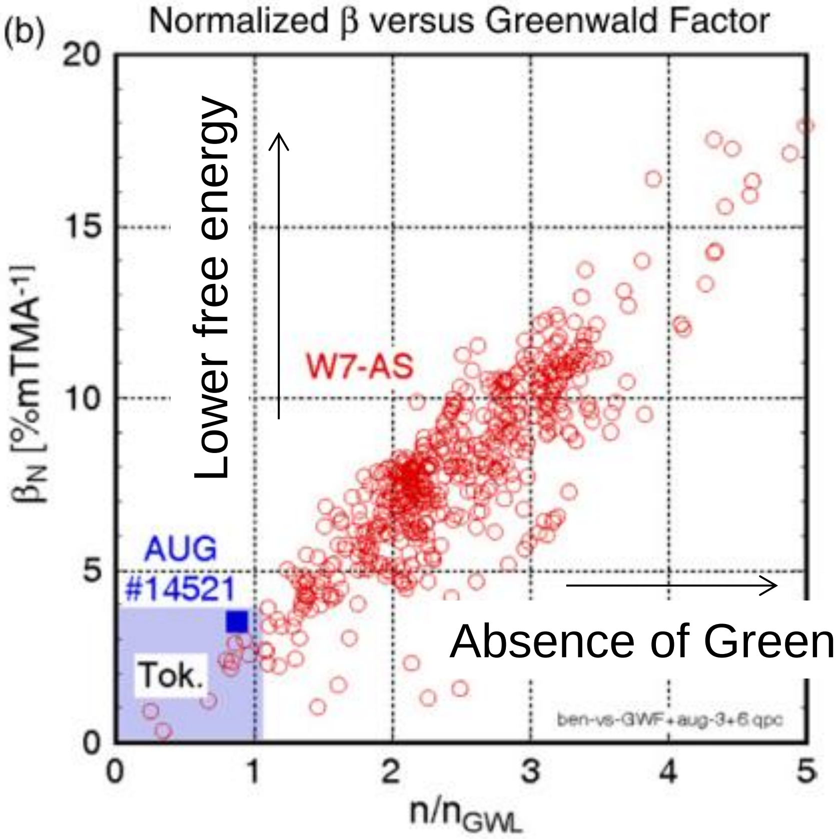
– low population of fast particles (may actually become necessary)

$$I_p = 5 \frac{a^2 B}{R} \cdot \frac{1+\kappa^2}{2} \cancel{t_a}$$



$$n_{GW} \propto \frac{I_p}{\pi a^2}$$

Examples (W7-AS, LHD)



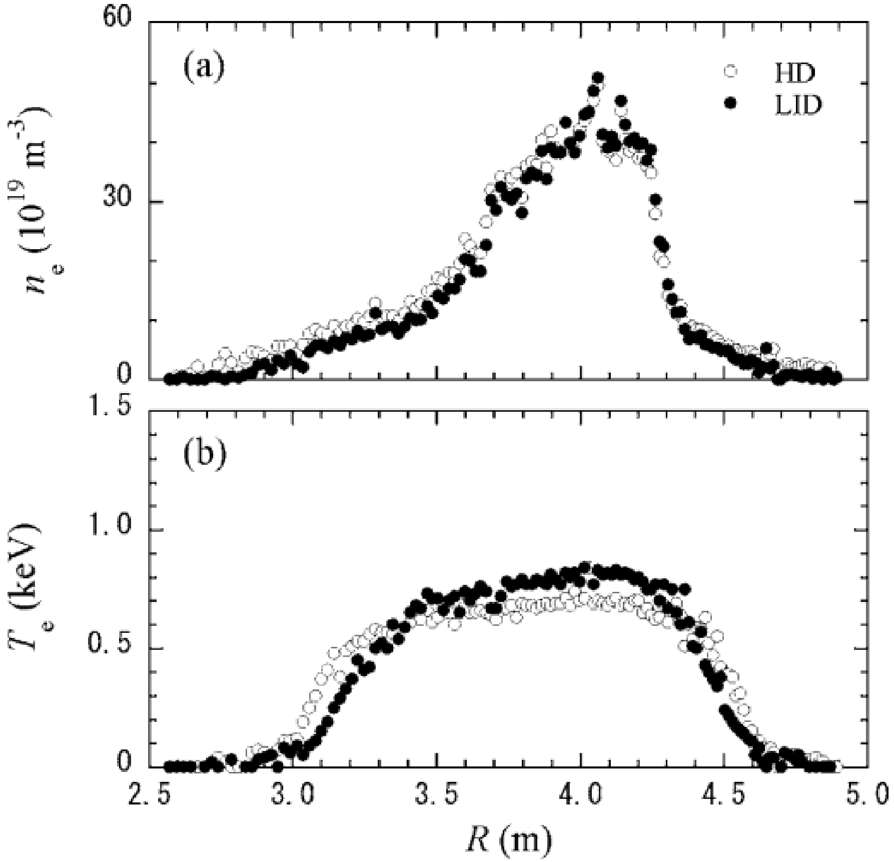
Lower free energy

Absence of Greenwald limit

$$I_{\text{eq}} = f(\varepsilon)t(a)(5a_{\text{eff}}^2 B/R)[1+\kappa^2(1+2\delta^2-1.2\delta^3)]/2\kappa$$

Hirsch et al., PPCF 50 (2008) 053001

LHD SDC

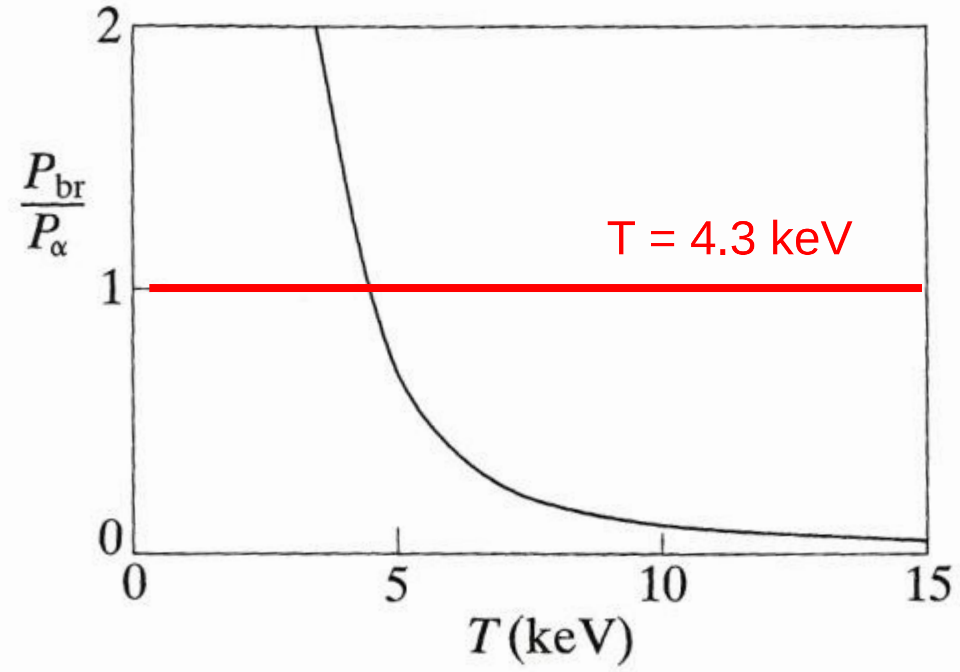


Morisaki et al., PoP 14 (2007) 056113

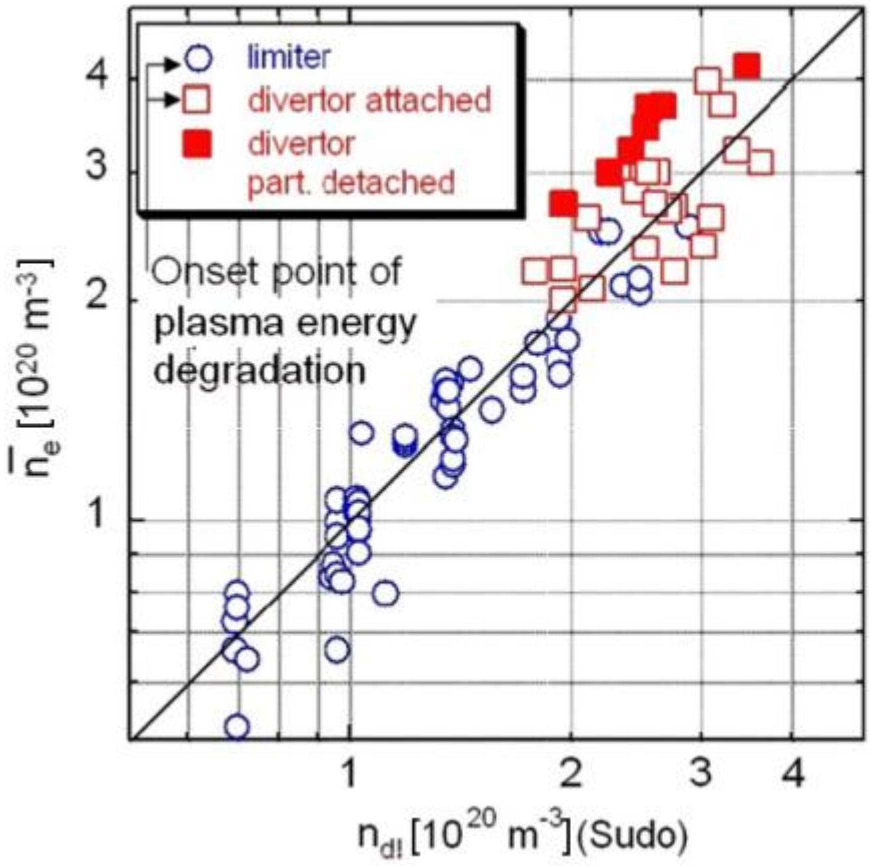
→ Ultimately density is limited by bremsstrahlung

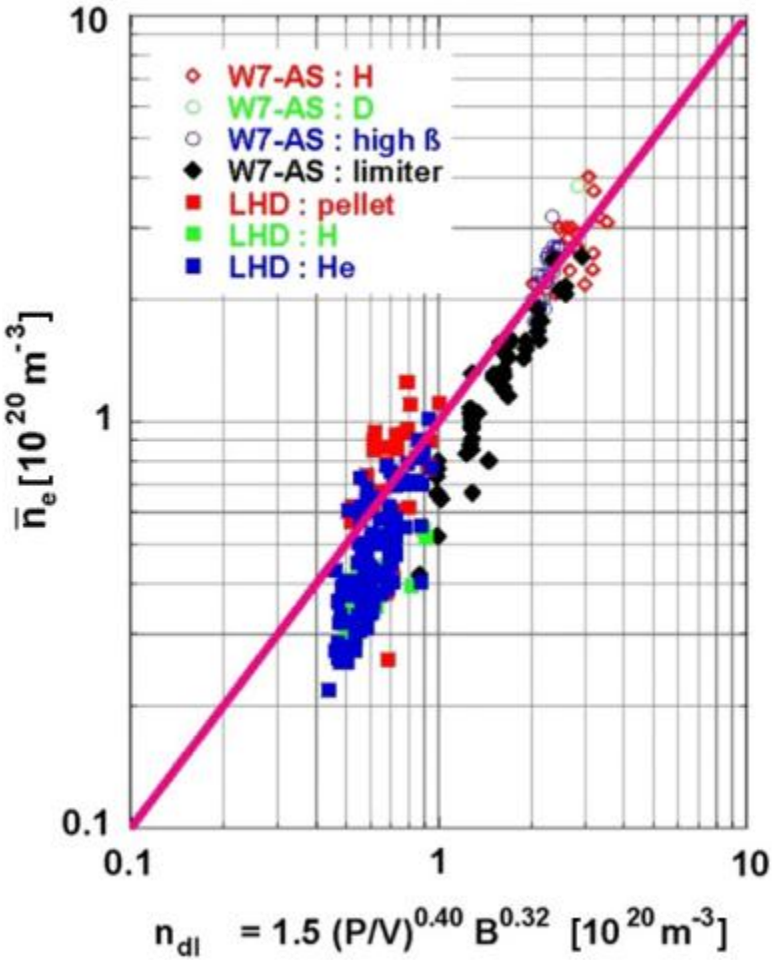
Bremsstrahlung limit in a pure hydrogen plasma

- Ultimately the density is limited by the bremsstrahlung exceeding the heating power (fusion power)
- P_{br}/P_{α} is solely a function of temperature (assuming small dilution and Z not too large, here for a pure hydrogen plasma)



The radiation limit (Sudo limit)





Sudo density limit scaling (proportionality factor depends on dominant impurity species)

$$n_{dl} \propto \sqrt{\frac{PB}{a^2 R}}$$

Hirsch et al., PPCF 50 (2008) 053001
Sudo et al., NF 30 (1990) 11
Itoh K, Itoh S, JPSF 57 (1988) 1269

Density limit can lie below stability or equilibrium limits

Impurity control is an issue (and in particular in stellarators)

Plasma interacts with wall

– Particles fluxes ($\Gamma_{in} = \Gamma_{out}$)

– Heat fluxes ($Q_{in} = Q_{out}$)

Unlike in tokamaks temperature screening does not exist in stellarators

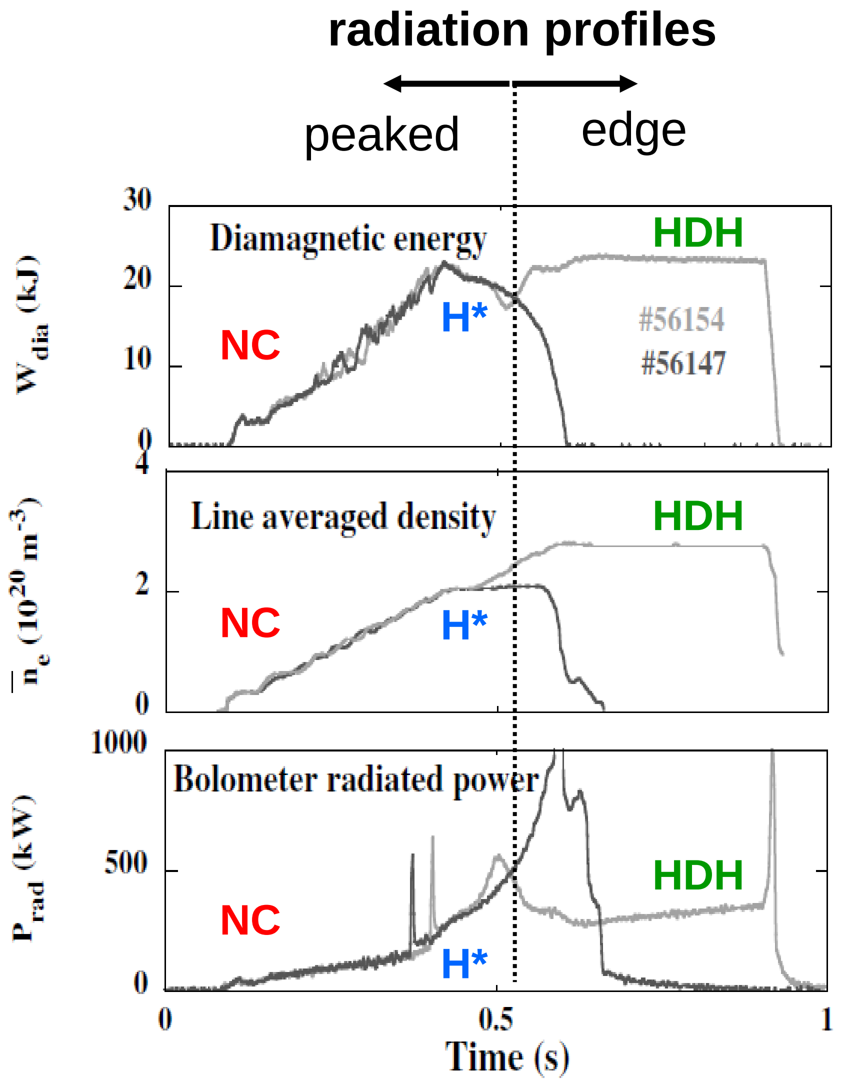
The concentration of impurities has to be low enough to avoid too high

– radiation in the plasma centre

– dilution of the plasma fuel

In addition, certain types of H-mode aggravate the problem

→ HDH mode see talk of T Sunn Pedersen



Example W7-AS

Summary – density limit

- The Greenwald density limit is not observed in stellarators; high density operation
  - required for operating at optimum fusion reactivity (at 15 keV)
  - Reduces fast ion instability drive
  - serves easier plasma solutions for the divertor

- Stellarators more or less follow the Sudo density limit scaling
  - can be interpreted as a radiation limit
  - depends on dominant impurities
  - in ion-root regime (ambipolarity condition) neoclassical transport (thermodynamic forces) are predicted to support impurity accumulation

- Ultimately the density is limited by bremsstrahlung, but impurity radiation can limit density at values even below stability and equilibrium limits

Summary

- Stable high- operation up to <β> = 5% seems feasible

- The Greenwald density limit does not exist
  - density limited by radiation

- In general stellarators have rather “soft” operational limits
  - no disruptions
  - equilibrium and stability boundaries take the form of a confinement saturation

- Achieving high β requires
  - optimized neoclassical transport (minimization of effective ripple of the magnetic field configuration) to achieve high β at high temperatures
  - in addition to limited heating power this is expressed by the fact that in present-day experiments high β requires reduced magnetic field
  - optimized equilibrium properties

End

Summary – the stellarator concept

Going from 2D / tokamak to 3D / stellarator increases the degrees of freedom to find the most suitable magnetic field configuration

Tokamak: Large experimental flexibility in a given device by tayloring the current profile

→ Easy way to explore configurational space

→ Large plasma control effort, critical behaviour at operation boundaries

Stellarator: Little degree of freedom in a given device, but much larger choice of possible magnetic field configurations when devising a device

→ Very costly way to explore configurational space

→ Much smaller plasma control effort, benign operation boundaries

Wendelstein 7-X: Entirely new approach – ask for the best (optimized) magnetic configuration and design and build the device accordingly

A rather simple stellarator: Two pairs of coils

Columbia Non-Neutral Torus (CNT)

Columbia University, New York, USA

T. S. Pedersen:
Stellarator News 2004

about 50cm

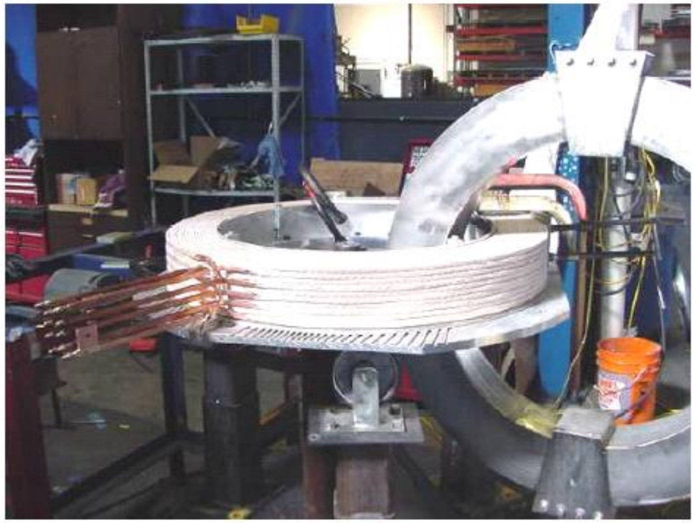
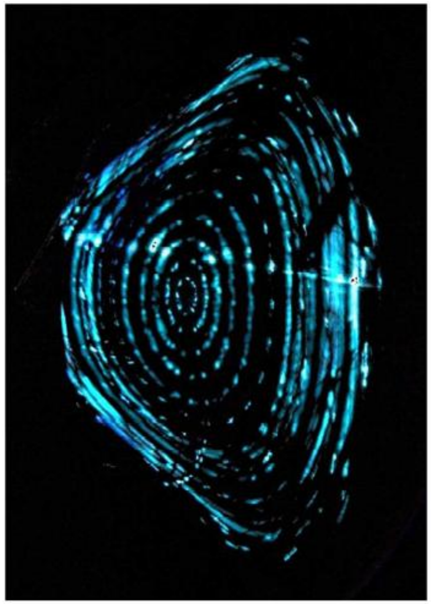


Fig. 2. The second internal coil at Alpha Magnetics after completion of winding, interlocked with the first internal coil.

Major radius: $R = 0.3$ m
Minor radius: $a = 0.1$ m
Magnetic field strength: $B = 0.2$ T
Rotational transform: $\iota/2\pi = 0.2 - 0.6$



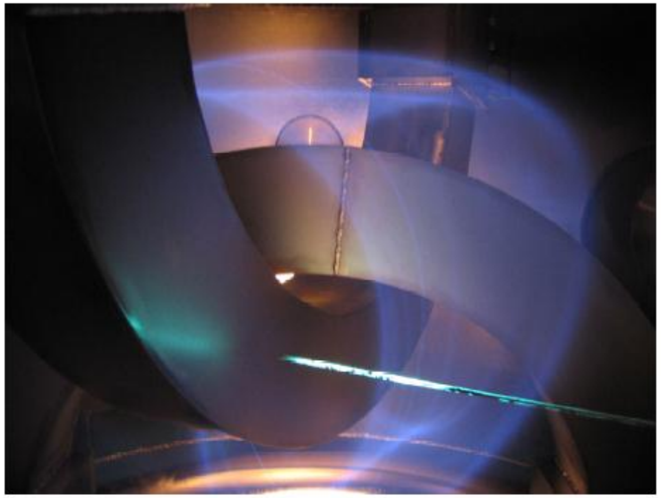


Fig. 4. A glowing magnetic surface created by collisions between an electron beam and air at 5×10^{-5} Torr. One of the phosphorescent rods is visible, lit up by electrons.

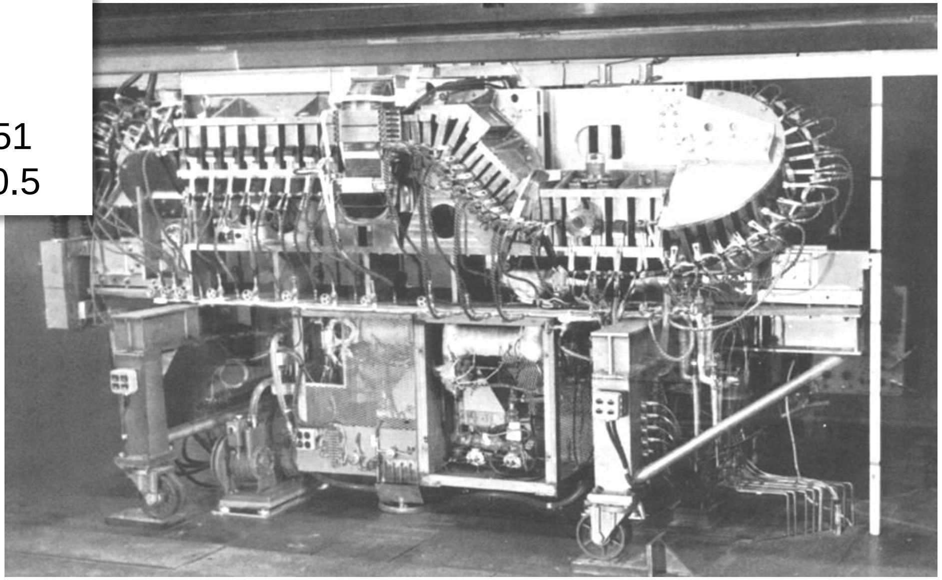
Adjustment of rotational transform
by changing the angle between two inner coils.

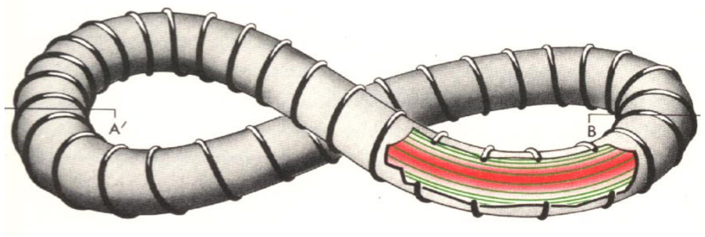
... but nested flux surfaces and a rotational transform

Some historical stellarators

Figure-8 stellarator
PPPL, Princeton, USA

Lyman Spitzer: proposed 1951
Rotational transform: $\iota/2\pi = 0.5$

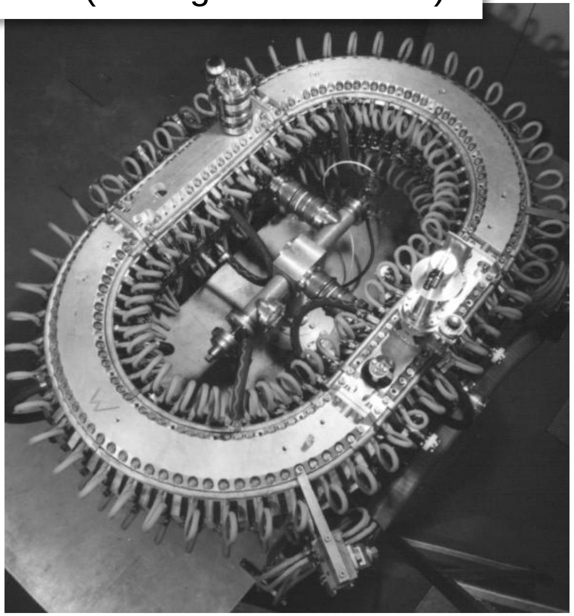




→ helicity achieved by twisting the torus and hence the magnetic field, ($\iota/2\pi$ =2)

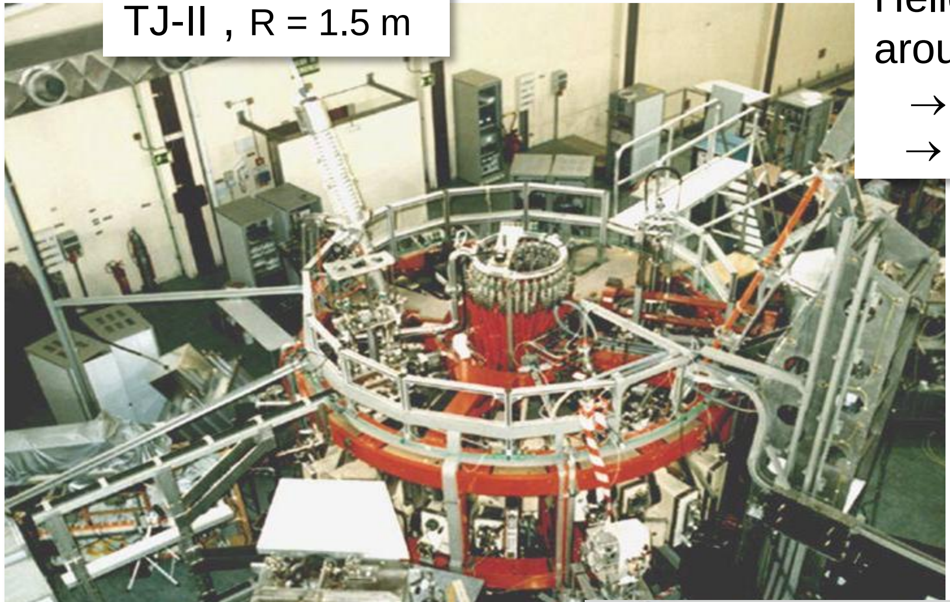
→ but rather poor confinement

W1-A
l = 3 helical windings, R = 0.35 m, a = 0.02 m, B = 1 T, Cs-plasmas (DeAngelo et al 1963)



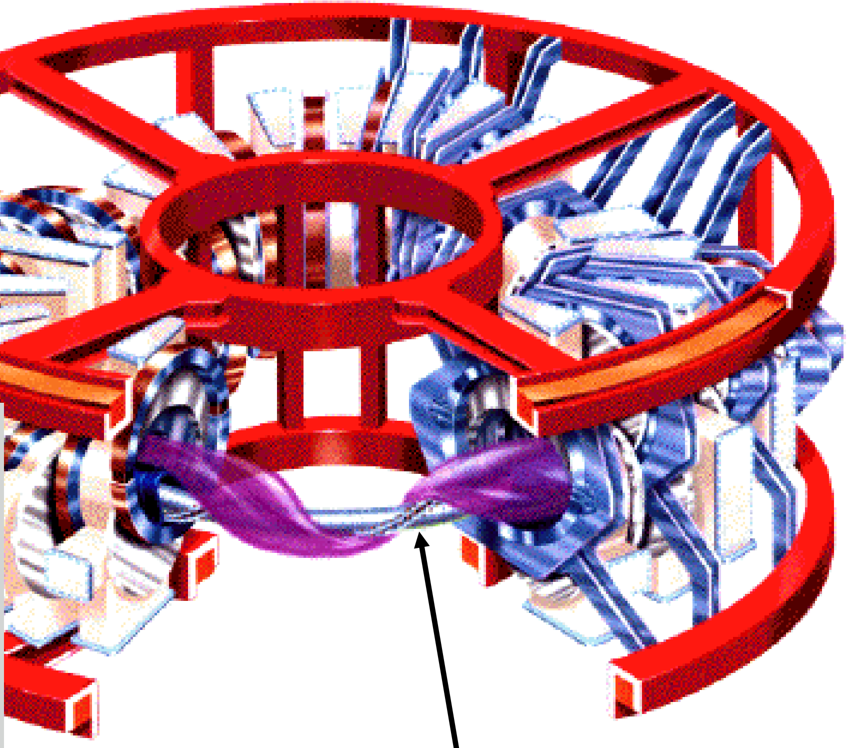
The Heliac TJ-II (Madrid, Spain)

TJ-II , R = 1.5 m

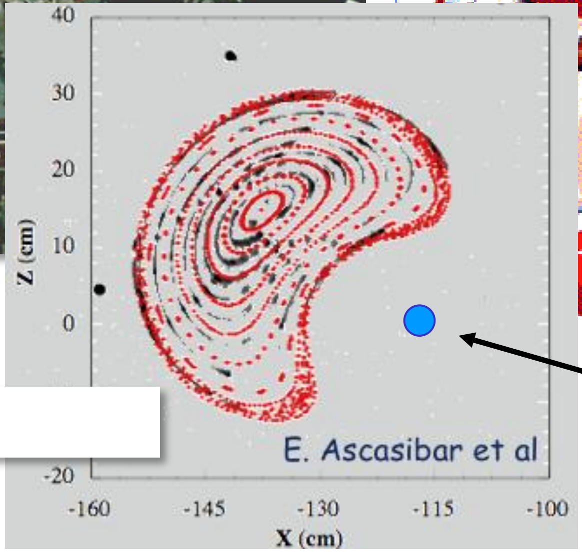


Helical component achieved by winding the plasma around a single central conductor (helical axis)

→ good access (diag. development)
→ high beta possible



Central conductor



E. Ascasibar et al

Others:
→ H1 in Canberra
→ Tohoku Heliac (JP)

Constant plasma cross section

The classical stellarator W7-A (Garching, Germany)

W7-A:
R = 2 m, a = 10 cm,
l = 2 (elliptical cross section)
m = 5; B = 3.5 T
$0.05 < \iota < 0.5$, low shear

Current in neighboured helical windings flows in opposite directions:

→ No net toroidal current in conductors

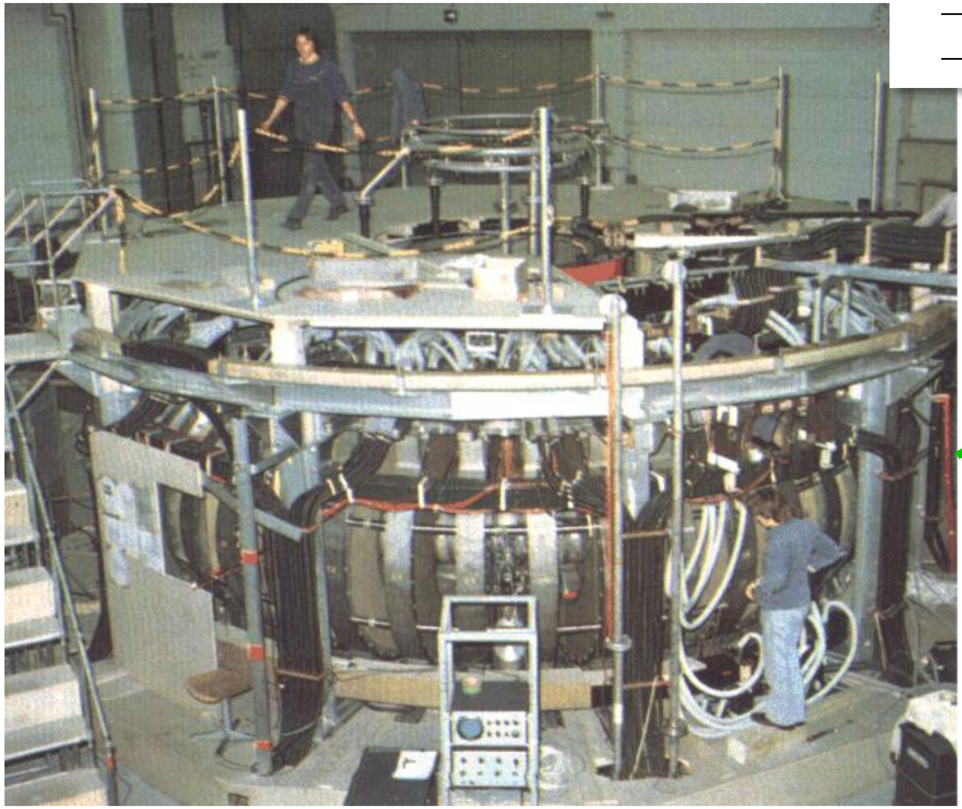
→ No currents induced in the plasma, low shear possible, $\iota < I_{hel}/I_{tor}$

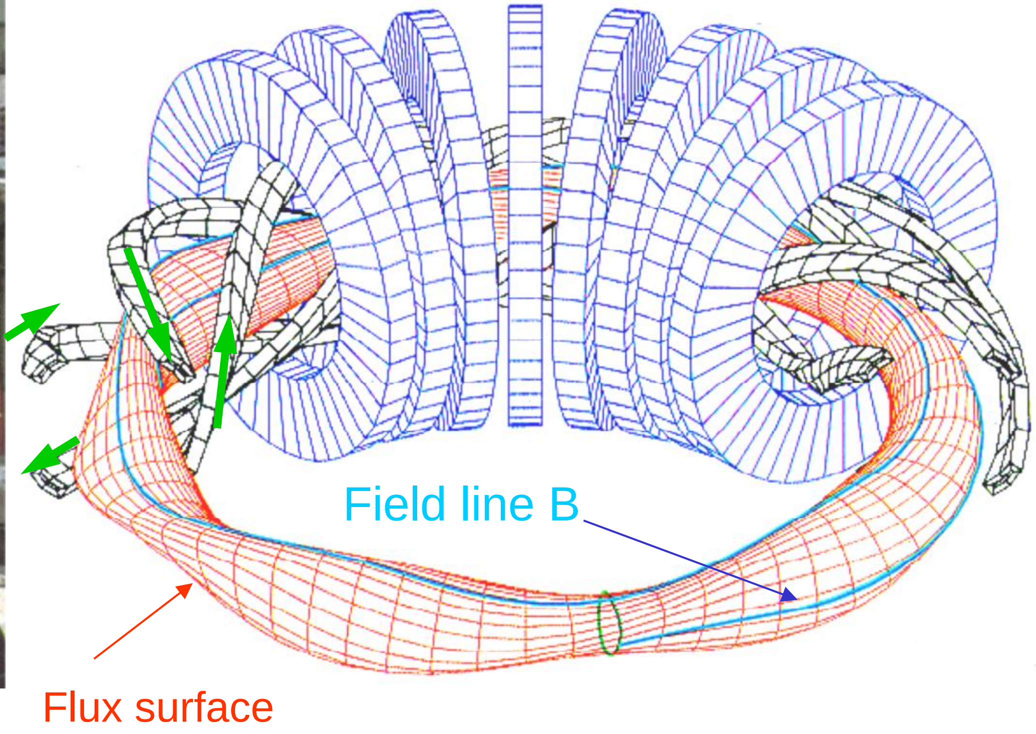
→ Strong forces between conductors may occur

W7-A achieved confinement without toroidal current with auxiliary heating (NBI, ECRH) only

→ No current driven instabilities, no disruptions

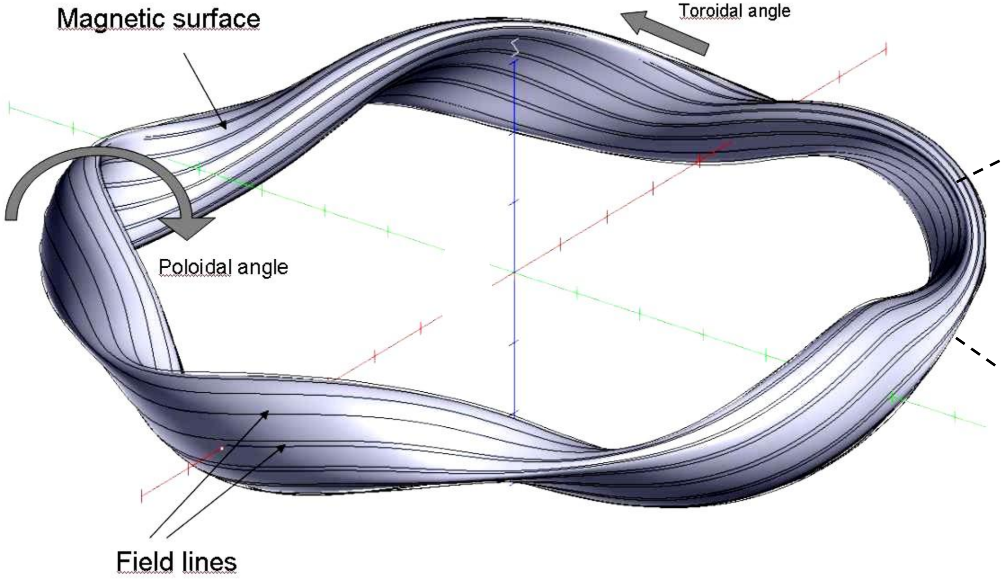
→ Density limit determined by radiation

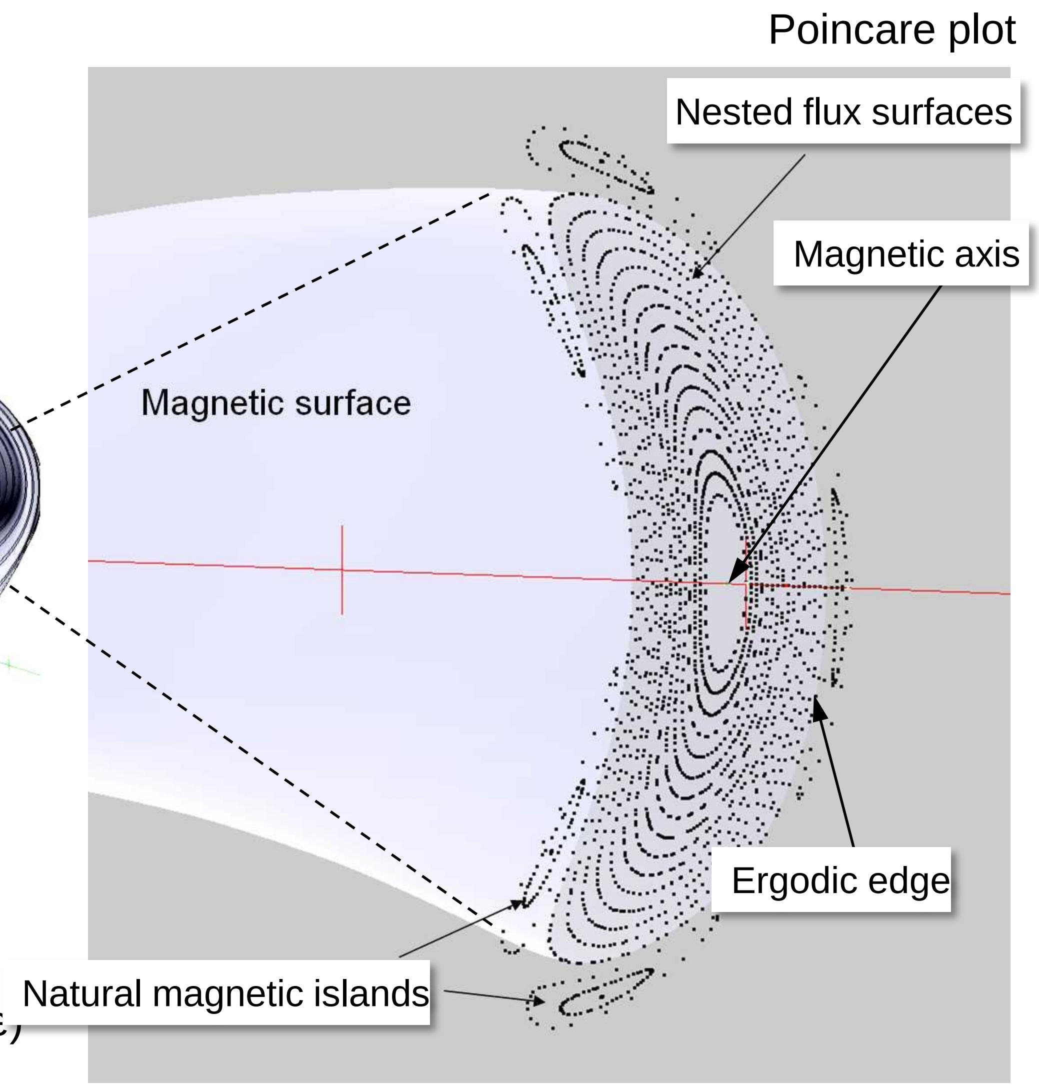




Flux surfaces

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





Rotational transform $\iota = R \langle B_\theta \rangle / r_{eff} \langle B_\phi \rangle$

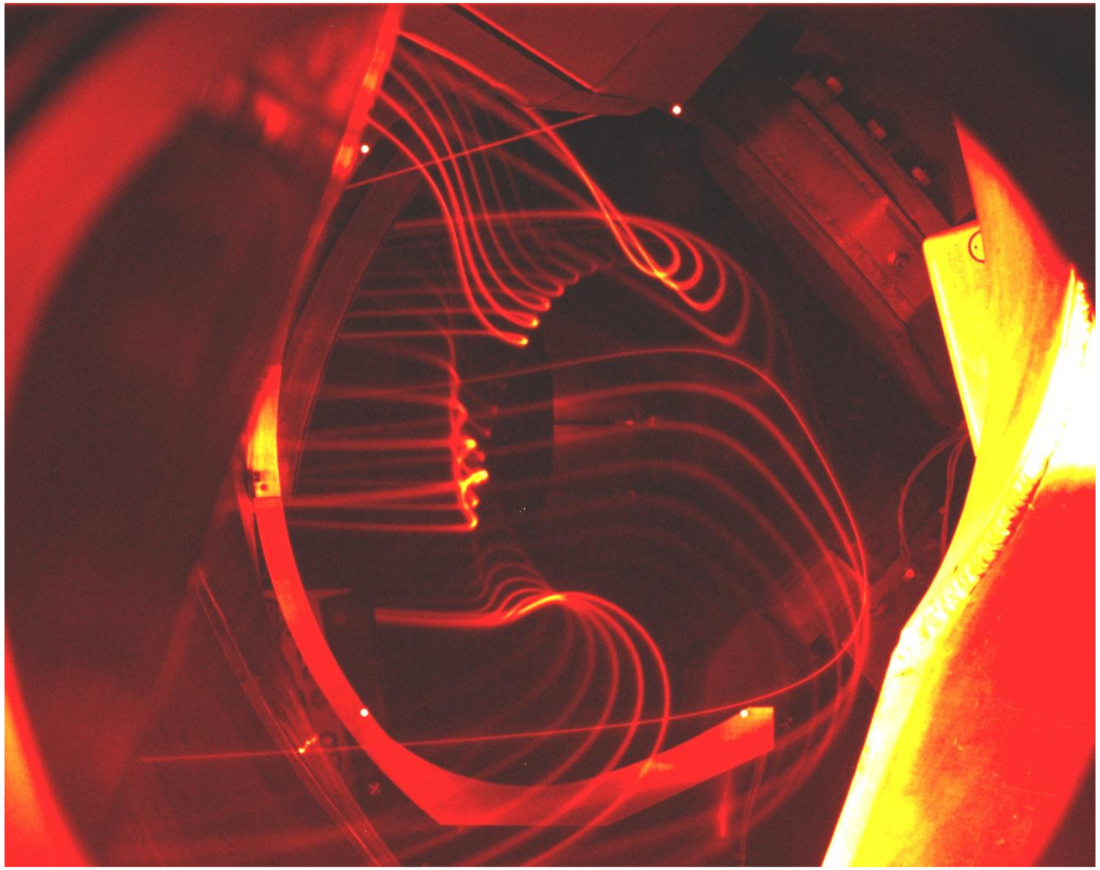
(*local* pitch angle may vary strongly on flux surface)

- 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

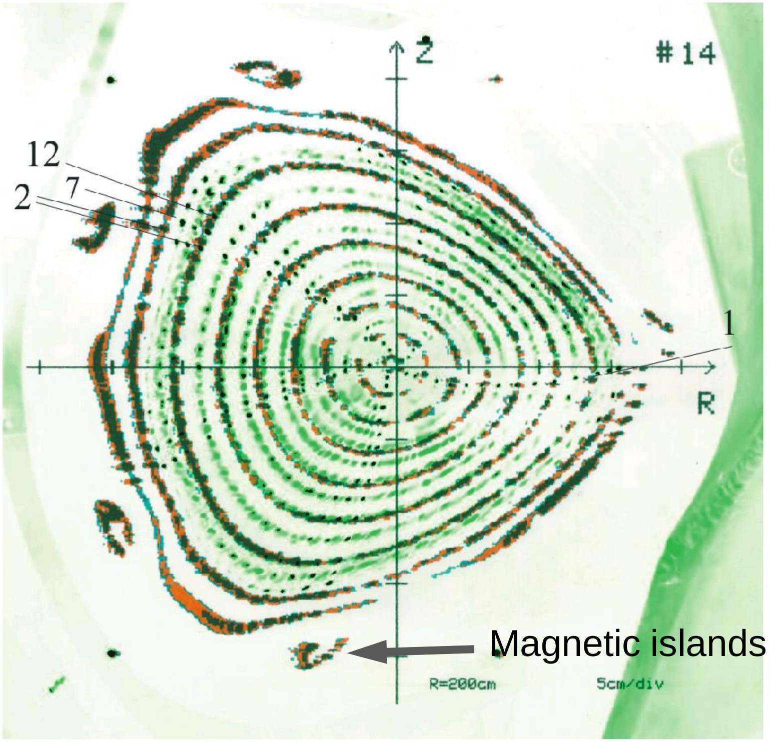
... in a tokamak they exist by symmetry considerations

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



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

... extremely sensitive measurement



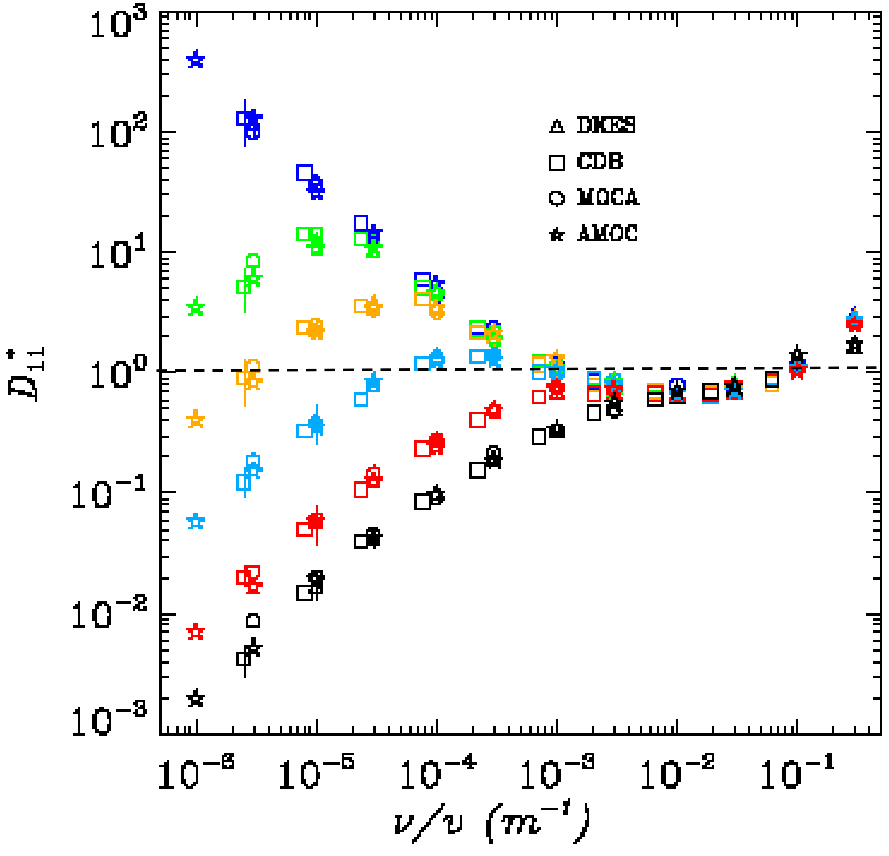
Magnetic islands

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

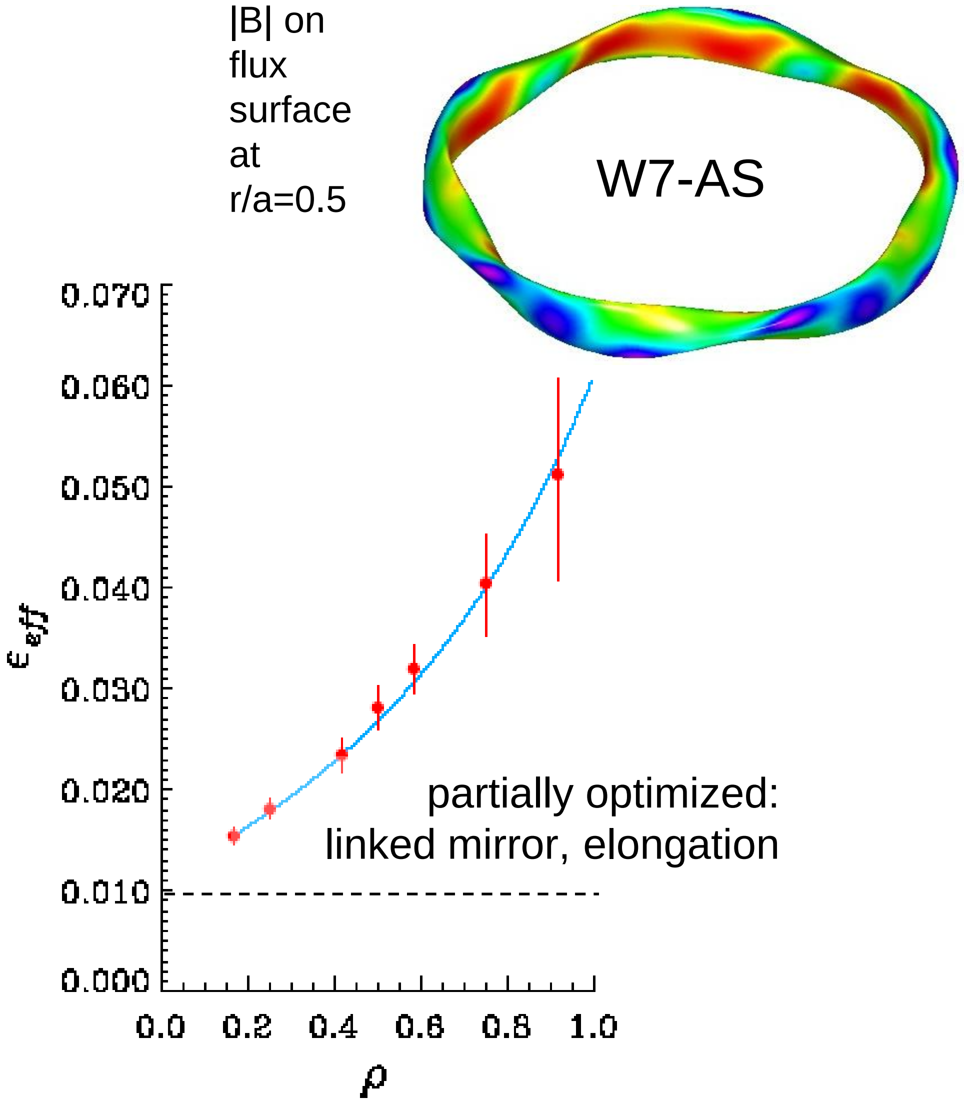
M. Otte, R. Jaenicke, Stell. News (2006)

Neoclassical transport coefficients in W7-AS (partially optimized)

W7-AS $\iota = 0.35$ Configuration



|B| on flux surface at r/a=0.5



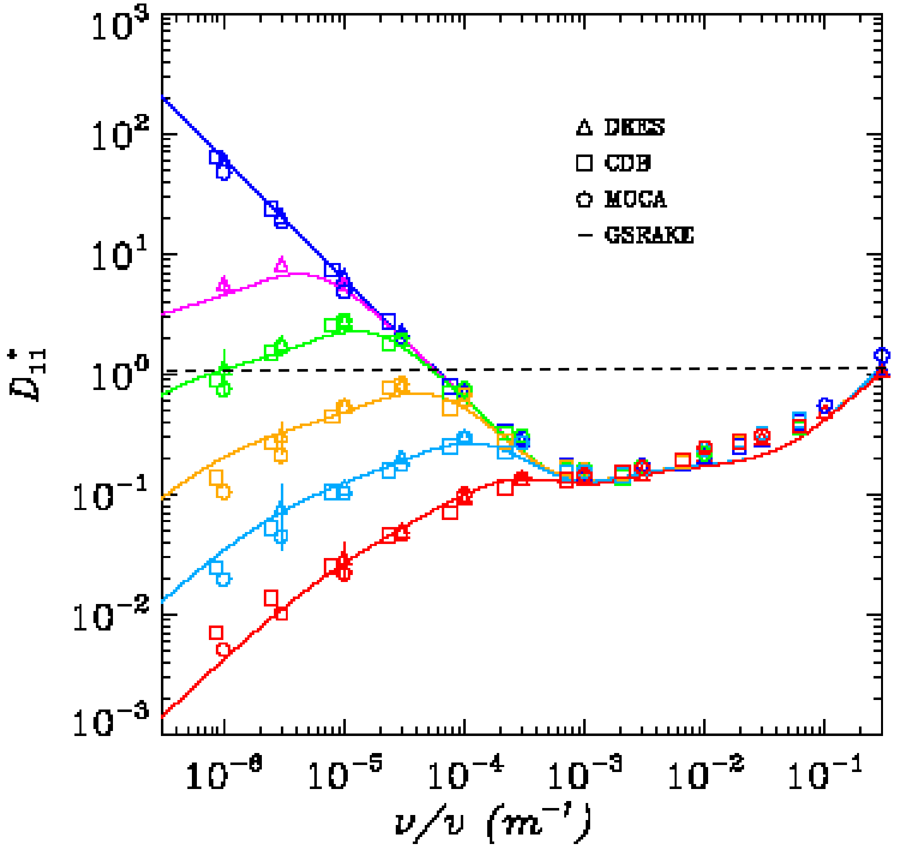
partially optimized:
linked mirror, elongation

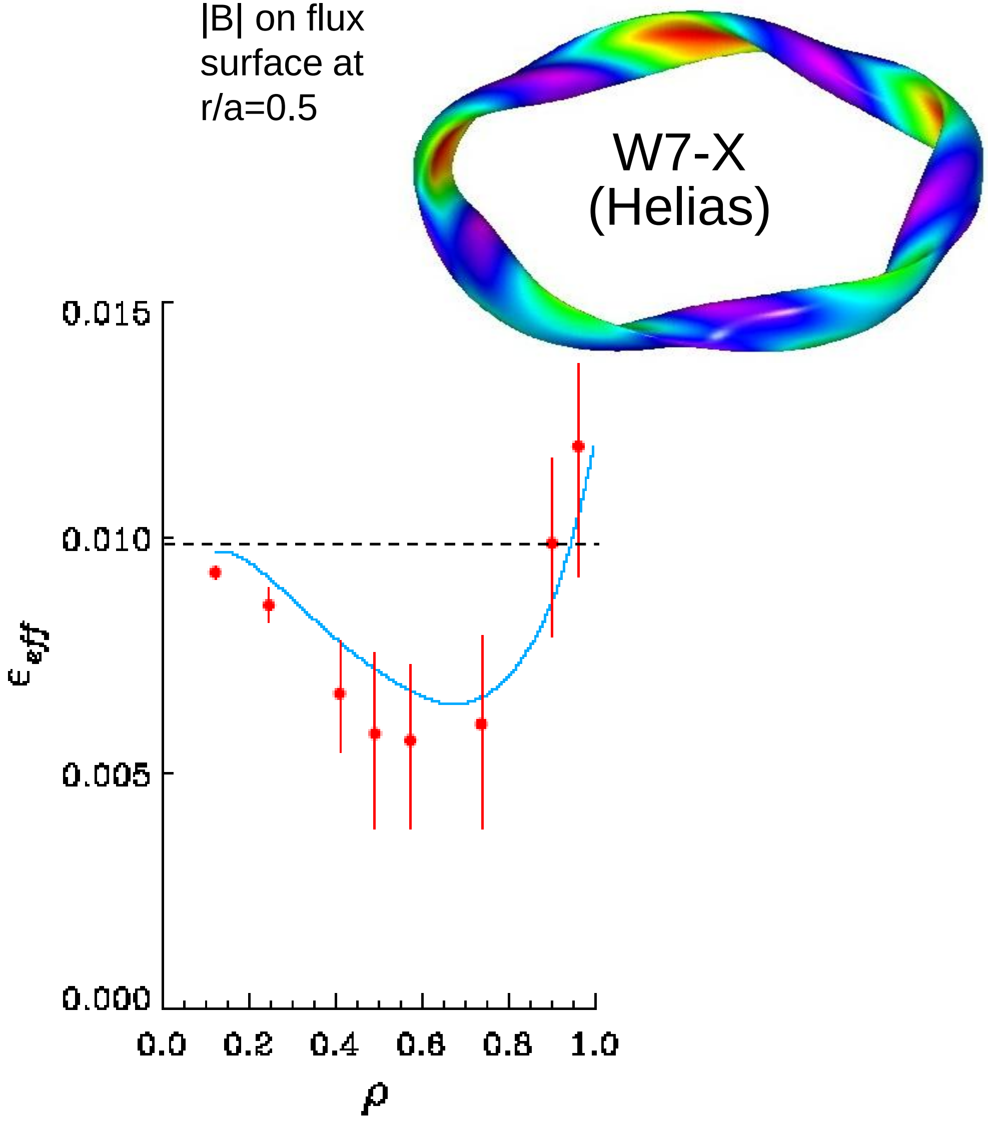
$|E|/vB_0 = 3\times10^{-3}$ 1×10^{-3} 3×10^{-4} 1×10^{-4} 3×10^{-5} zero

H. Maaßberg, C Beidler

Neoclassical transport coefficients in W7-AS (“fully” optimized)

W7-X Standard Configuration



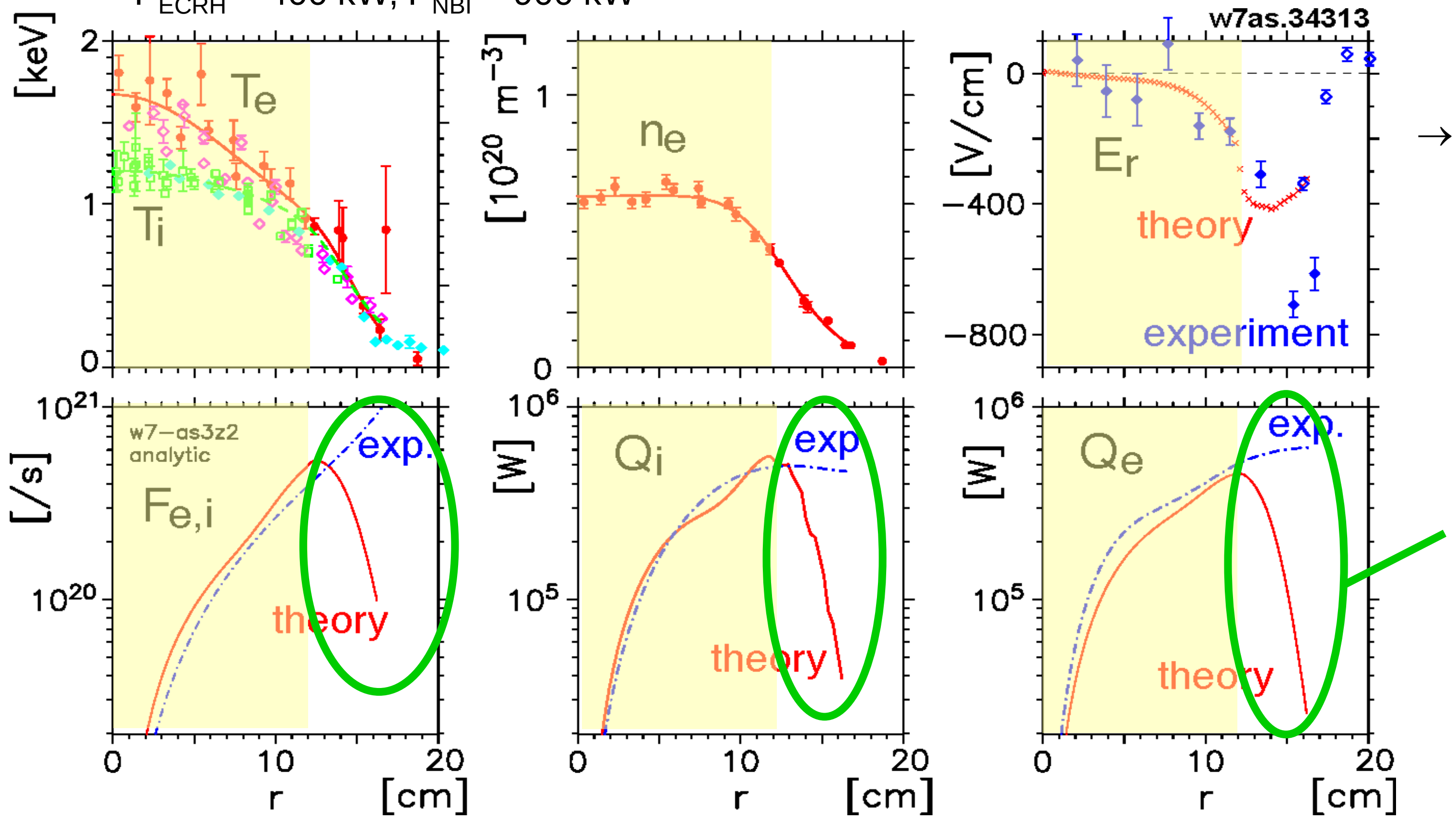


$|E|/vB_0 = 1\times10^{-3}$ 3×10^{-4} 1×10^{-4} 3×10^{-5} 1×10^{-5} zero

H. Maaßberg, C Beidler

Neoclassical transport confirmed in core region of W7-AS

P_{ECRH} = 400 kW, P_{NBI} = 900 kW



→ "ion-root" conditions (negative E_r)

plasma edge: anomalous / turbulent transport

(in particular for electrons)

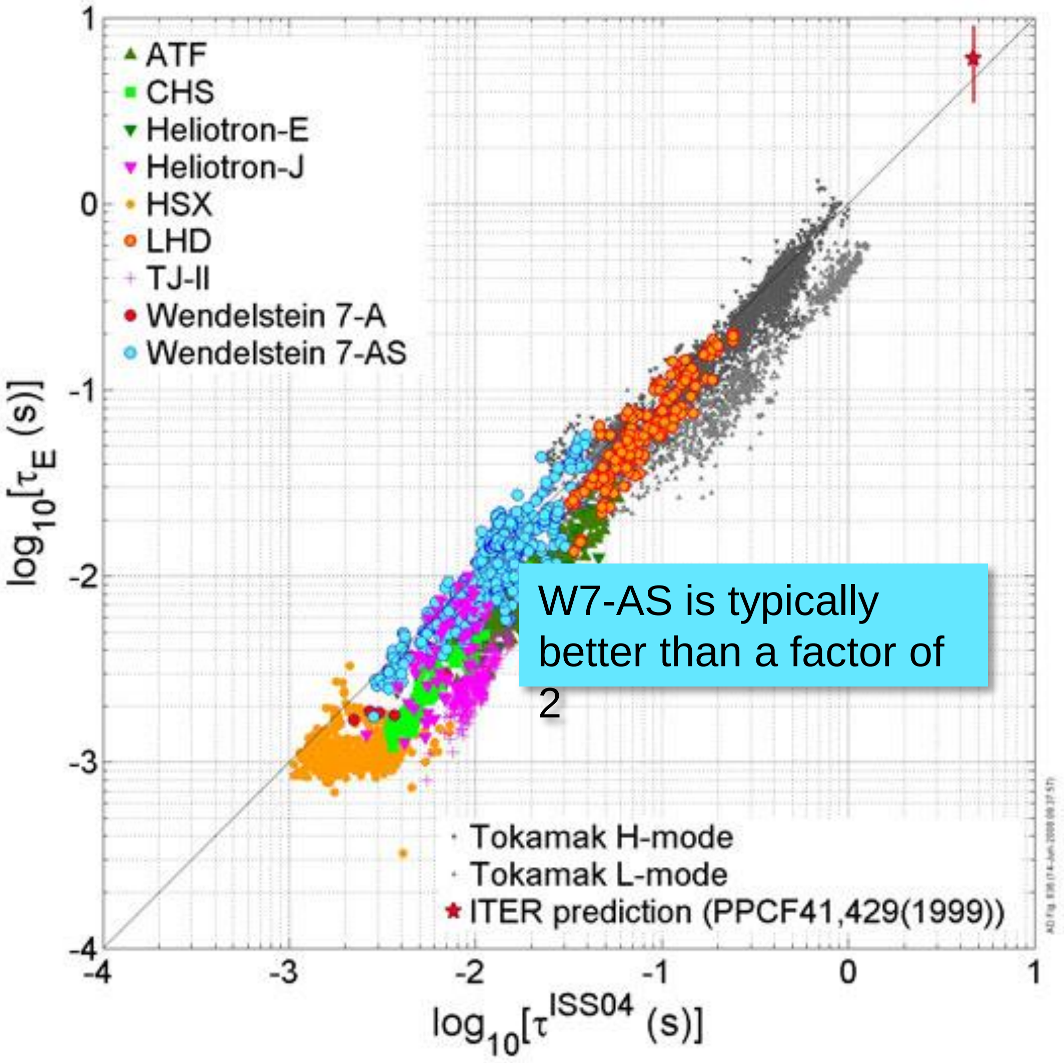
W7-AS : with neoclassical core best confinement was achieved

→ maximum Ti =1.5 keV max. Te=7 keV, max. t=55 ms (in different scenarii !!)

J. Baldzuhn, Plasma Phys. Contr. Fus., 40 967 (1998)

R. Jaenicke, Plasma Phys. Contr. Fus., 37A 163 (1995)

Confinement scaling



W7-AS is typically better than a factor of 2

- On a first glance:

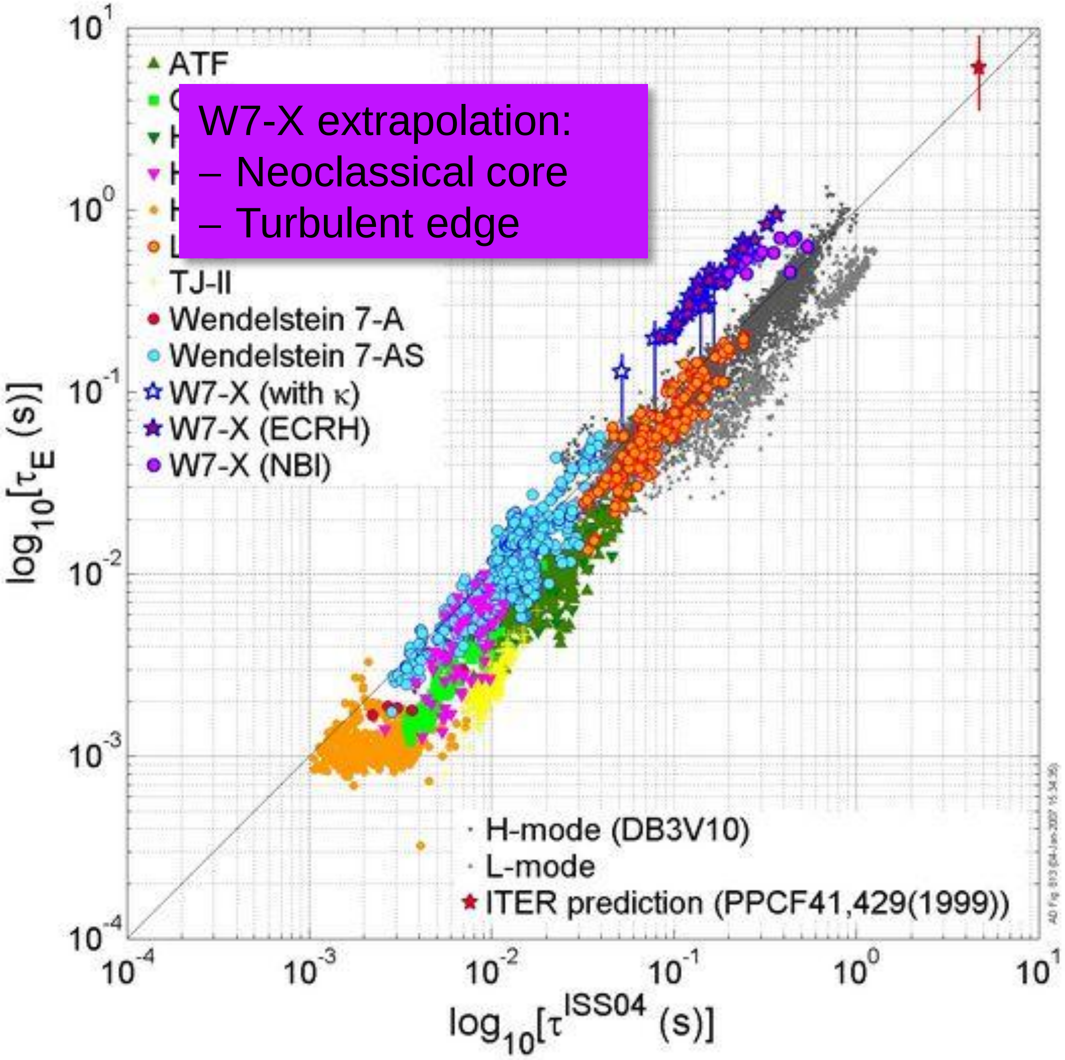
  Tokamak and stellarator scalings are very similar

$$\tau_E^{ISS04} = 0.134 a^{2.28} R^{0.64} P^{-0.61} \bar{n}_e^{0.54} B^{0.84} \iota_{2/3}^{0.41}$$

  (substitute I_p, B_T by ι and neglect isotope dependence)

- But, rather large scatter suggests unknown parameters
  - Variation of contributions from neoclassical and turbulent transport
  - Magnetic shear
  - Quasi-symmetry
  - … ???

Confinement scaling



W7-X extrapolation:
– Neoclassical core
– Turbulent edge

- On a first glance:

  Tokamak and stellarator scalings are very similar

$$\tau_E^{ISS04} = 0.134 a^{2.28} R^{0.64} P^{-0.61} \bar{n}_e^{0.54} B^{0.84} \text{ɨ}_{2/3}^{0.41}$$

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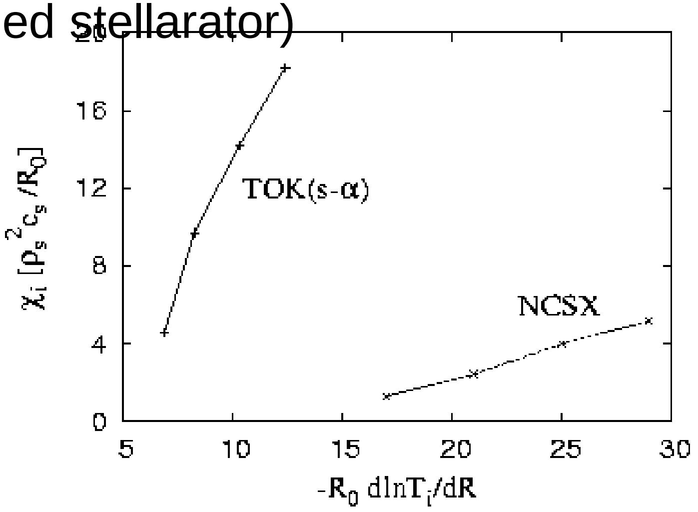
- But, rather large scatter suggests unknown parameters
  - Variation of contributions from neoclassical and turbulent transport
  - Magnetic shear
  - Quasi-symmetry
  - … ???

Summary

Plasma core:

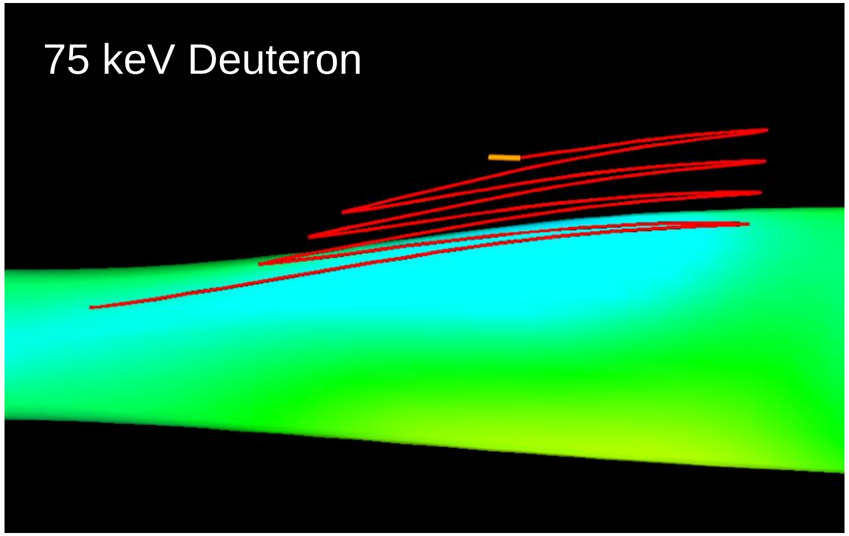
- Dominated by neoclassical transport
  - depending on temperature and configuration ($D \sim \varepsilon_{eff}^{3/2} T^{7/2}$)
  - interesting question: what happens to the relation between turbulent and neoclassical transport if ε_{eff} is very small (optimized stellarator)



Plasma edge:

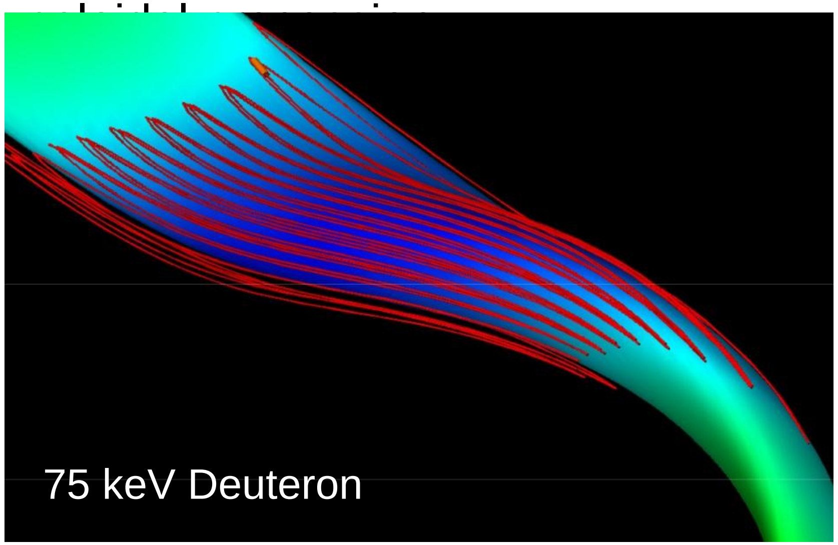
- As in tokamaks, turbulent behaviour prevails
- H-mode: suppression / reduction of this turbulence
  - confinement improvement typically below a factor of 2 (similarities to limiter H-mode in tokamaks)
  - no clear power threshold
  - dependence on magnetic field configuration (type of stellarator) and thus on a large number of paramaters

Burning fusion plasma requires drift optimization

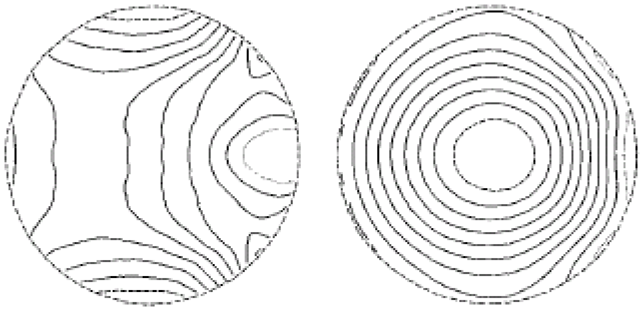


In partially optimized W7-AS fast ions were not confined (at low collision frequency)

Drift optimization in W7-X (introducing quasi-symmetry / quasi-isodynamicity) serves the confinement of fast ions:
Radial drift is transforemd into a



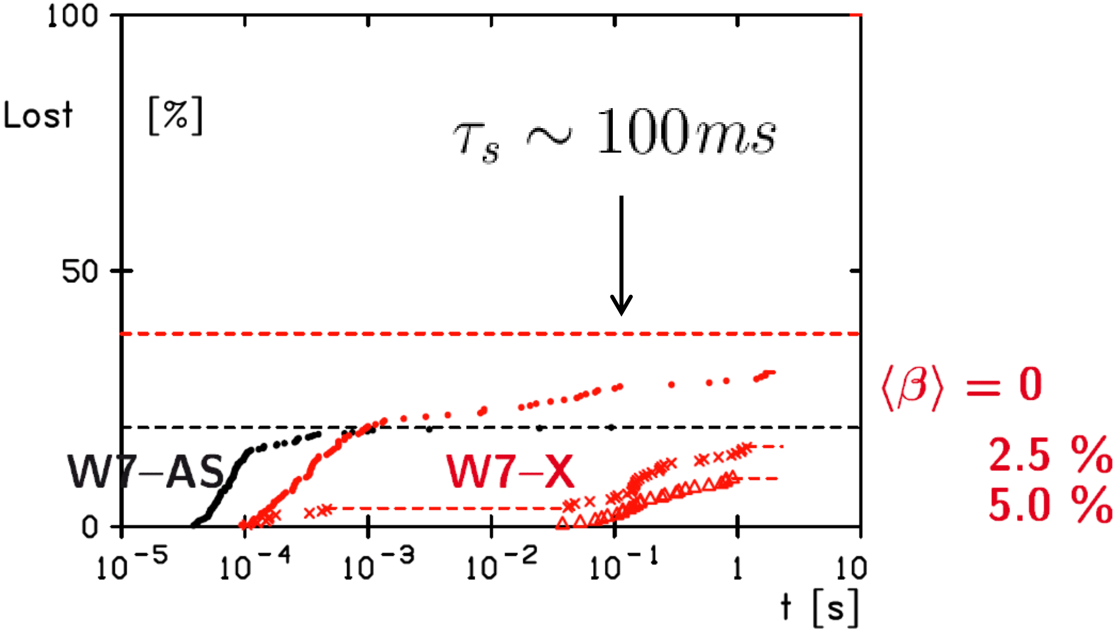
… but drift-optimization (W7-X) depends on β



$\langle\beta\rangle = 0$ $\langle\beta\rangle = 0.049$

constant-$\mathcal{J}$ contours

$$J = \oint v_{||} ds$$

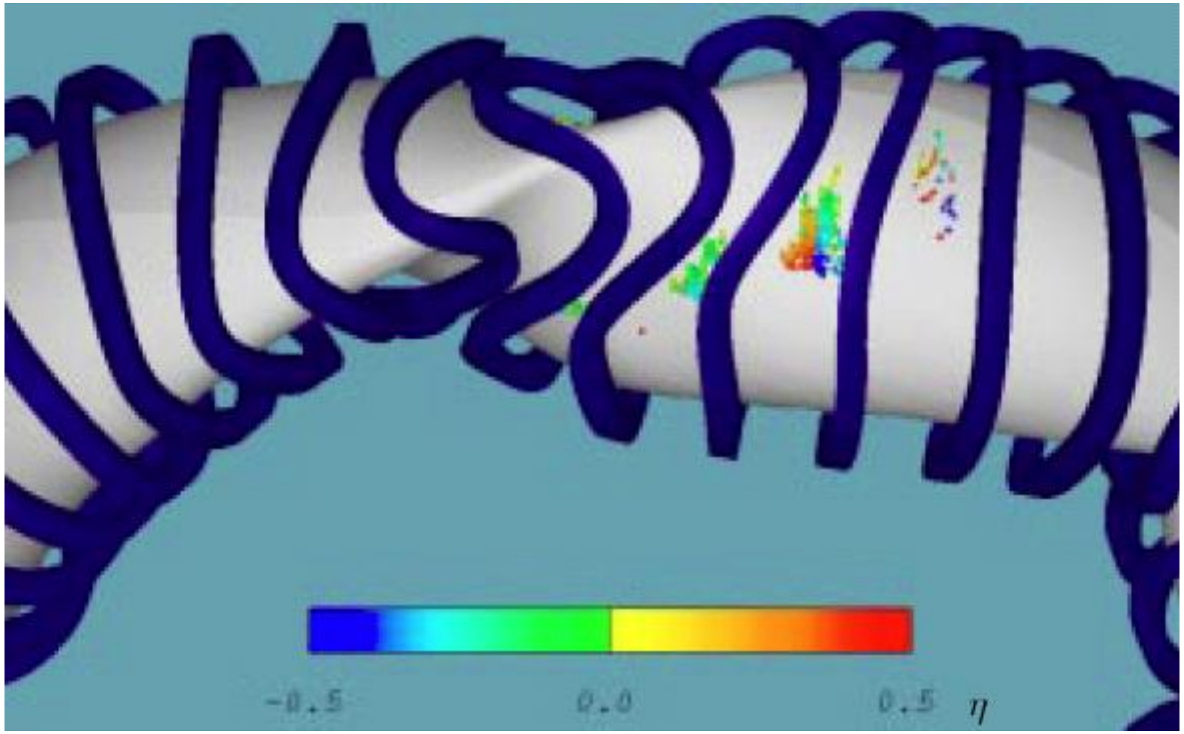


collisionless α-particle losses

W. Lotz et al, Plasma Phys. Control. Fusion 34 (1992) 1037

However, the problem may not be the loss power …

… but the localized ion loss trajectories



Latest calculations for W7-X show local heat fluxes of up to ~ 1 MW/m^2 !

from Strumberger et al., NF 40 (2000) 1698

A different ordering of the issues

- **Sufficient confinement of thermal plasma and fast ions (α-particles in a fusion reactor)**
  - **Flux surfaces**
  - **Plasma confinement, introduction of quasi-symmetries, H-modes in stellarators, the dependence of drift optimization on β**
- Steady state magnetic field
  - Inductive current not required
  - Superconducting coils
- Reliable operation at high plasma densities, high plasma pressure (β)
  - High density operation beyond the "Greenwald limit"
  - β-limits (stability and equilibrium)
- Wall materials compatible with heat and particle fluxes (neutron fluxes) and plasma operation, feasible exhaust concept
  - Divertor concepts, island divertor and sensitivity error fields and residual currents
  - Impurity control, the role of confinement scenarios (HDH-mode)
- Bringing everything together: The optimized stellarator – some technical issues

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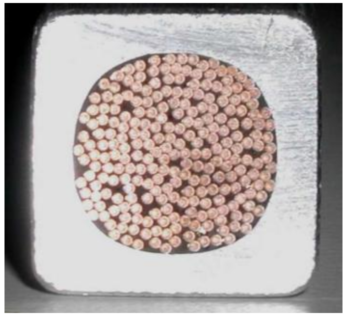
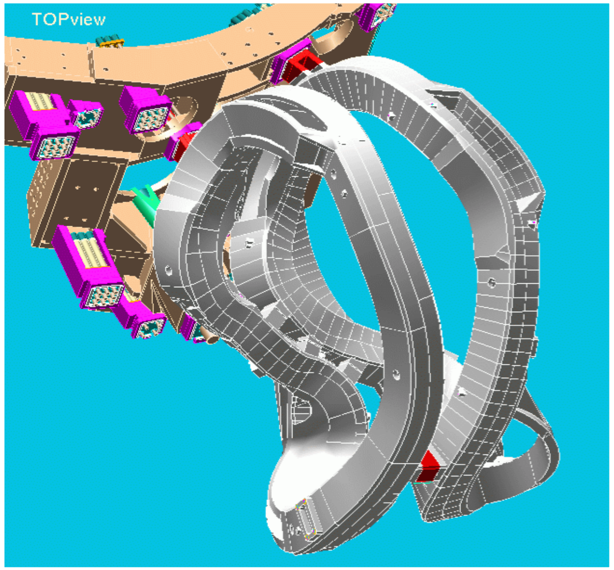
Some technical issues of (modular) coils

- 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

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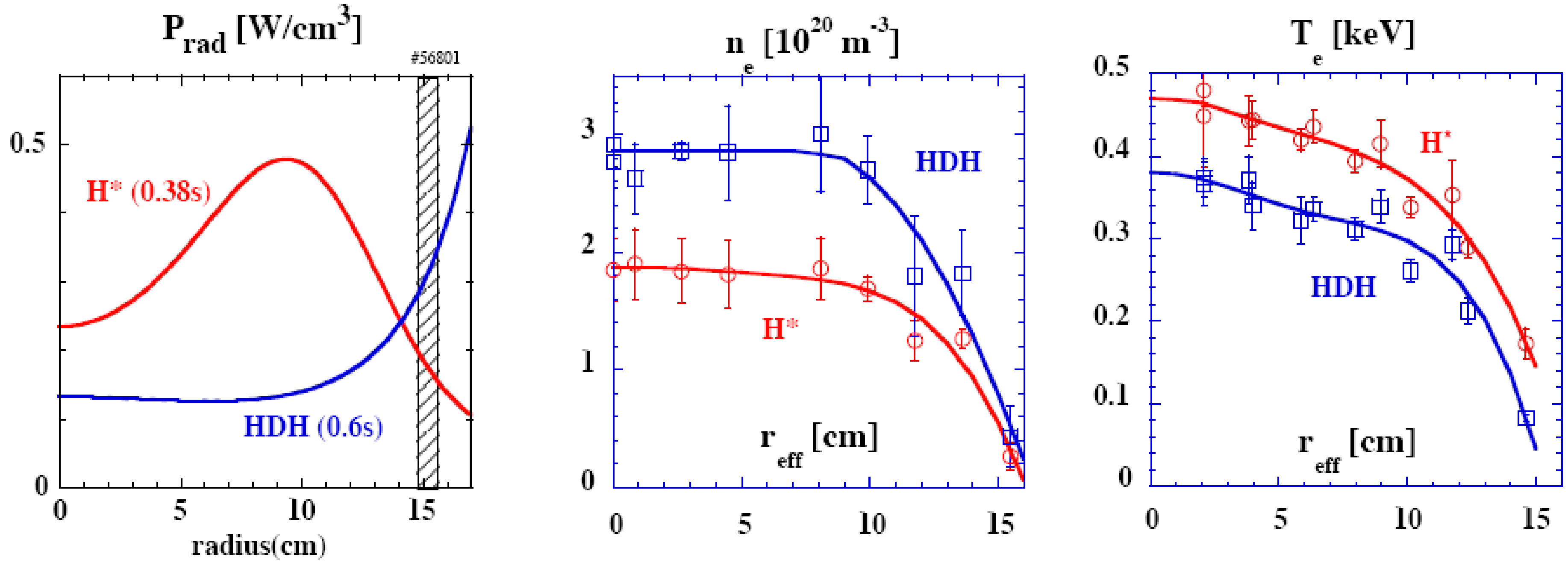
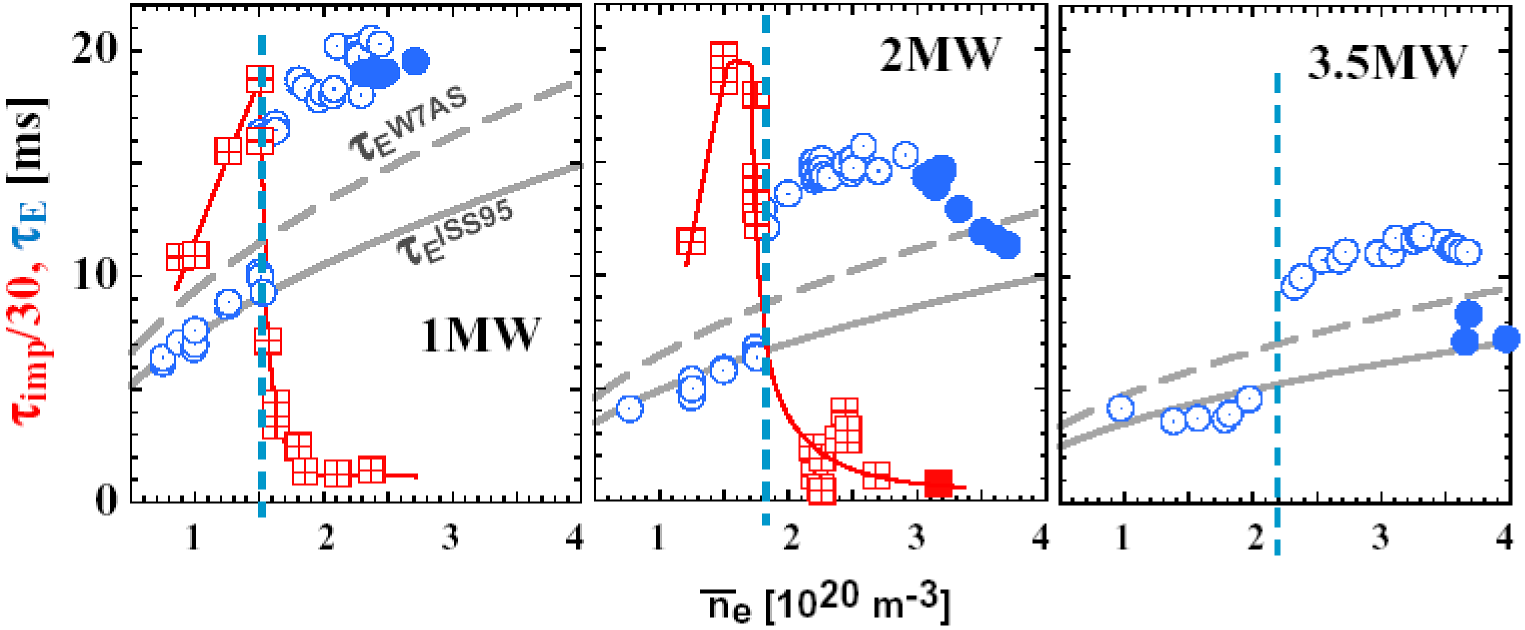
High Density H-mode (HDH) solved the problem in W7-AS



Combination of high density, improved energy confinement and reduced impurity confinement

High Density H-mode (HDH) solved the problem in W7-AS



- **HDH mode benefits from high density limit**
- **HDH Sensitively depends on (island) X-point position with respect to divertor target**
- **Incompatible with large bootstrap currents (remember discussion before)**
- **Unclear whether and how this extrapolates to W7-X (e.g. different collisionality, different connection lengths of the open field lines in the magnetic islands)**

A different ordering of the issues

- Sufficient confinement of thermal plasma and fast ions (α-particles in a fusion reactor)
  - Flux surfaces
  - Plasma confinement, introduction of quasi-symmetries, H-modes in stellarators, the dependence of drift optimization on β
- Steady state magnetic field
  - Inductive current not required
  - Superconducting coils
- Reliable operation at high plasma densities, high plasma pressure (β)
  - High density operation beyond the "Greenwald limit"
  - β-limits (stability and equilibrium)
- **Wall materials compatible with heat and particle fluxes (neutron fluxes) and plasma operation, feasible exhaust concept**
  - **Divertor concepts, island divertor and sensitivity error fields and residual currents**
  - **Impurity control, the role of confinement scenarios (HDH-mode)**
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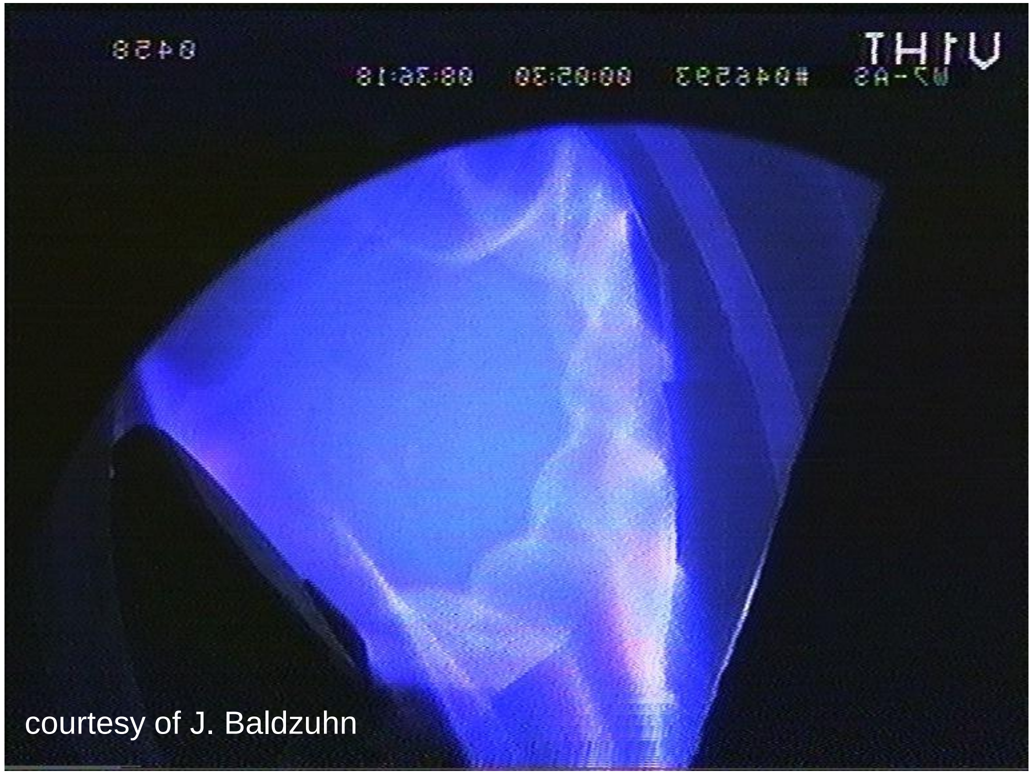
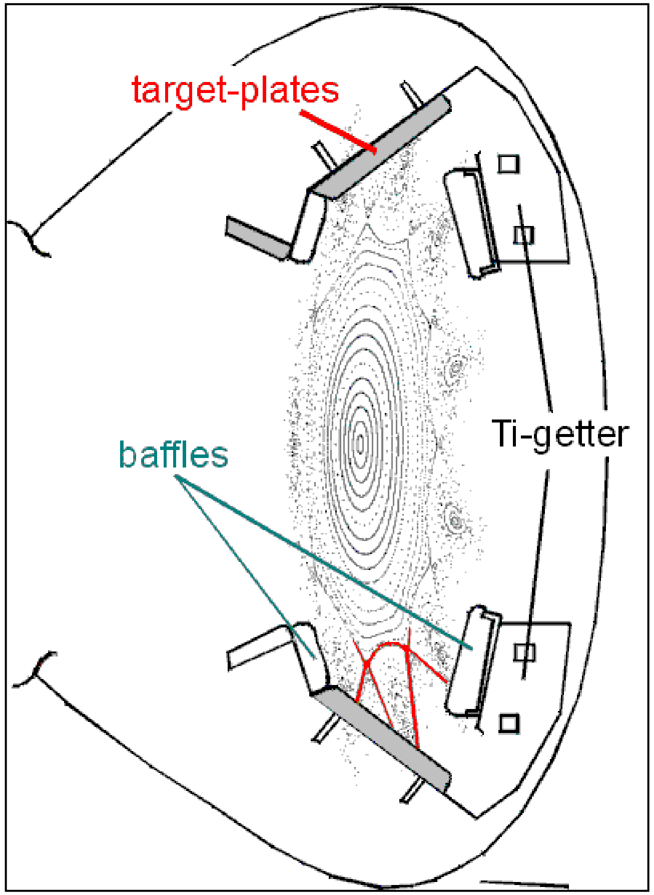
Plasma exhaust requires a feasible divertor concept

Magnetic island divertor in Wendelstein 7-AS

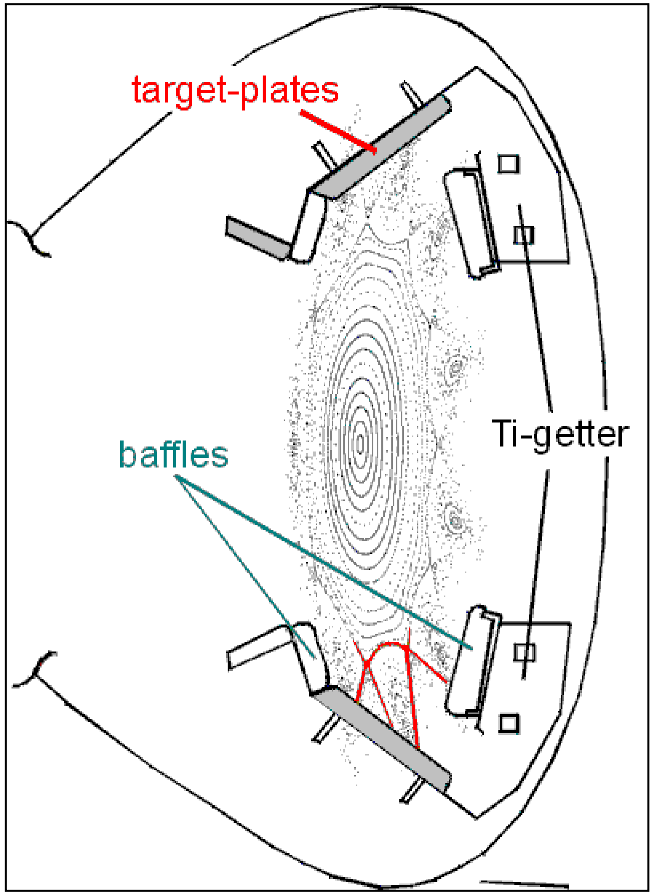

courtesy of J. Baldzuhn



Plasma exhaust requires a feasible divertor concept

Magnetic island divertor in Wendelstein 7-AS

Divertor module

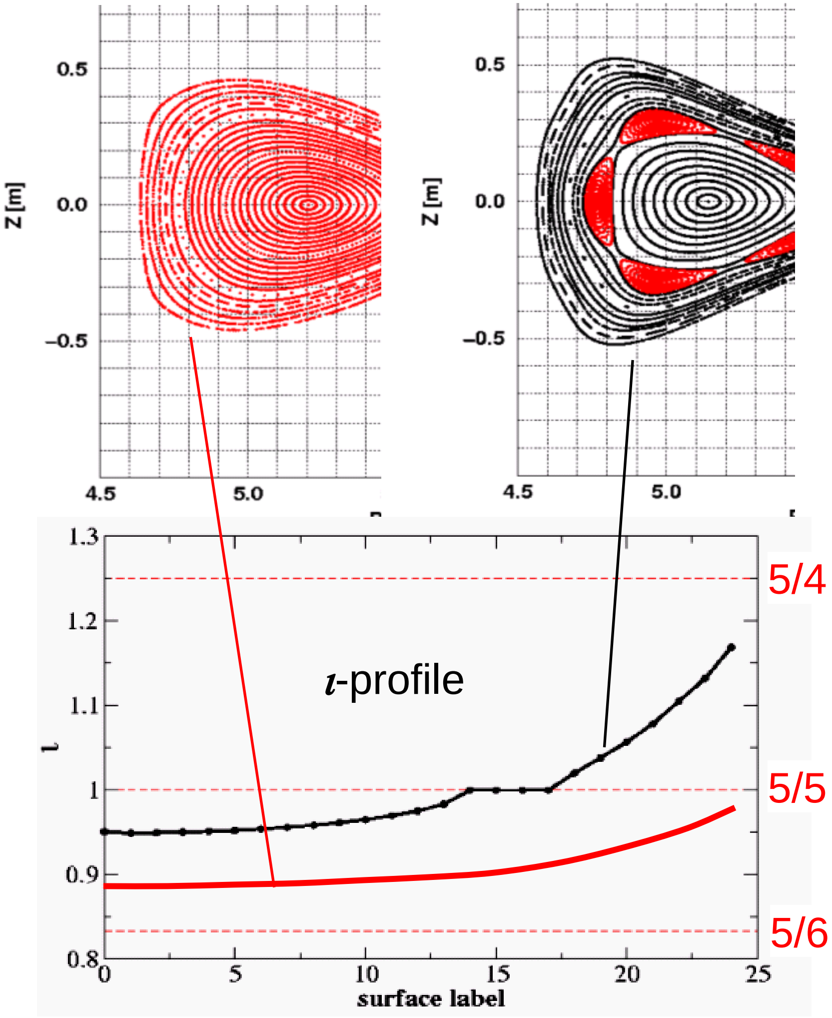


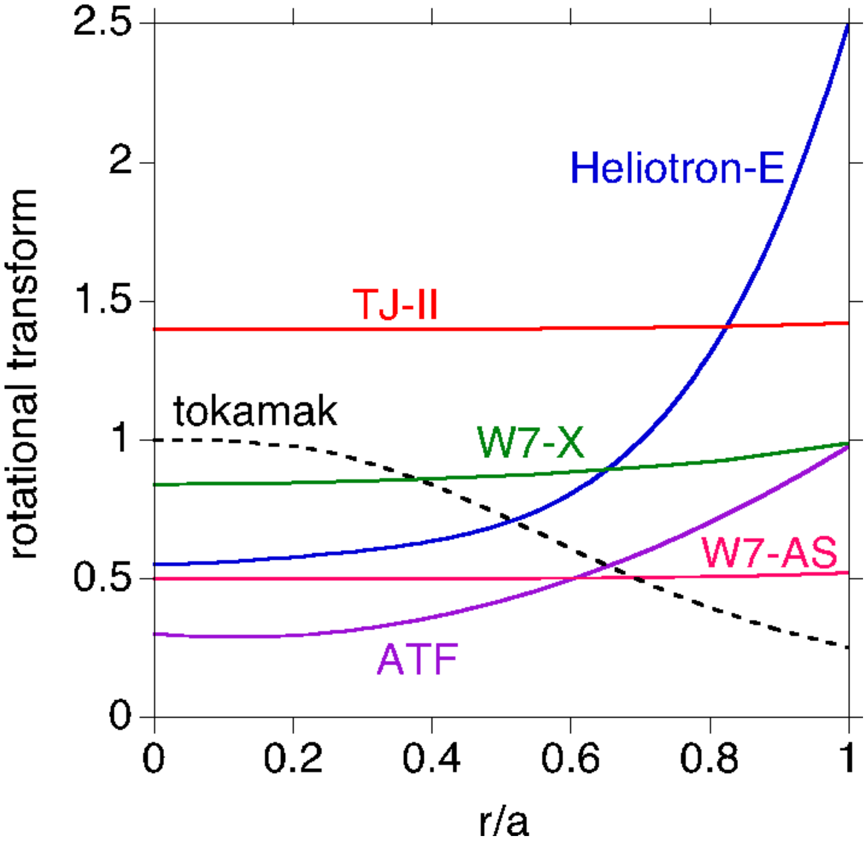
Requirements for an island divertor …

… low magnetic shear and resonance at the plasma boundary

W7-X standard case :
low m,n rationals
avoided

W7-X: high-iota case:
ι = 5/5 resonance
with islands





→ Magnetic shear determines the width of island

→ Cutting of islands with target plates produces divertor configuration

Magnetic field errors have to be kept small

Consequences of error fields (for W7-X)

- B_{11} causes deformation of 1/1 island → asymmetric divertor load for ɨ = 1
- For ɨ > 1 additional helicity of magnetic axis
- Magnetic islands in confinement region reduces effective plasma radius → confinement properties
- Additional ergodization at the edge → broadening of strike zones ?

Criterium
$$\frac{\Delta B}{B_0} \leq 1 \times 10^{-4}$$

e.g.
$$\Delta B = \sqrt{B_{11}^2 + B_{22}^2 + B_{33}^2 + B_{44}^2}$$

⇒
$$\Delta R \approx \frac{\Delta B}{B_0} R_0$$

with R_0 = 5.5 m ΔR ≈ 5 mm

Magnetic field errors have to be kept small

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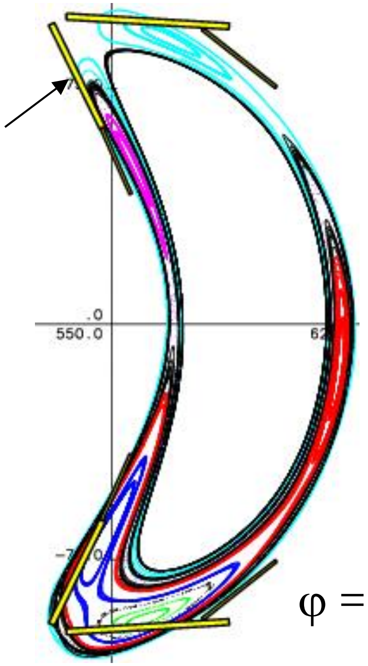
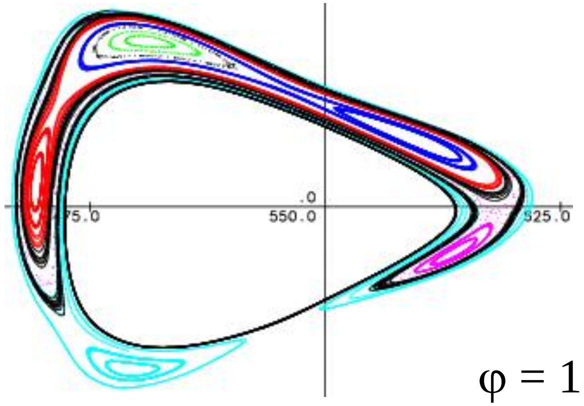


First contact with target

$\varphi = 0°$

$B_{11}/B_0 = 2.3 \cdot 10^{-4}$



$\varphi = 180°$

from Kisslinger et al.

Magnetic field errors have to be kept small

Consequences of error fields (for W7-X)

- B_{11} causes deformation of 1/1 island → asymmetric divertor load for ι = 1
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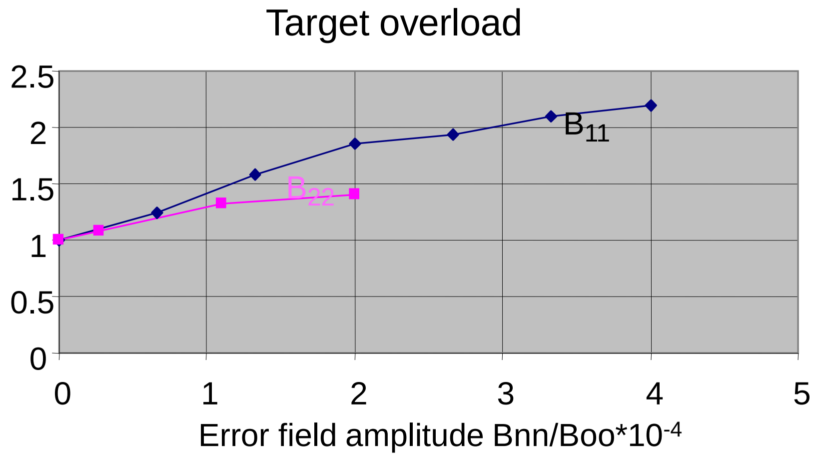
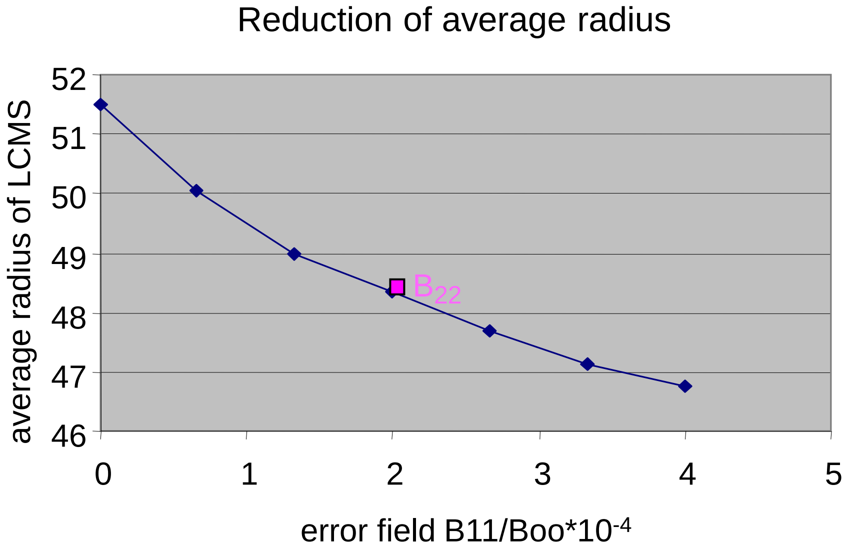
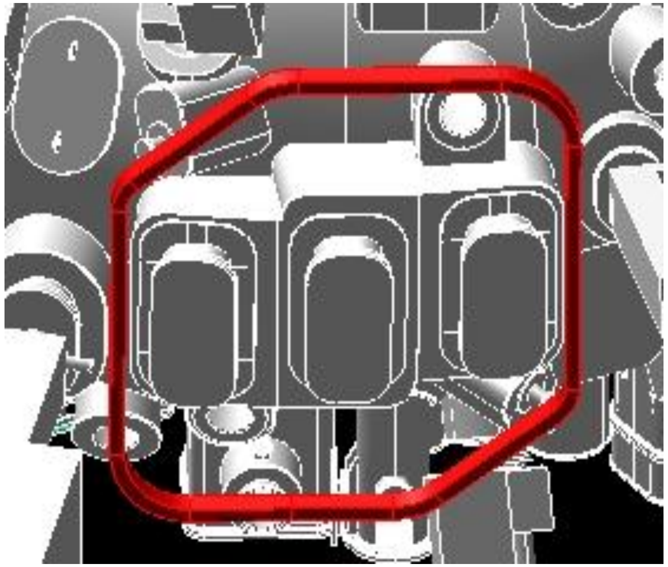


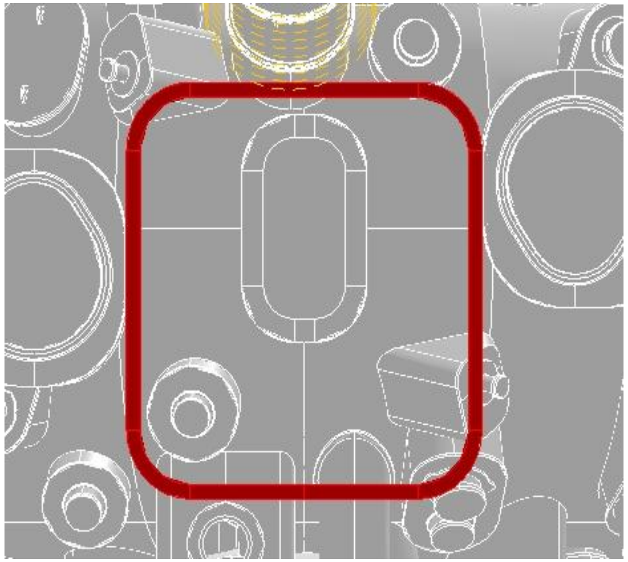


from Kisslinger et al.

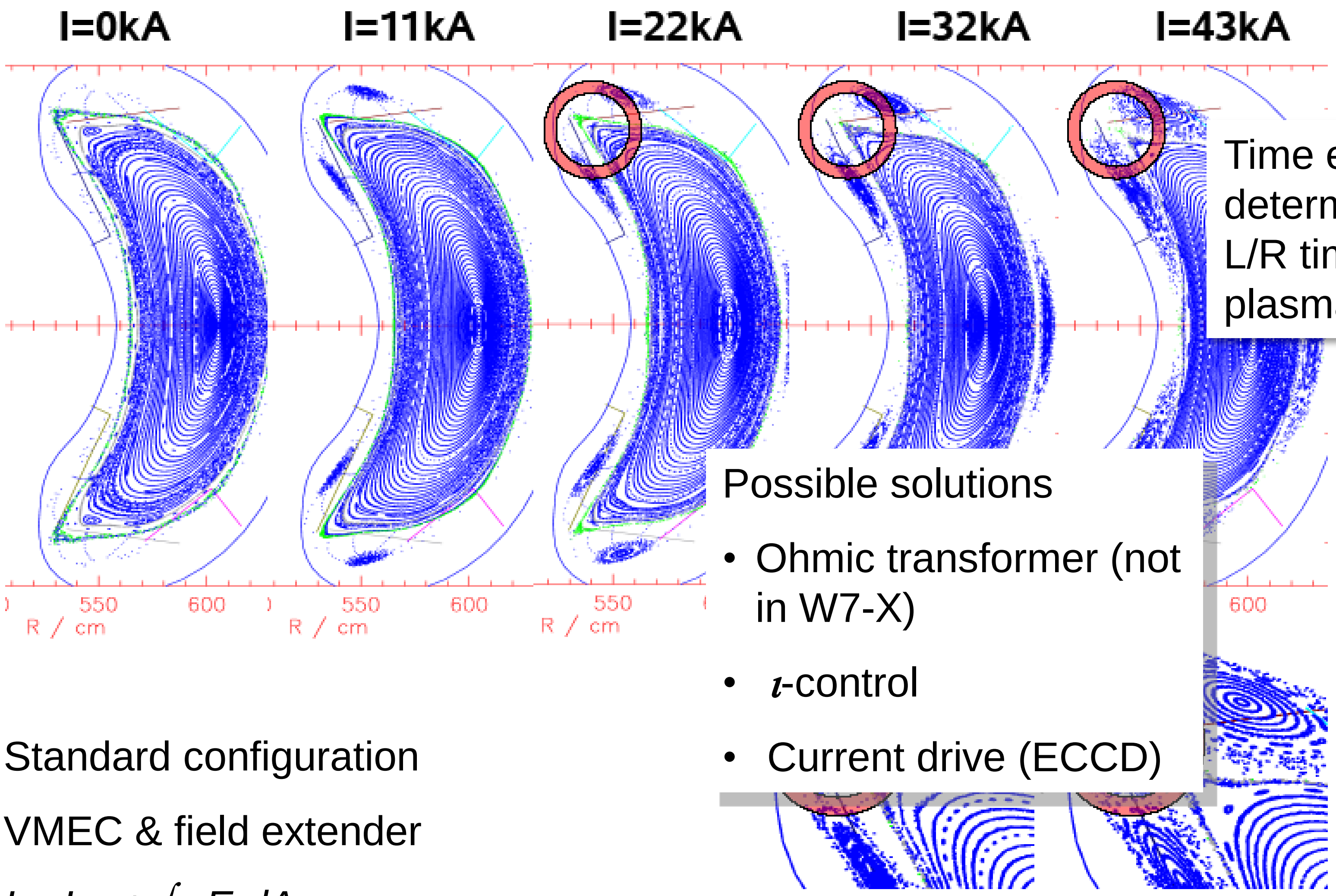
Low order error field correction …

… by 5 saddle coils (1 per magnetic field module) on the outboard side around the torus (copper coils outside cryostat)





The bootstrap current has to remain small



Time evolution determined by L/R time of the plasma ~ 40 s

Possible solutions

- Ohmic transformer (not in W7-X)
- ι-control
- Current drive (ECCD)

Standard configuration

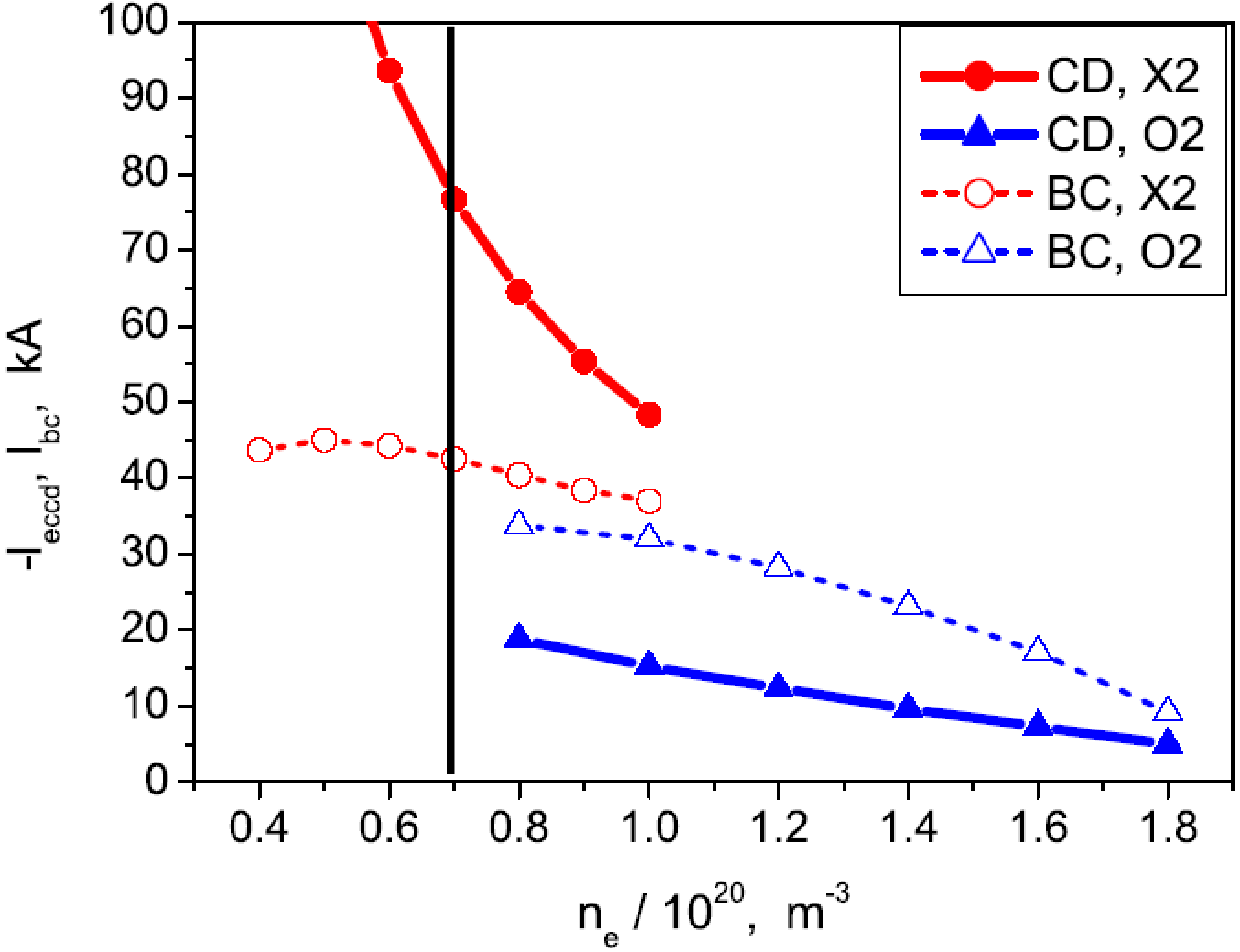
VMEC & field extender

$I = I_{BS} + \int \sigma E \, dA$

J. Geiger

In so-called high mirror configuration the bootstrap current is effectively zero !

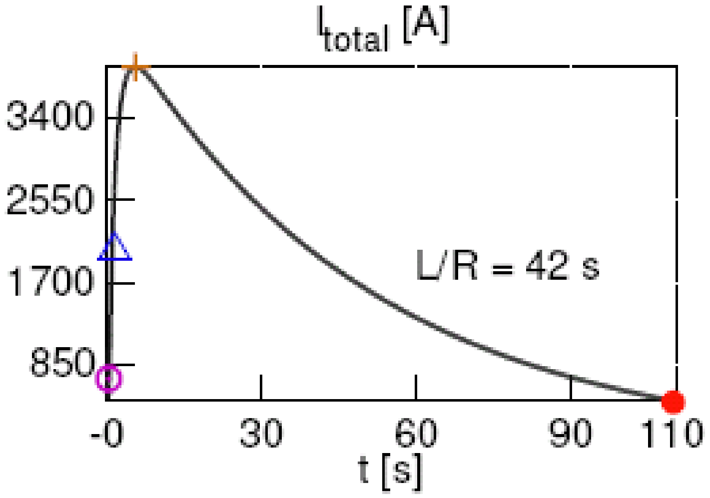
The bootstrap current compensation by EC current drive



5 MW ECCD
Standard configuration

C. Beidler, H. Maasberg et al.

The bootstrap current compensation by EC current drive



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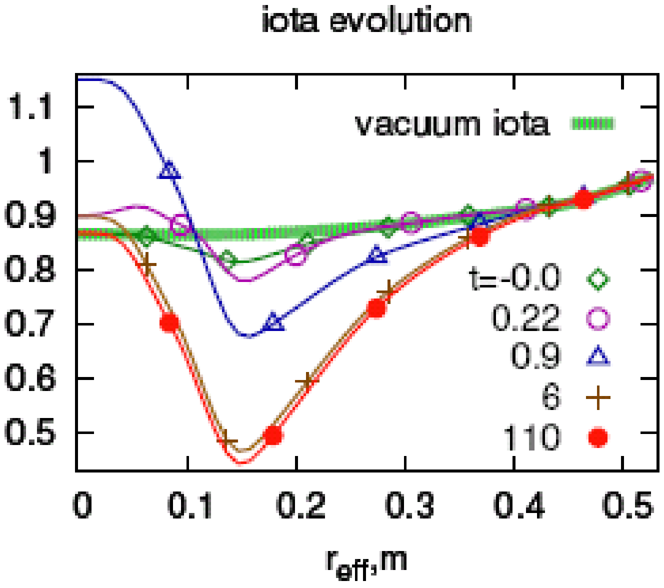


Problem:
- No low shear profile anymore
- Rational surfaces appear inside the plasma

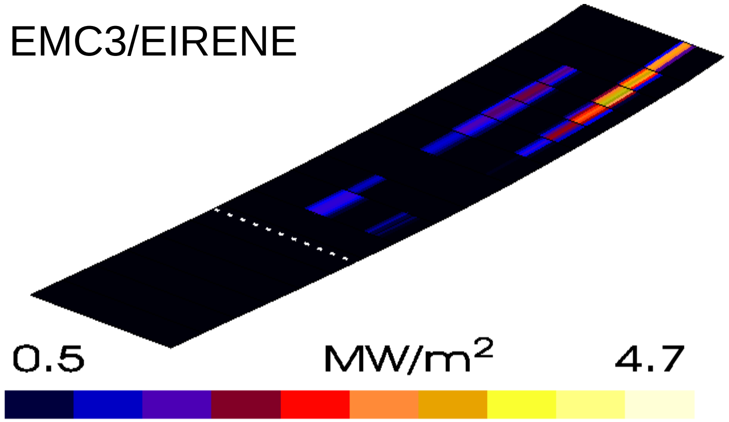
C. Beidler, H. Maasberg et al.

Summary resonant island divertor

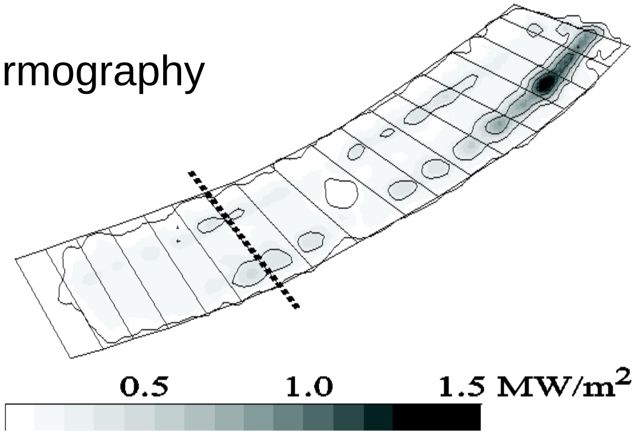
- Broader heat deposition profiles due to longer connection lengths
  - compared to the standard poloidal divertor
  - to be confirmed experimentally by W7-X
- Discontinuous heat distribution in helical direction
  - in fact in W7-X the divertor modules are discontinuous (see below)
- Resonance condition with low magnetic shear must be fulfilled
  - sensitive to plasma currents and resonant magnetic field perturbations

Heat flux pattern from W7-AS

EMC3/EIRENE



Thermography



A different ordering of the issues

- Sufficient confinement of thermal plasma and fast ions (α-particles in a fusion reactor)
  - Flux surfaces
  - Plasma confinement, introduction of quasi-symmetries, H-modes in stellarators, the dependence of drift optimization on β

- Steady state magnetic field
  - Inductive current not required
  - Superconducting coils

- Reliable operation at high plasma densities, high plasma pressure (β)
  - High density operation beyond the "Greenwald limit"
  - β-limits (stability and equilibrium)

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- **Bringing everything together: The optimized stellarator – some technical issues**

The optimization criteria of W7-X

- Stiff equilibrium configuration: Small Pfirsch-Schlüter and bootstrap currents resulting in small Shafranov shift and high equilibrium beta limit
- MHD stability up to $\langle\beta\rangle = 5\%$
- Small neoclassical transport $D \sim \varepsilon_{eff}^{3/2}\, T^{7/2}$
- Drift optimization (quasi-isodynamic configuration): Good fast particle confinement

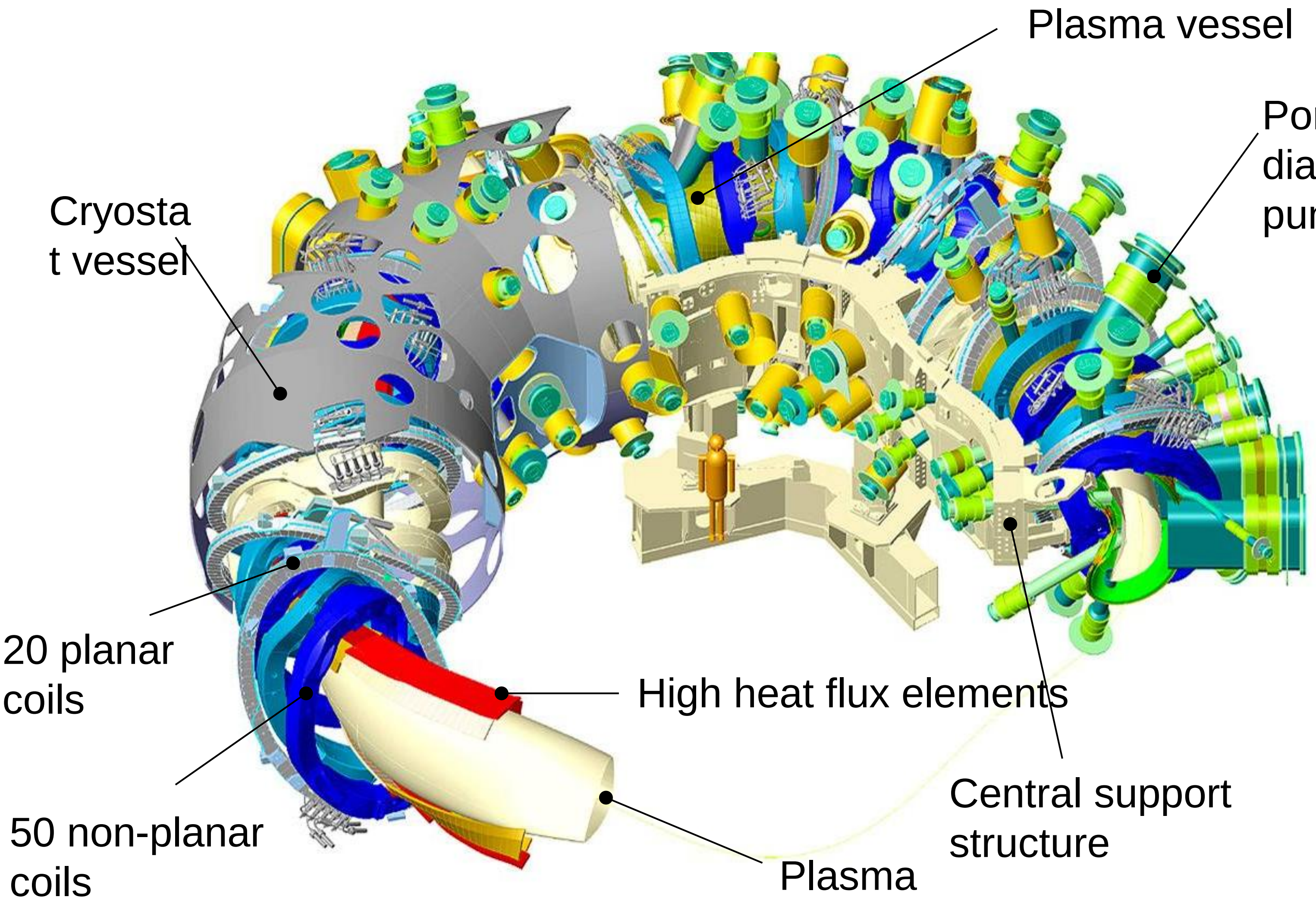
Additional objectives: Steady state operation including particle and energy exhaust with island divertor concept

- Superconducting coils
- Actively cooled divertor and first wall components
- Low magnetic shear with large islands at the plasma boundary
- ι as much as possible independent of β

→ In short: Plasma and magnetic field are as much as possible decoupled

→ Other optimization criteria are thinkable (e.g. NCSX: tokamak-stellarator hybrid with maximum bootstrap current)

Wendelstein 7-X



Magnetic field (superconducting)
3 T

Magnetic field energy
900 MJ

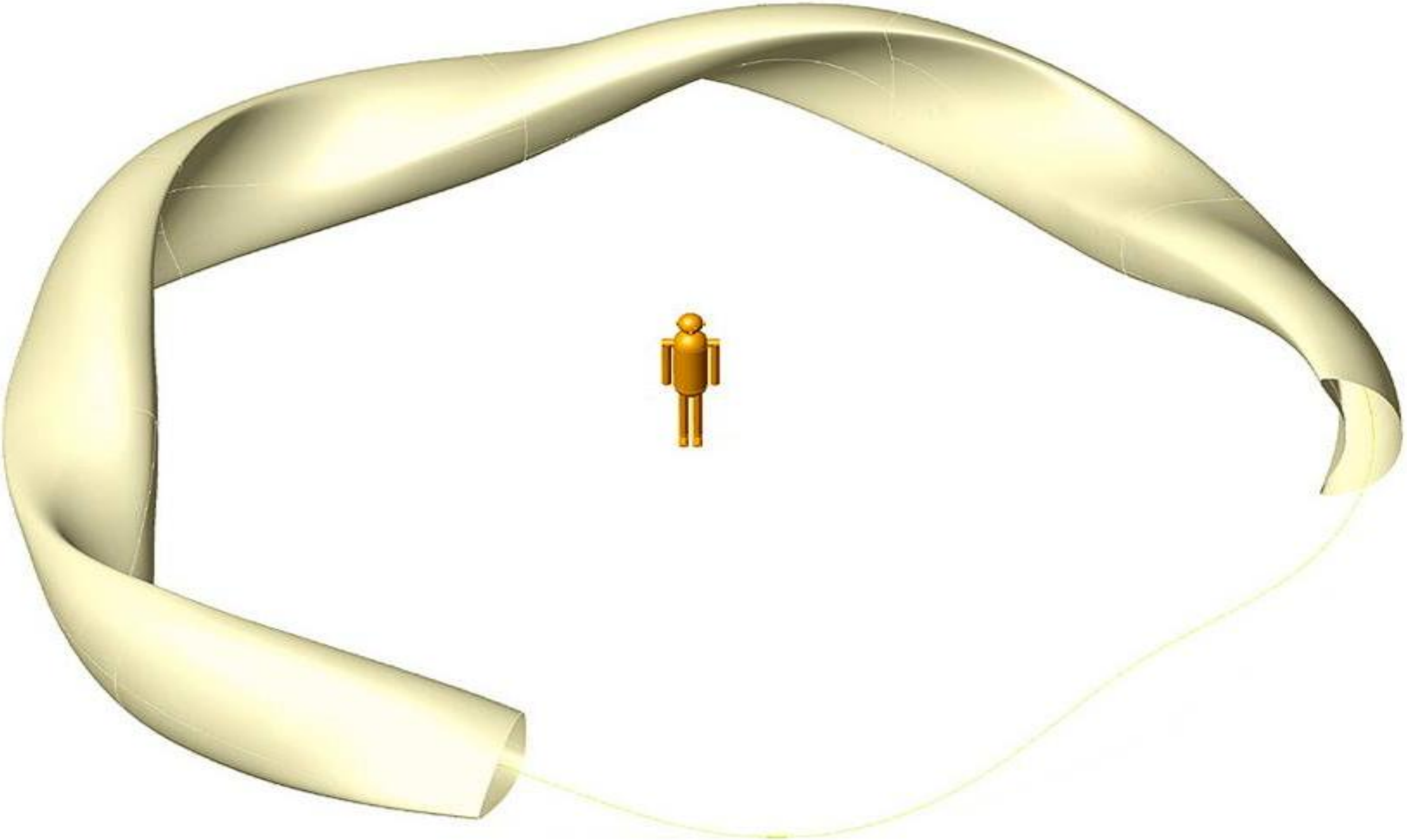
Plasma volume
30 m^3

Discharge duration
30 minutes

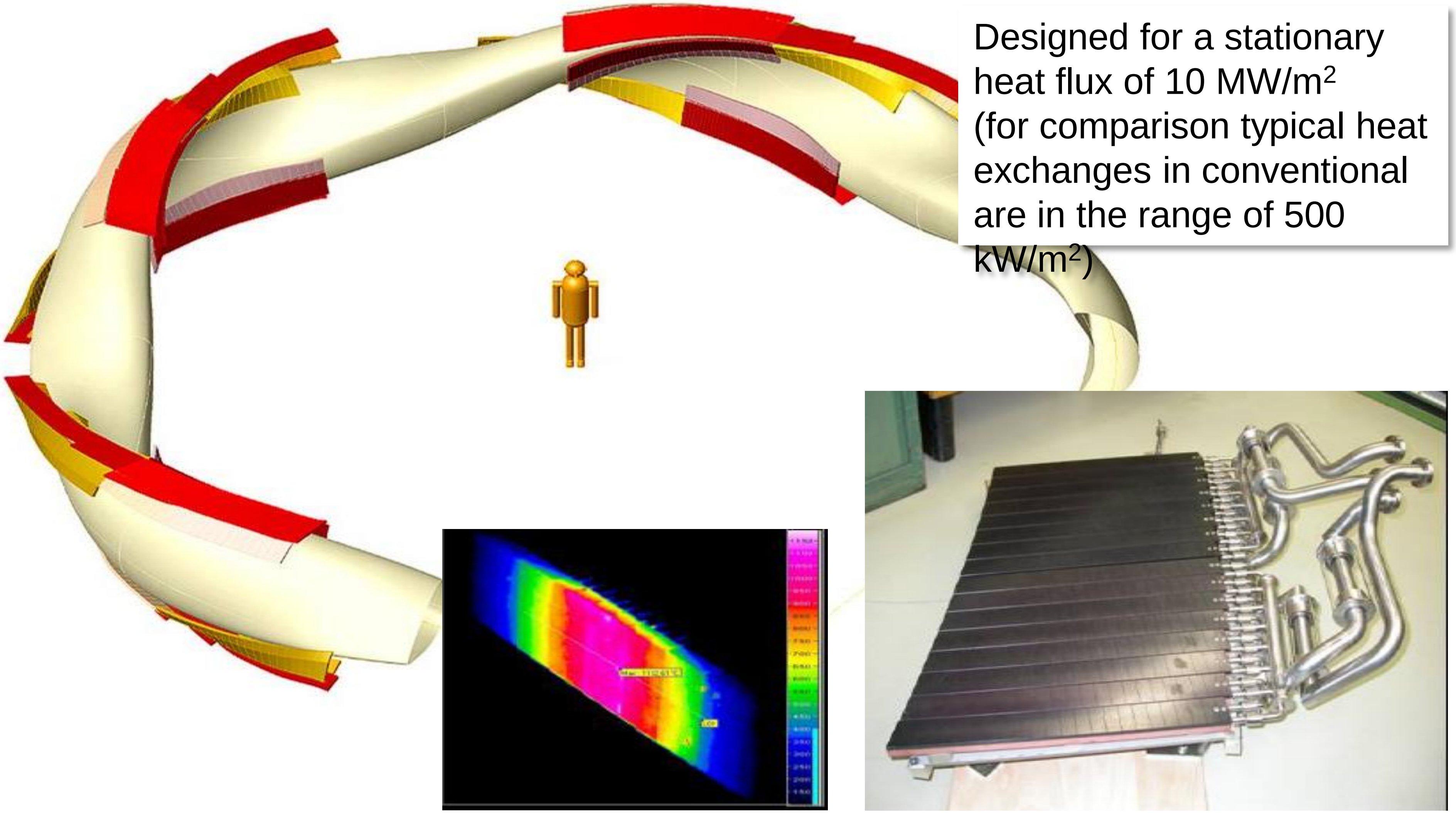
Heating power
10 MW (30 MW)

Maximum heat load
10 MW/m^2

Wendelstein 7-X – plasma



Wendelstein 7-X – island divertor

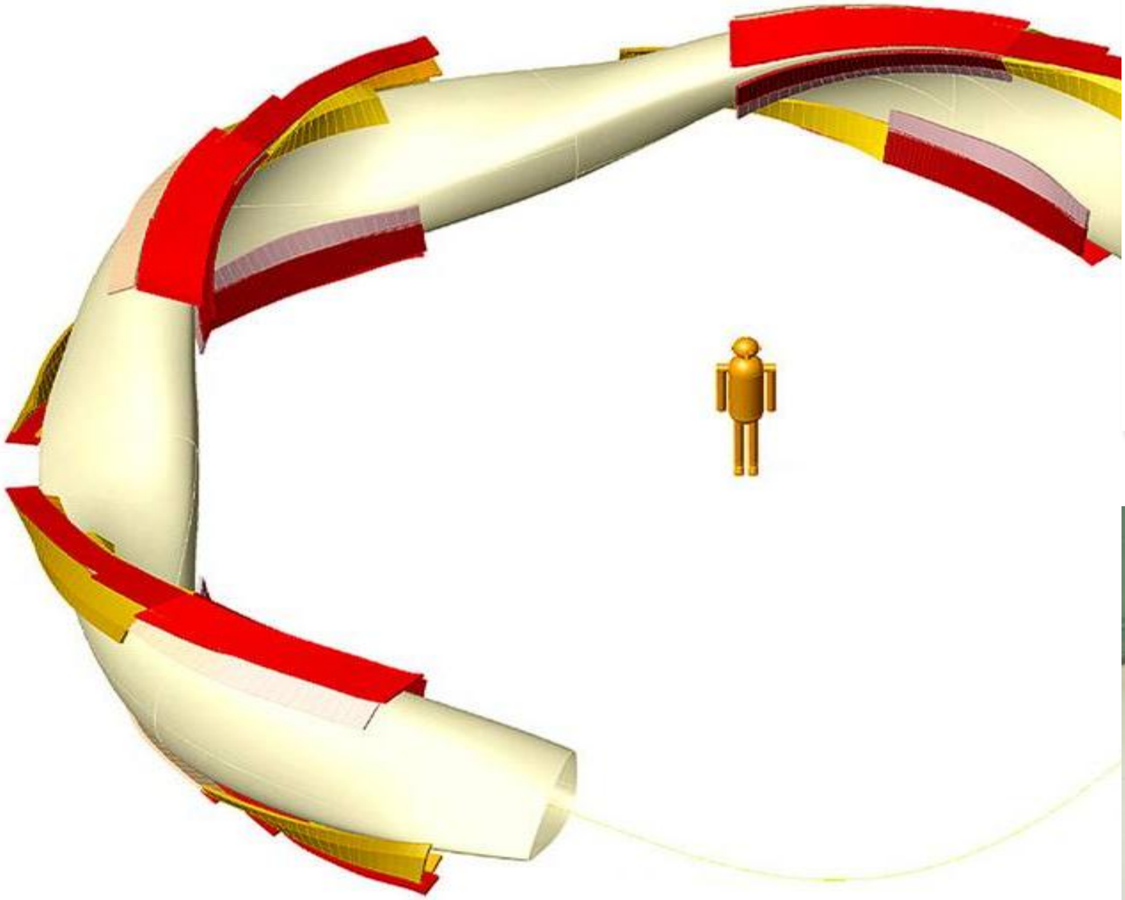


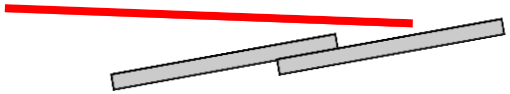
Designed for a stationary heat flux of 10 MW/m^2 (for comparison typical heat exchanges in conventional are in the range of 500 kW/m^2)

Wendelstein 7-X – island divertor

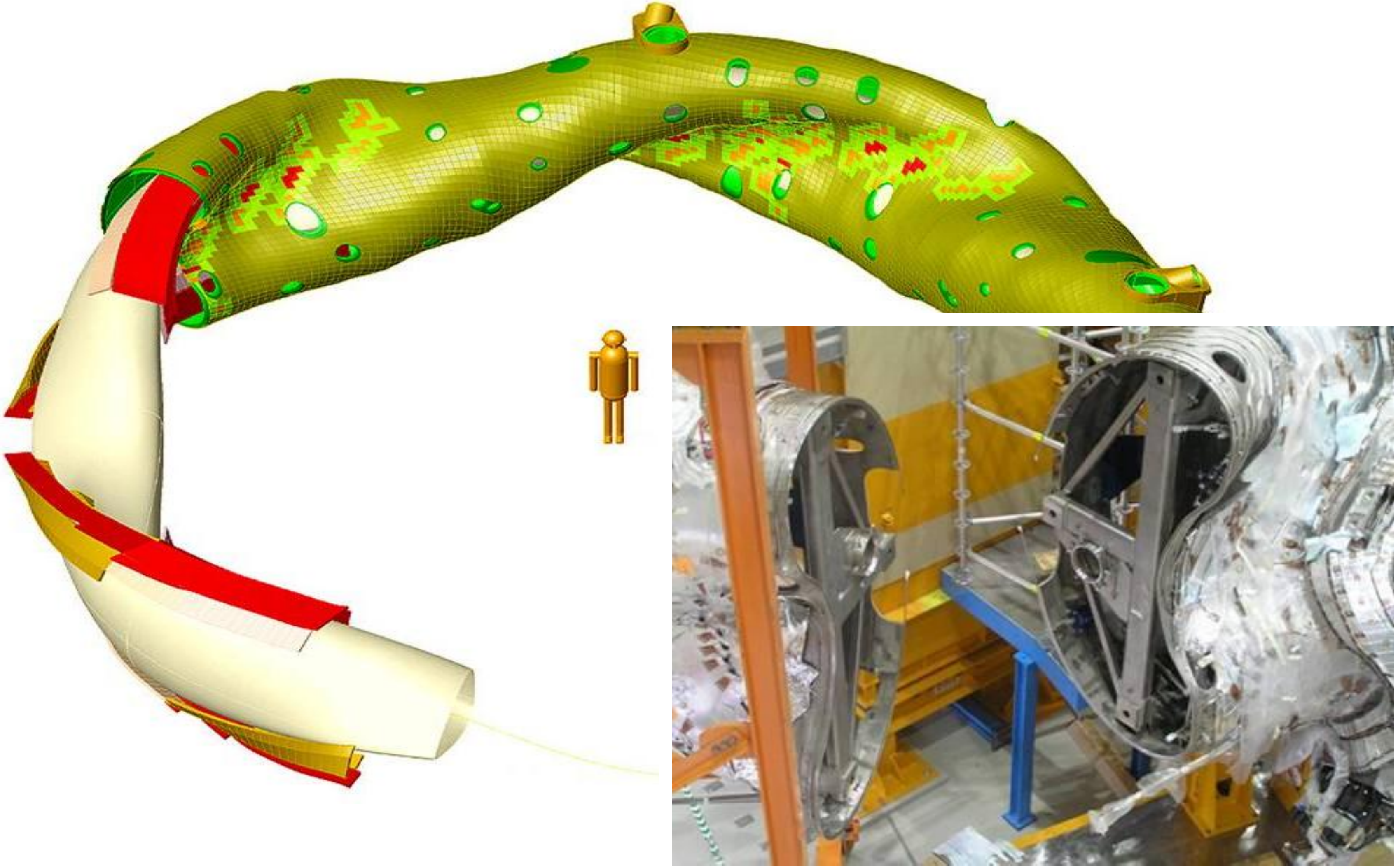
Very demanding to align !

- Neighbouring field lines can come from different directions
- Roof tile solutions like in poloidal divertors of tokamaks not possible (at least if divertor configuration can change)

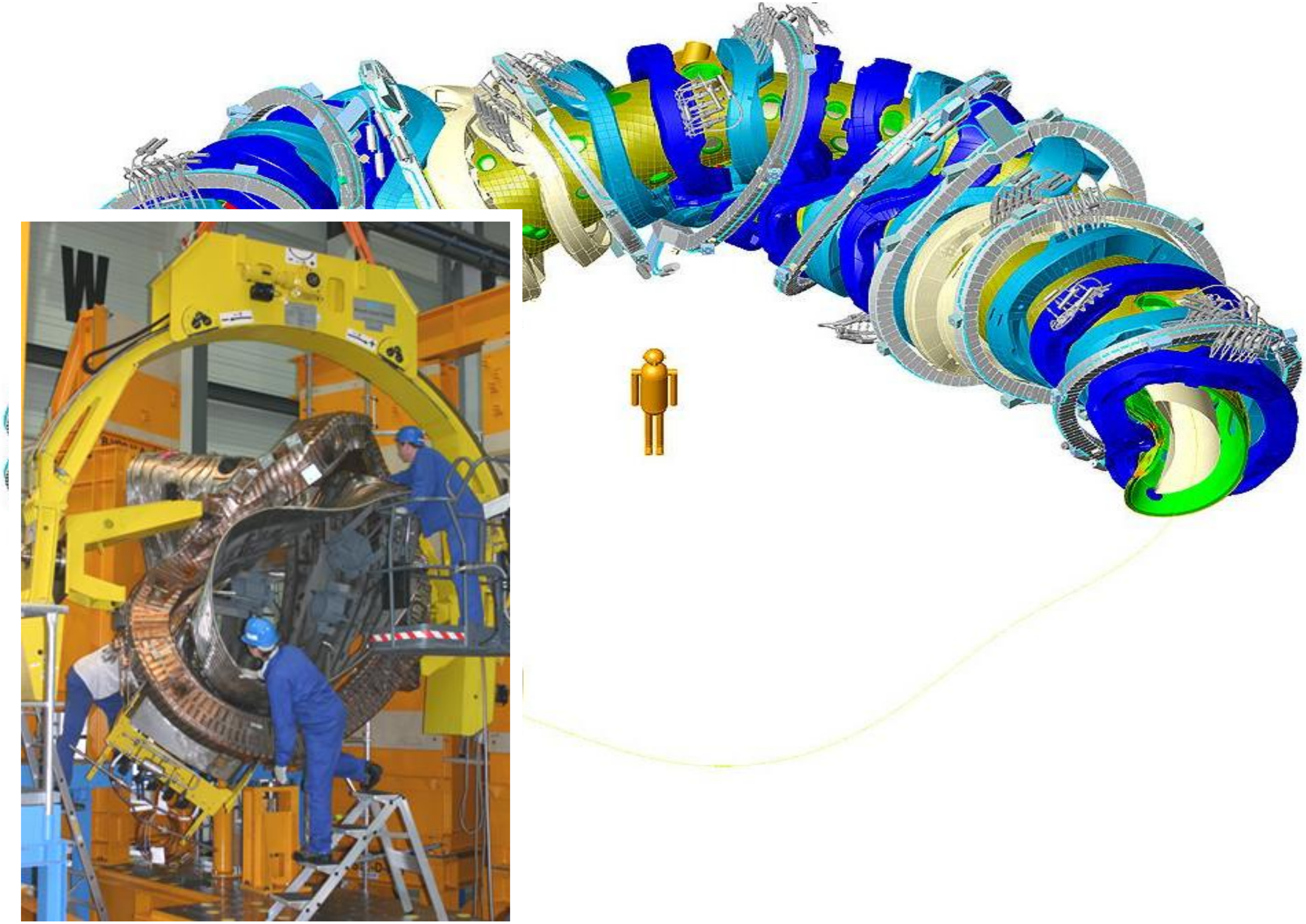




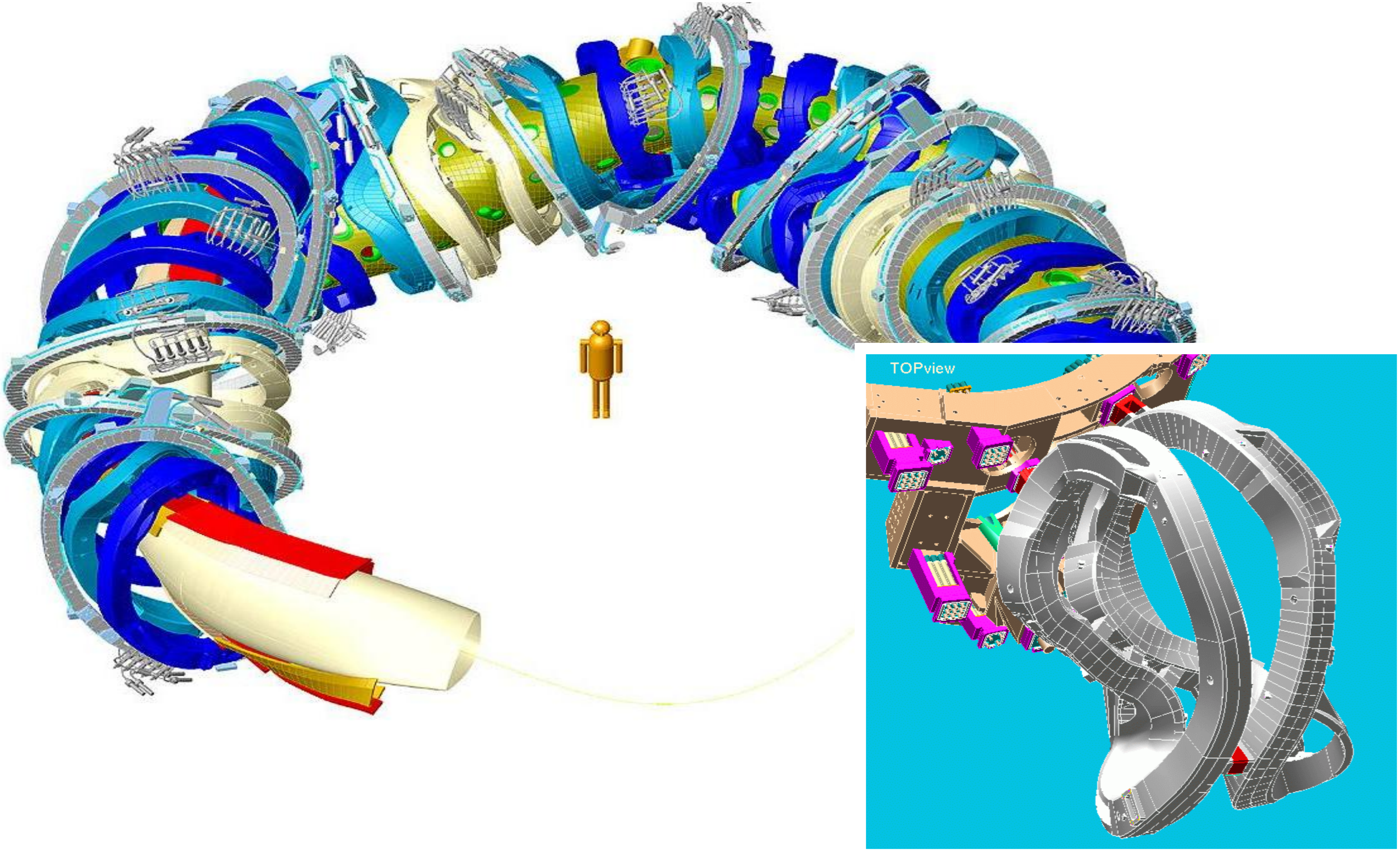
Wendelstein 7-X – plasma vessel



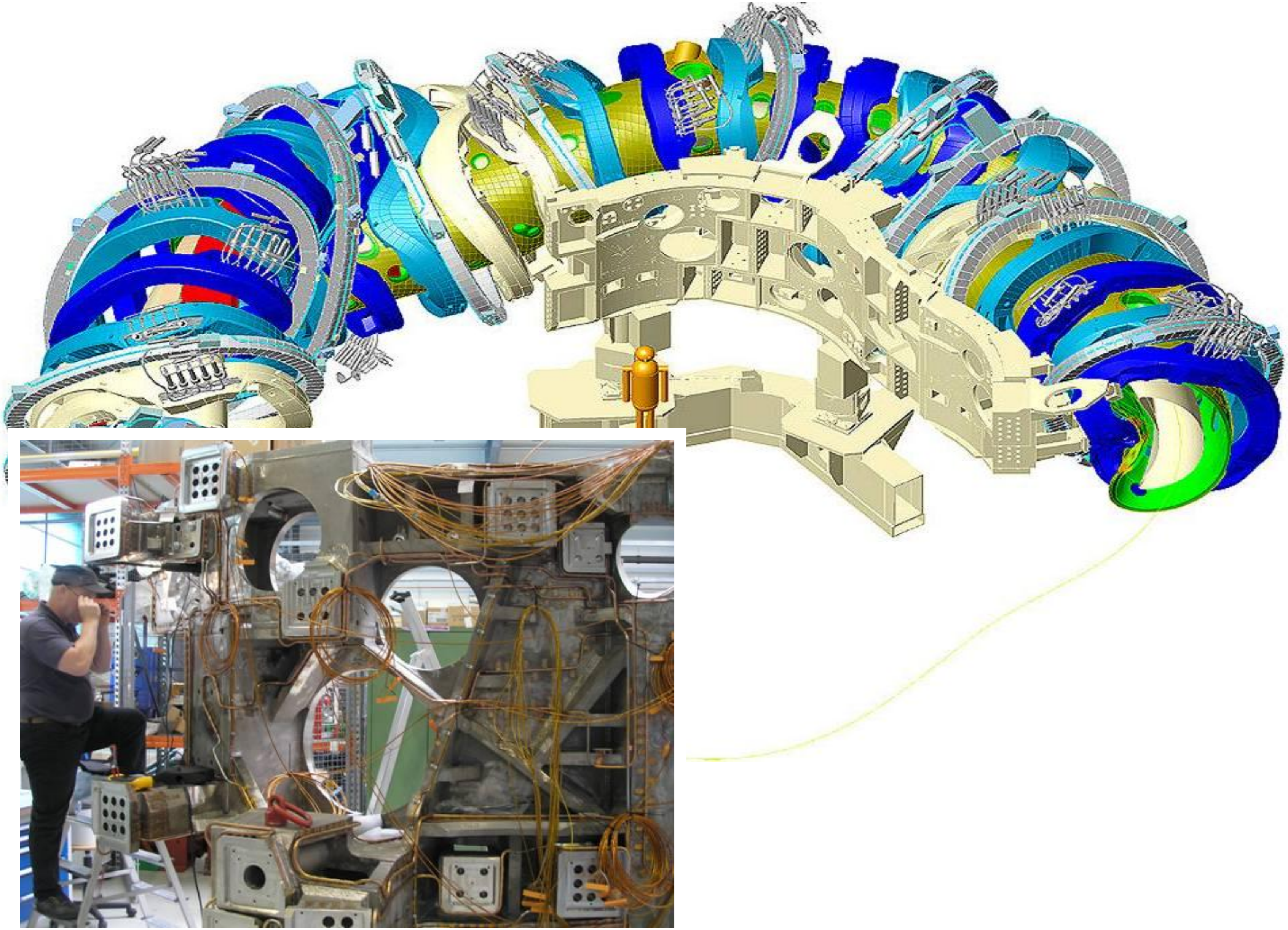
Wendelstein 7-X – superconducting coils



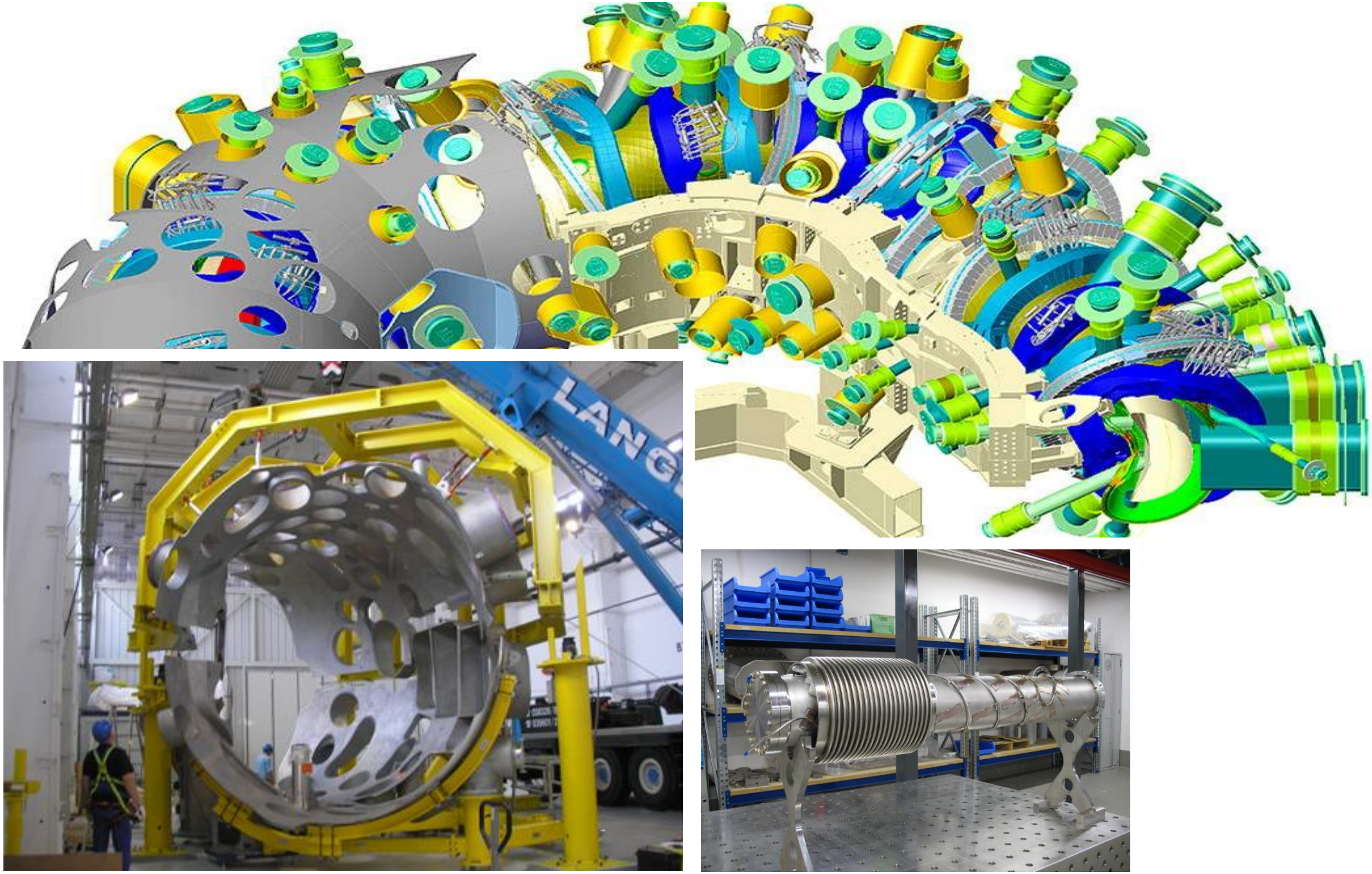
Wendelstein 7-X – coils supports



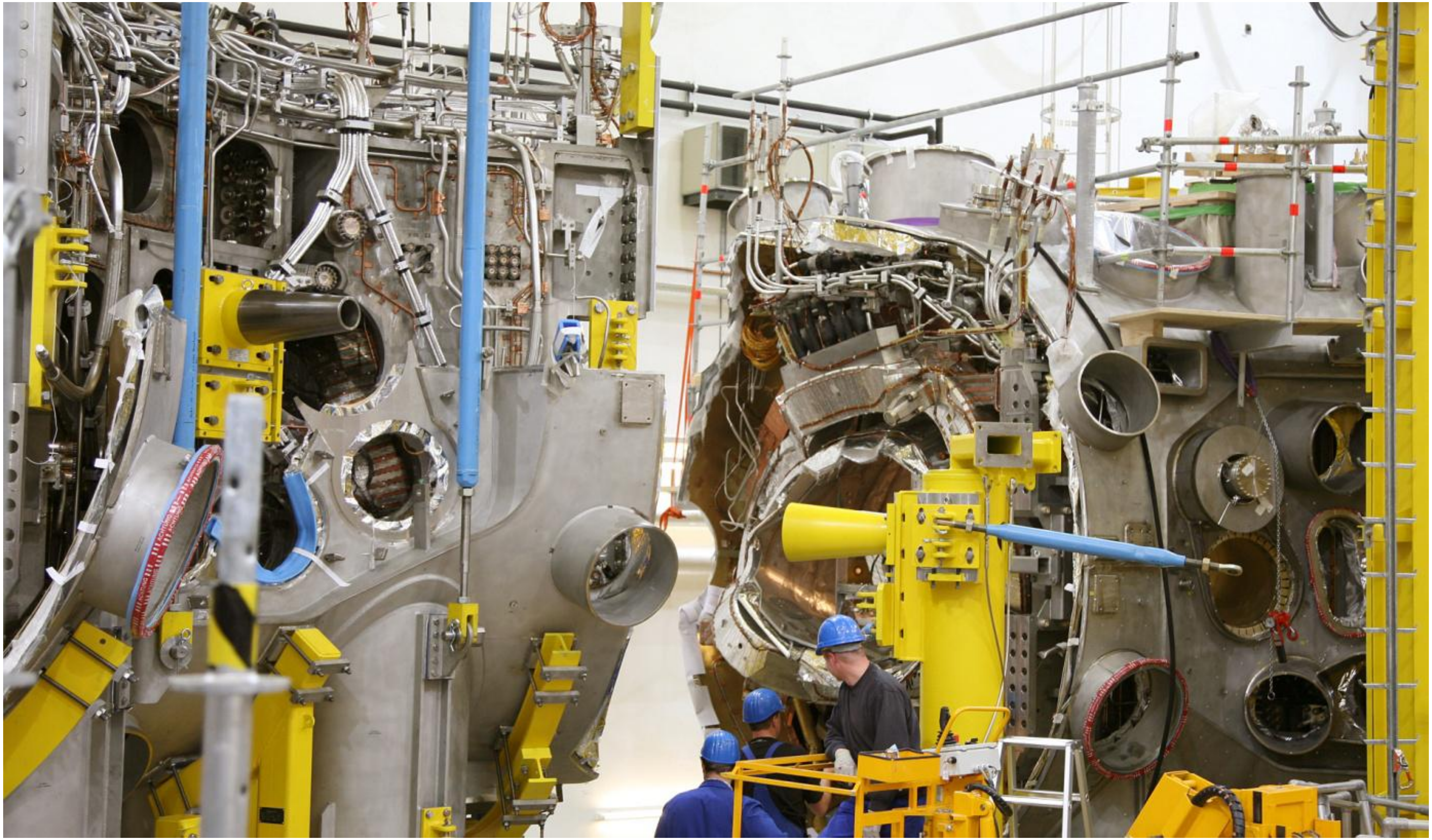
Wendelstein 7-X – coils support ring



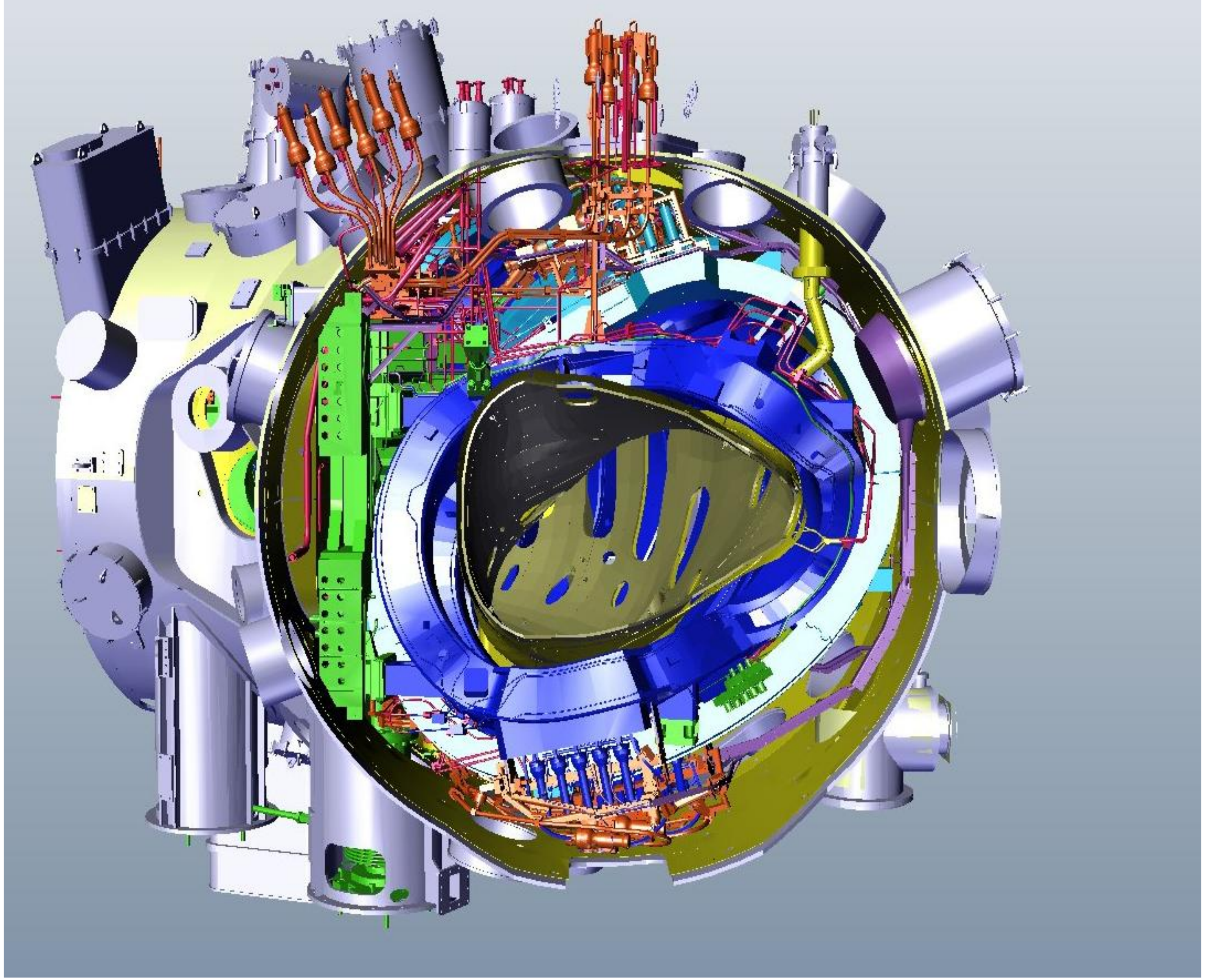
Wendelstein 7-X – cryostat vessel and ports



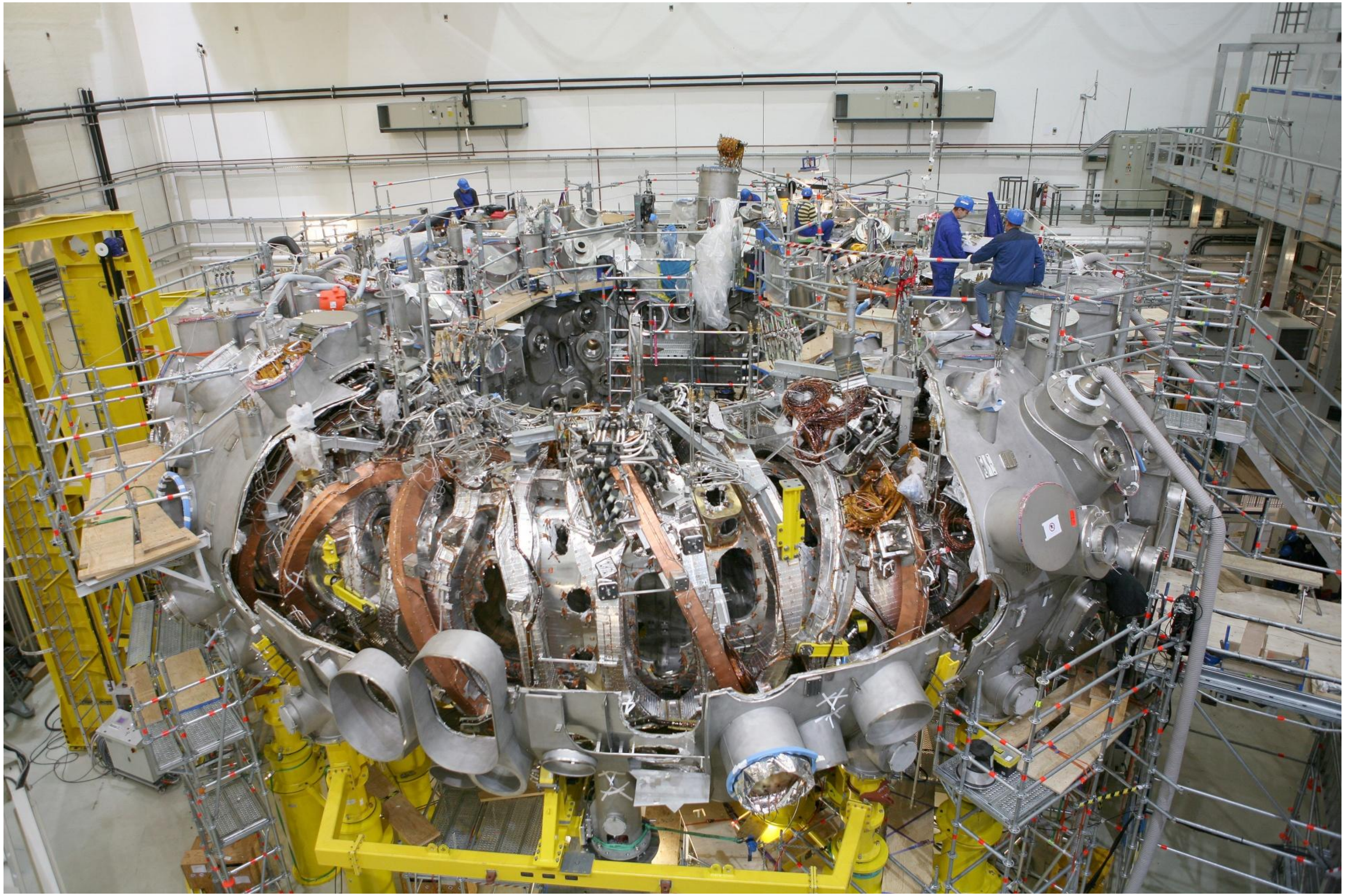
Wendelstein 7-X



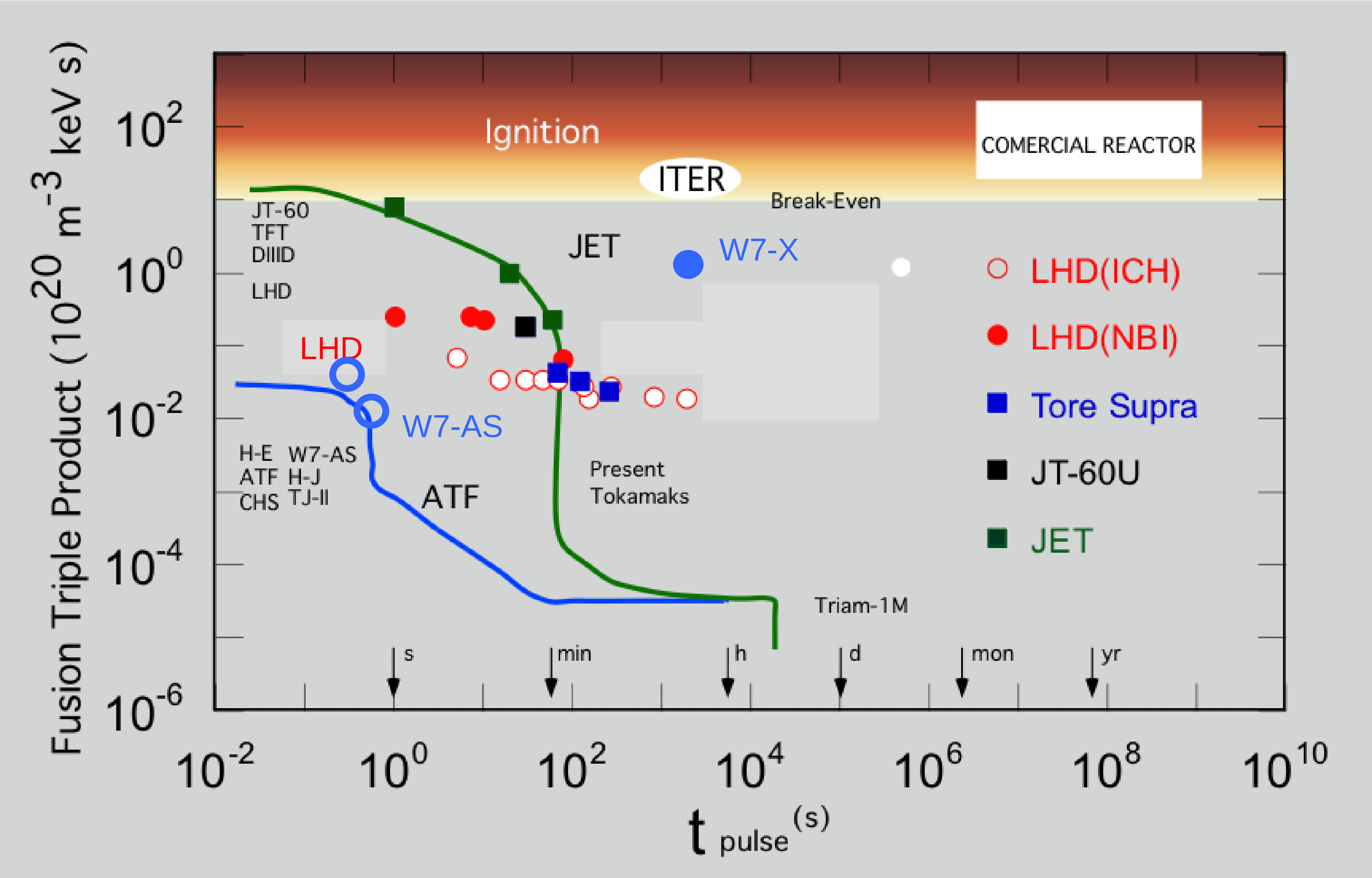
Wendelstein 7-X



Wendelstein 7-X (November 2011)



Wendelstein 7-X



Wendelstein 7-X will be equipped with steady state …

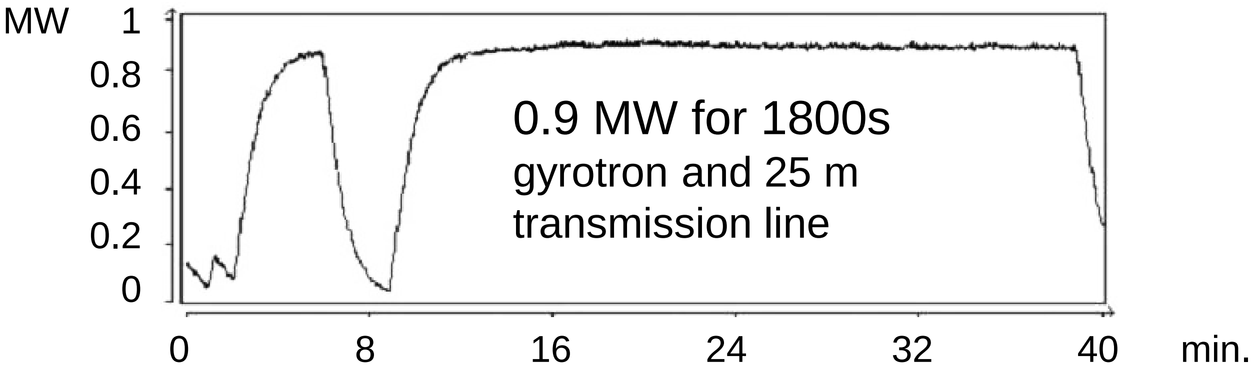
… cooling (divertor, first wall)

… heating

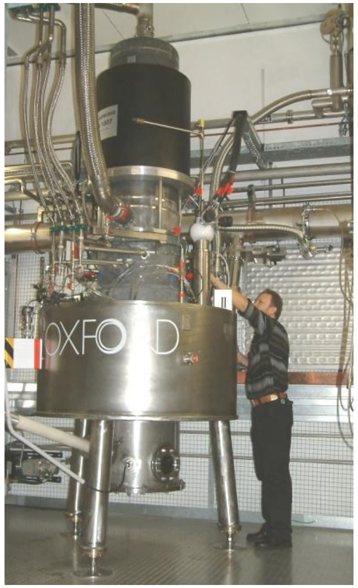
… diagnostics

… control and data acquisition

10 MW stationary micro-wave heating(140 GHz bei 2.5 T)



0.9 MW for 1800s
gyrotron and 25 m
transmission line



In the first operational phase the divertor will not be cooled (~ 10 s plasmas at 10 MW)

Contents

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  - **Wall materials compatible with heat and particle fluxes (neutron fluxes) and plasma operation, feasible exhaust concept**
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- Extrapolation to a stellarator reactor
- Summary

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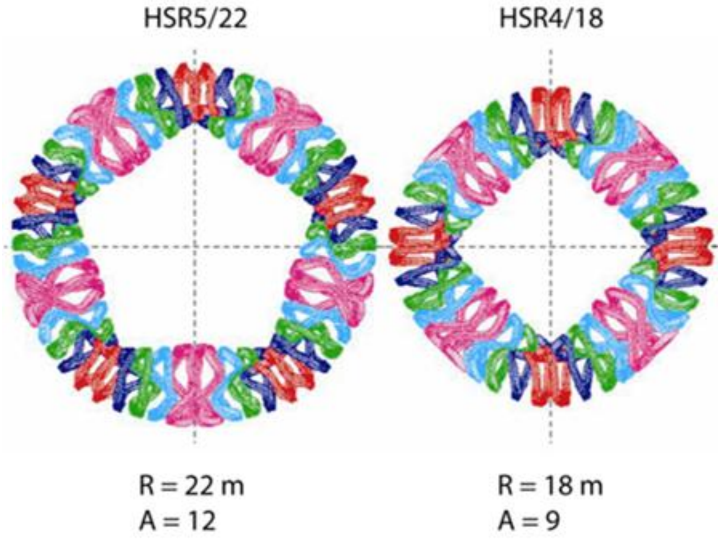
The Helical Advanced Stellarator (HELIAS) – W7-X type

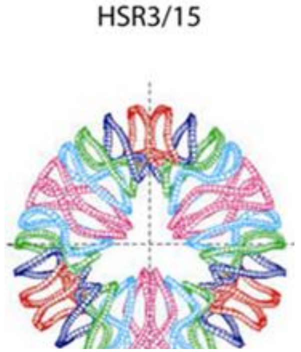
Requirements

- Sufficiently good confinement to provide ignition
- Average magnetic field on axis 5T (max. field at coils 10 T)
  → NbTi with super-fluid He at 1,8 K
  (or Nb_3Al at higher temperatures)
- Sufficient space for blanket (~1.3 m)

Consequences, additional aspects

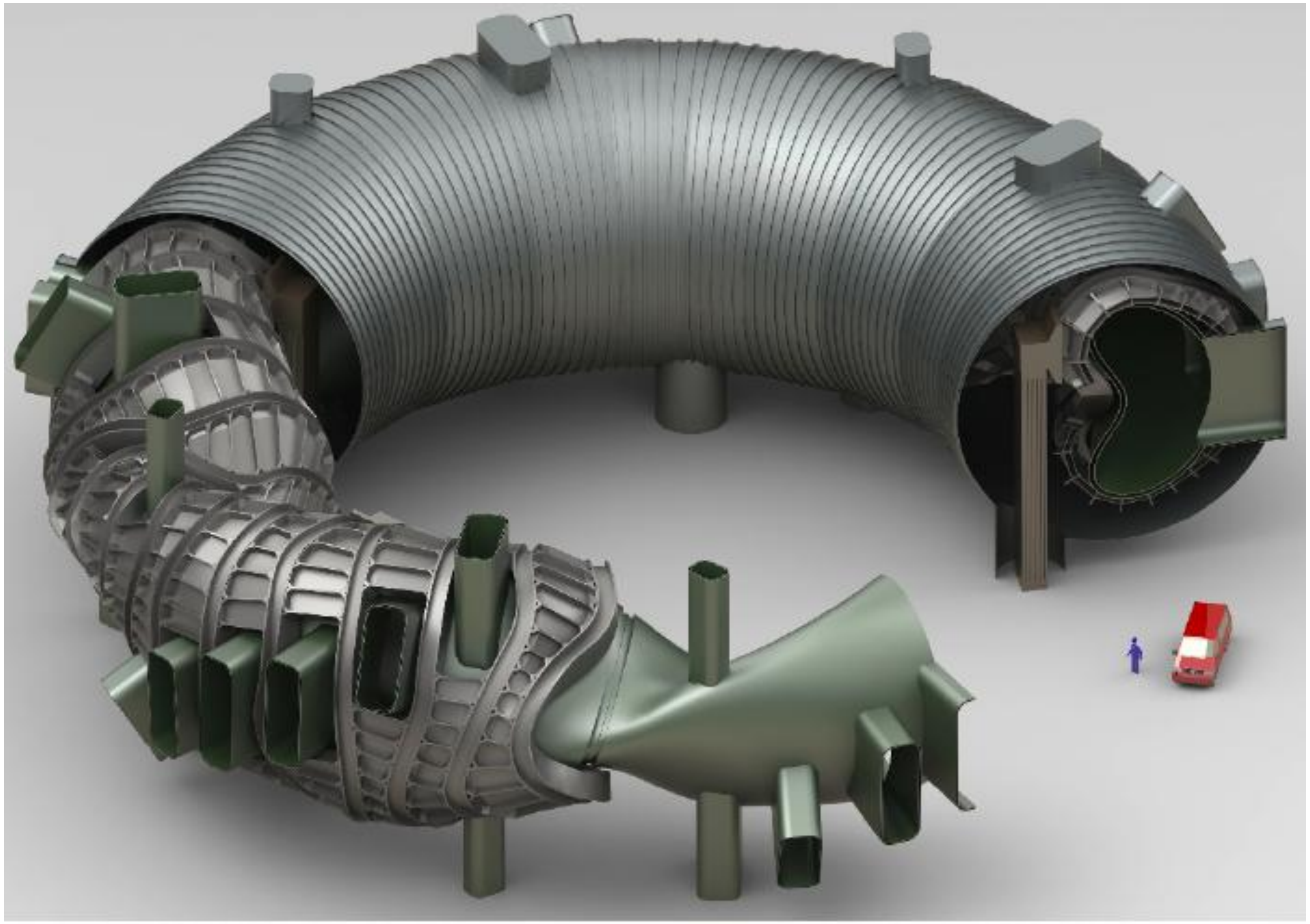
- <β> = 4 – 5 % (W7-X value!)
- Similar volumes, fusion power ~ 3GW
- Advantage of large aspect ratio
  → reduced neutron flux to the wall
  (average 1 MW/m^2, peak 1.6 MW/m^2)





α-confinement ?

HELIAS engineering study

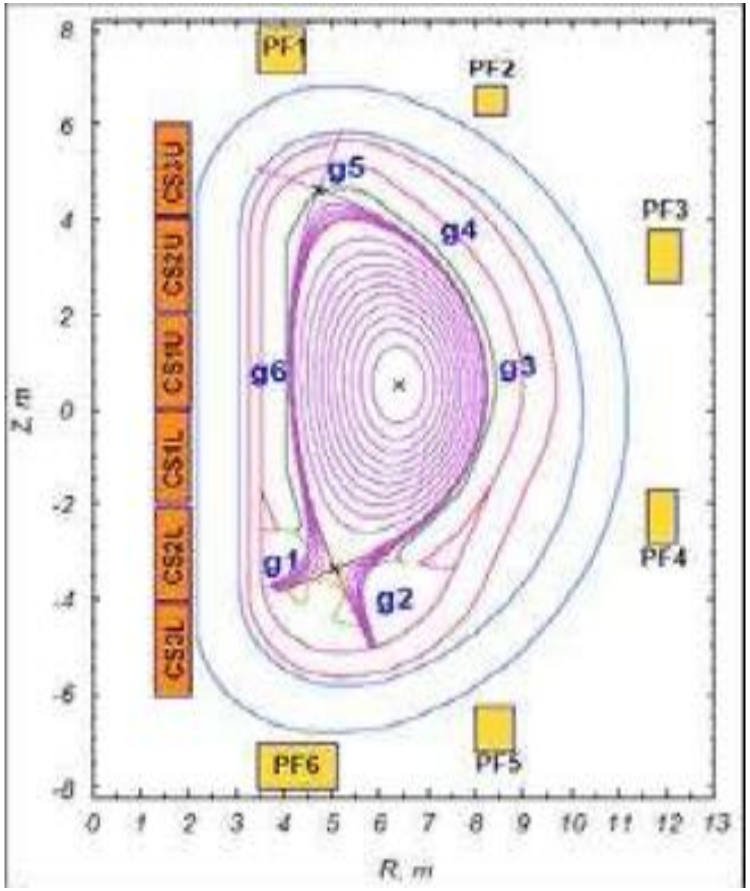


courtesy K. Egorov, F. Schauer, 2011

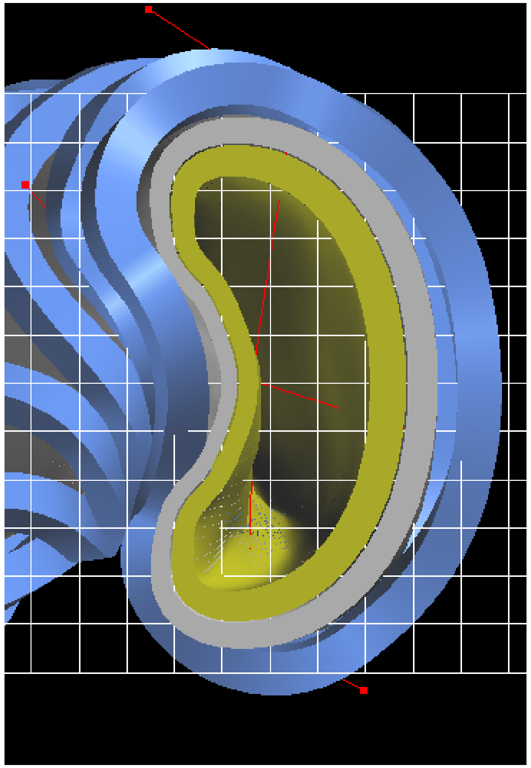
Assuming ITER coil technology

Comparison of ITER and HSR5 coils

(same scale)



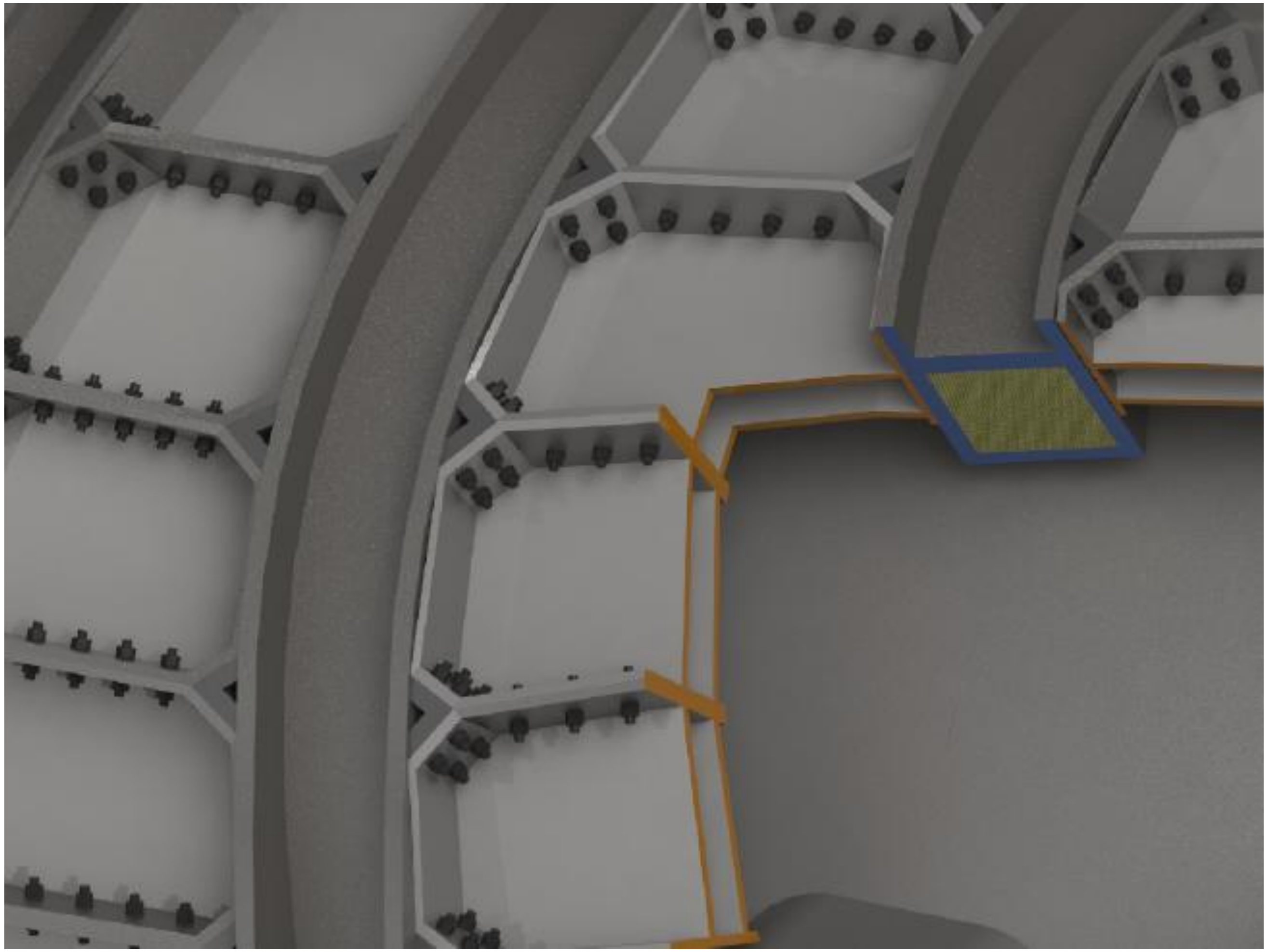
ITER toroidal field (TF) coil



HSR50a coil #5

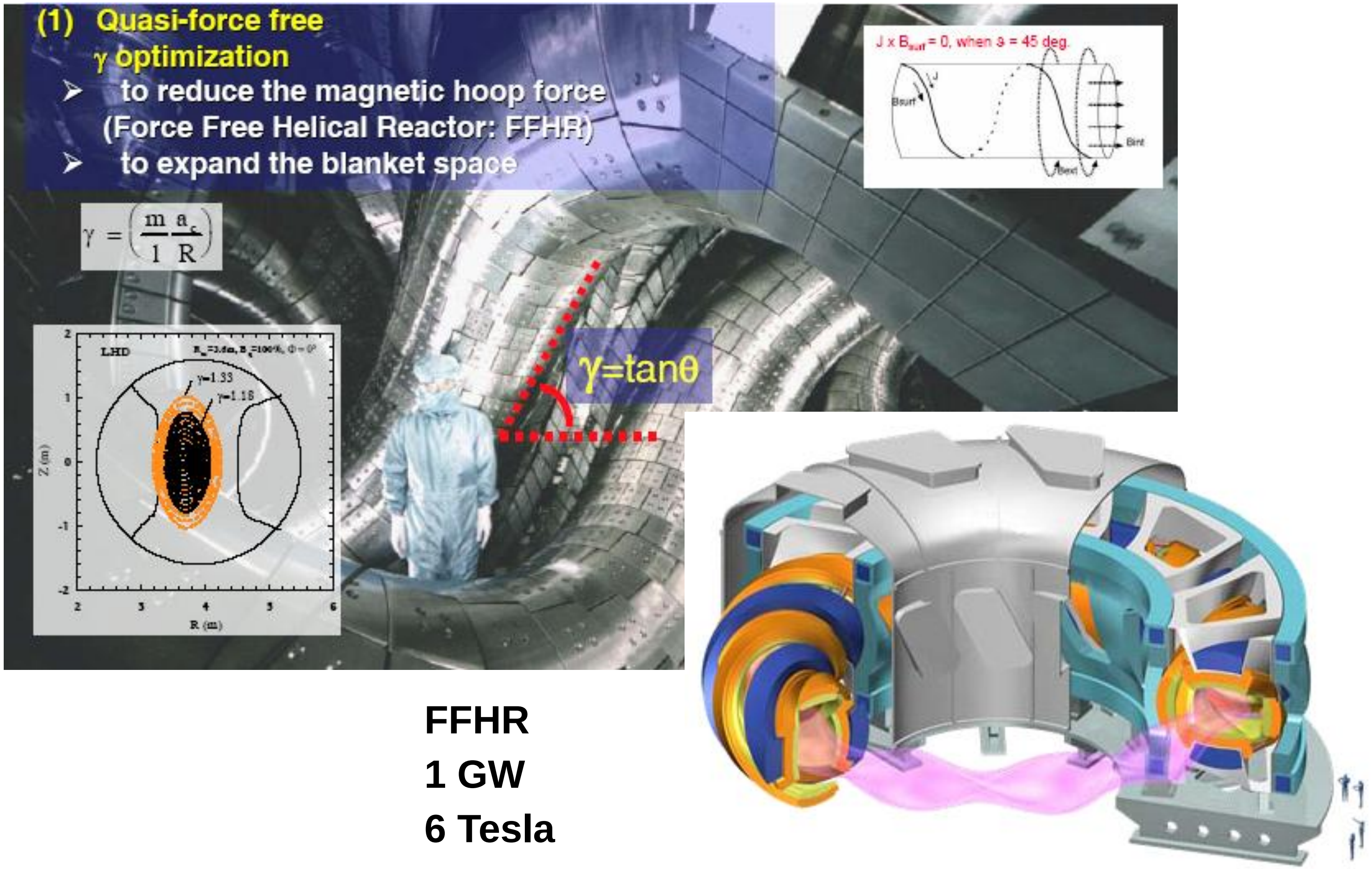
from F. Schauer et al., PFR 2010

Building block structure for coil support and maintenance



courtesy K. Egorov, F. Schauer, 2011

Force Free Helical Reactor (FFHR) – LHD type



(1) Quasi-force free
γ optimization

- to reduce the magnetic hoop force (Force Free Helical Reactor: FFHR)
- to expand the blanket space

$$\gamma = \left(\frac{m}{l}\frac{a_c}{R}\right)$$

$\gamma = \tan\theta$

FFHR

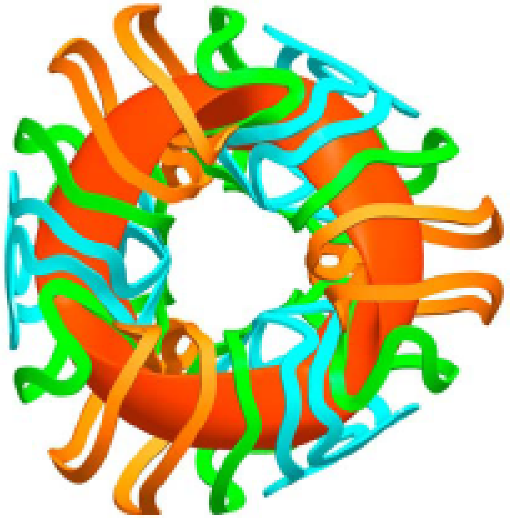
1 GW

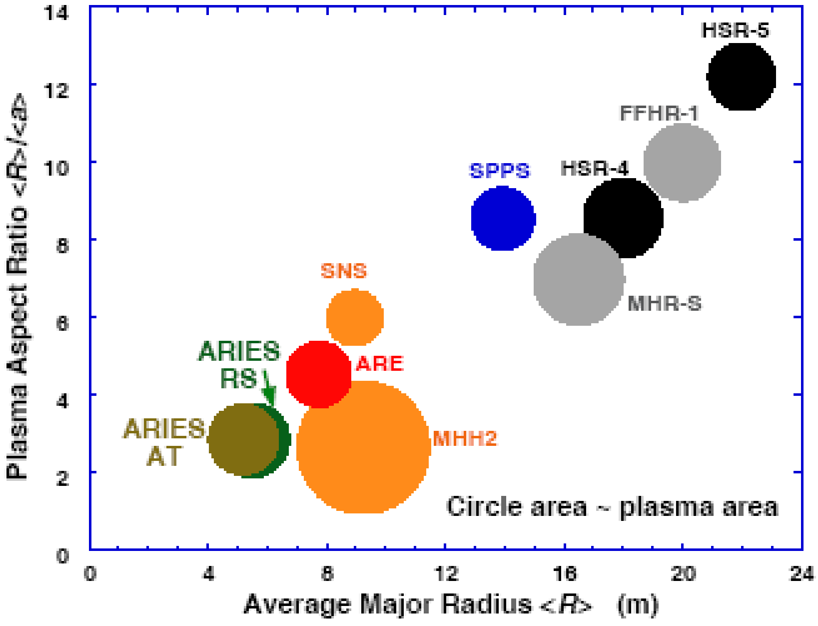
6 Tesla

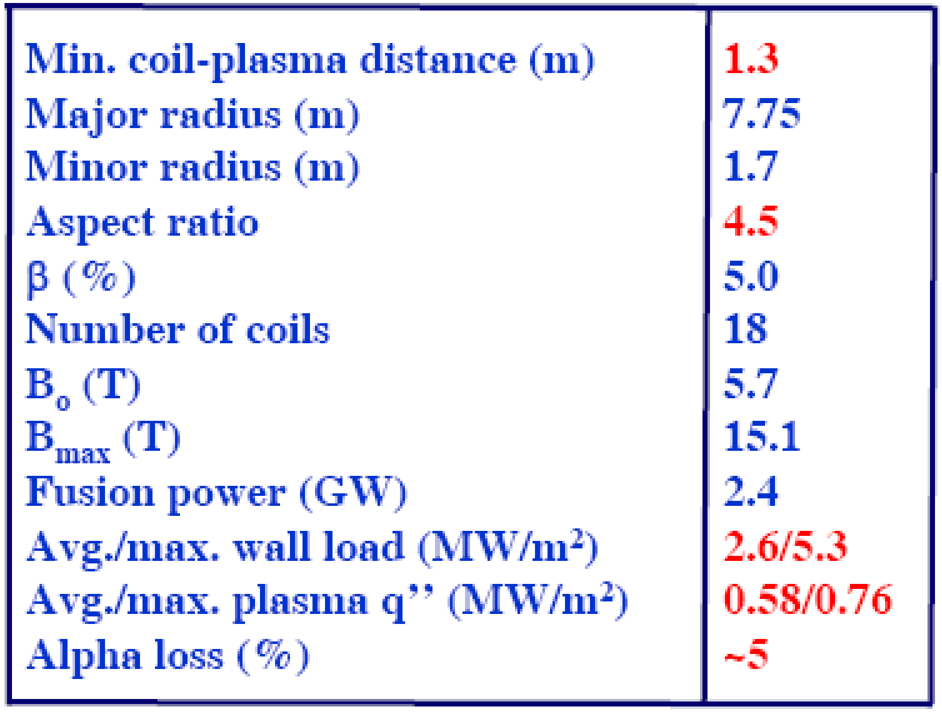
25,000 tons

Motojima et al. 2007

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/m^2)	2.6/5.3
Avg./max. plasma q'' (MW/m^2)	0.58/0.76
Alpha loss (%)	~5

Raffray et al. 2007

UC San Diego

Summary

- Many advantageous and not so advantageous properties of the stellarator have been demonstrated
- To achieve
  - **Sufficient confinement of thermal plasma and fast ions (α-particles in a fusion reactor)**
  - **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**

  at the same time requires optimization procedure
- W7-X is the first stellarator (magnetic confinement experiment) the design of which is derived from optimization criteria
- Its objective is to demonstrate the reactor capability of the stellarator concept
- Extrapolation to a stellarator reactor requires experimental (W7-X and hopefully more optimized stellarators) as well as engineering input (divertor, blanket, maintenance, remote handling in 3 D)