Stellarators and Stellarator Physics

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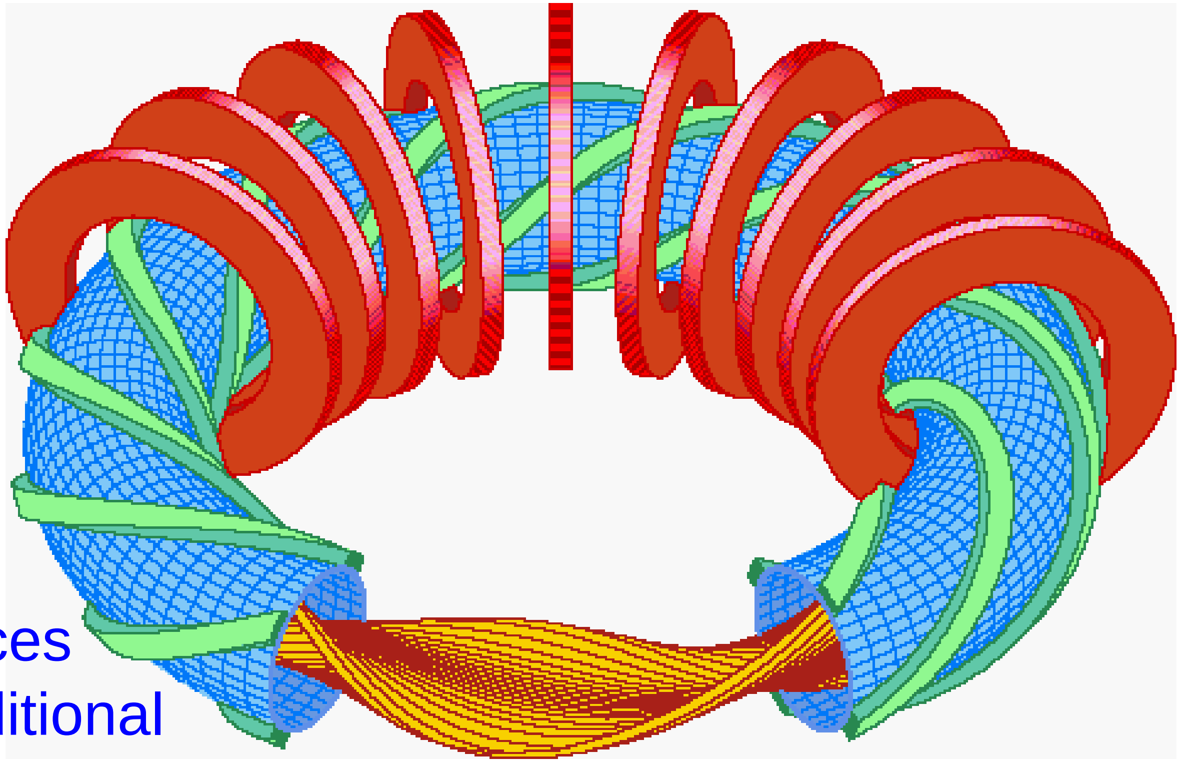
Outline

- **What is a stellarator?**
  - **Twisted magnetic fields and different types of stellarators**
- **(Performance-limiting) plasma transport in stellarators**
- **Advantages and disadvantages of stellarators over tokamaks ⇒ is one concept going to win the race?**
- **Field-optimized stellarators and the Wendelstein 7-X project**
- **Cost of fusion electricity**

A stellarator is magnetic confinement system based on currents solely driven external coils

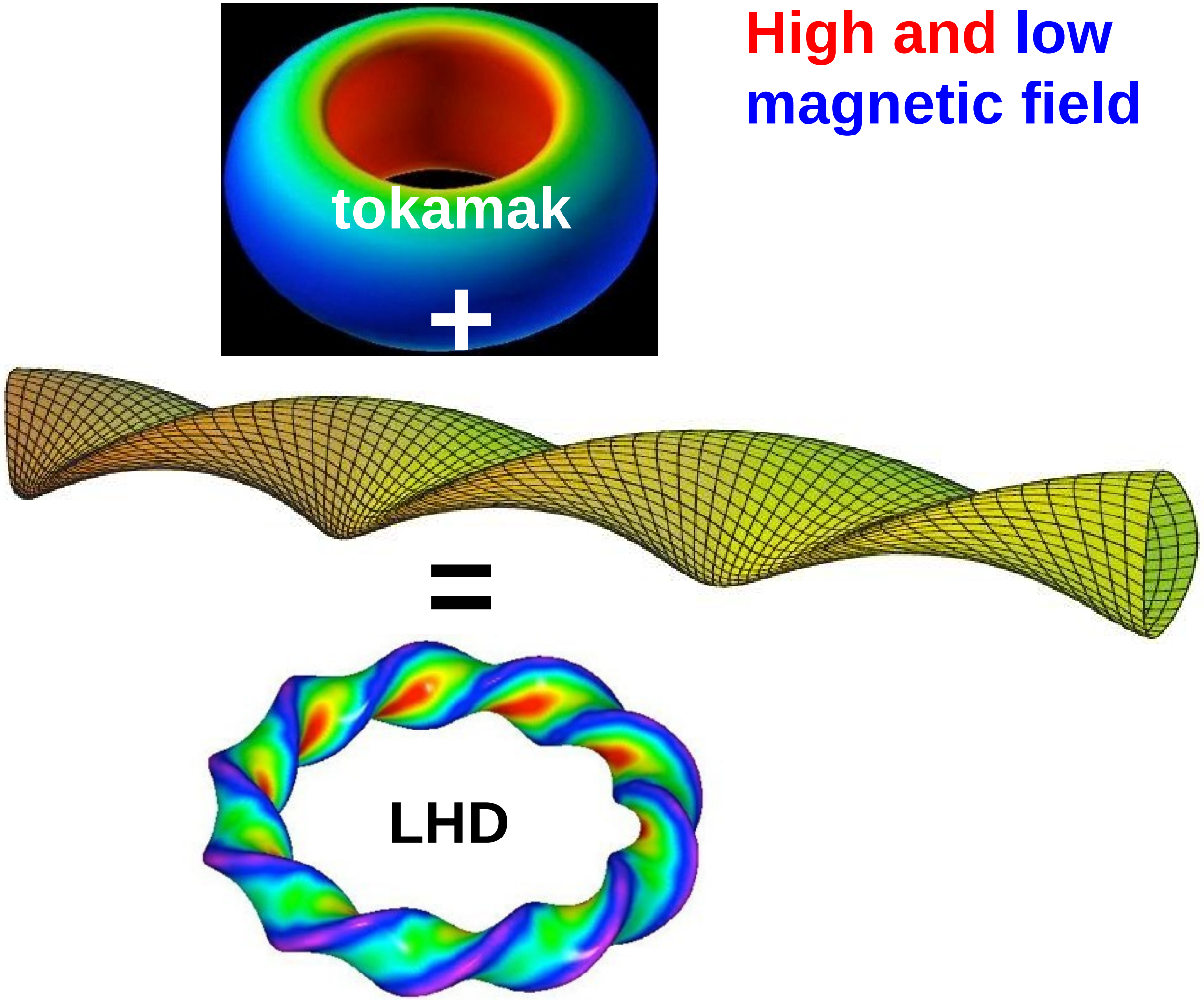


- **Conceptual advantages:**
    - Inherently steady-state
    - No current disruptions (or current-driven instabilities)

- **Drawbacks/opportunity:**
    - No guaranteed flux surfaces
    - Due to 3-D geometry, additional losses, complexity, localized heating of wall

Having to deal with a full 3-D magnetic field configuration allows dedicated design of it

- **Magnetic confinement requires:**
  - Nested flux surfaces
  - Finite toroidal transform

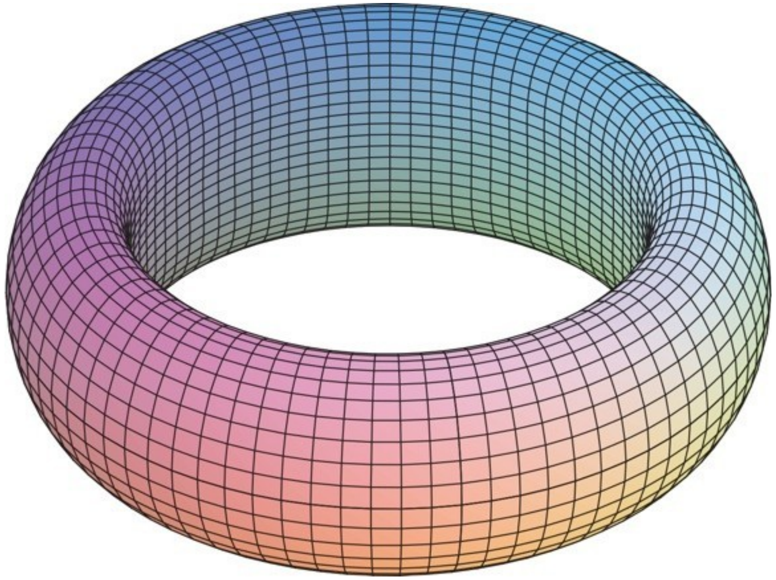


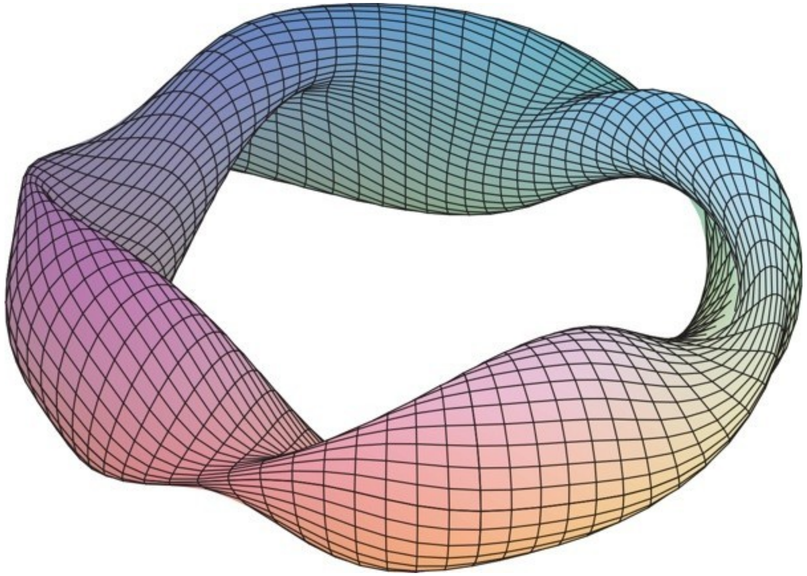
High and low magnetic field

In a tokamak toroidal symmetry is preserved, in stellarators imposed

Tokamak

Stellarator





- **Axisymmetry**

- **Periodicity $\varphi \rightarrow \varphi+2\pi/P$**
  (P: number of field periods)

- **Stellarator (flipping) symmetry: $(\varphi,\theta) \rightarrow (-\varphi,-\theta)$**

Tokamaks and stellarators produce two different types of rotational transforms

Tokamak

- transform produced by plasma current
- transform decreases with radius (safety factor increases)
- Axisymmetric

plasma current

- 2-D configuration

- Current-driven instabilities and disruptions
- Pulsed

Stellarator

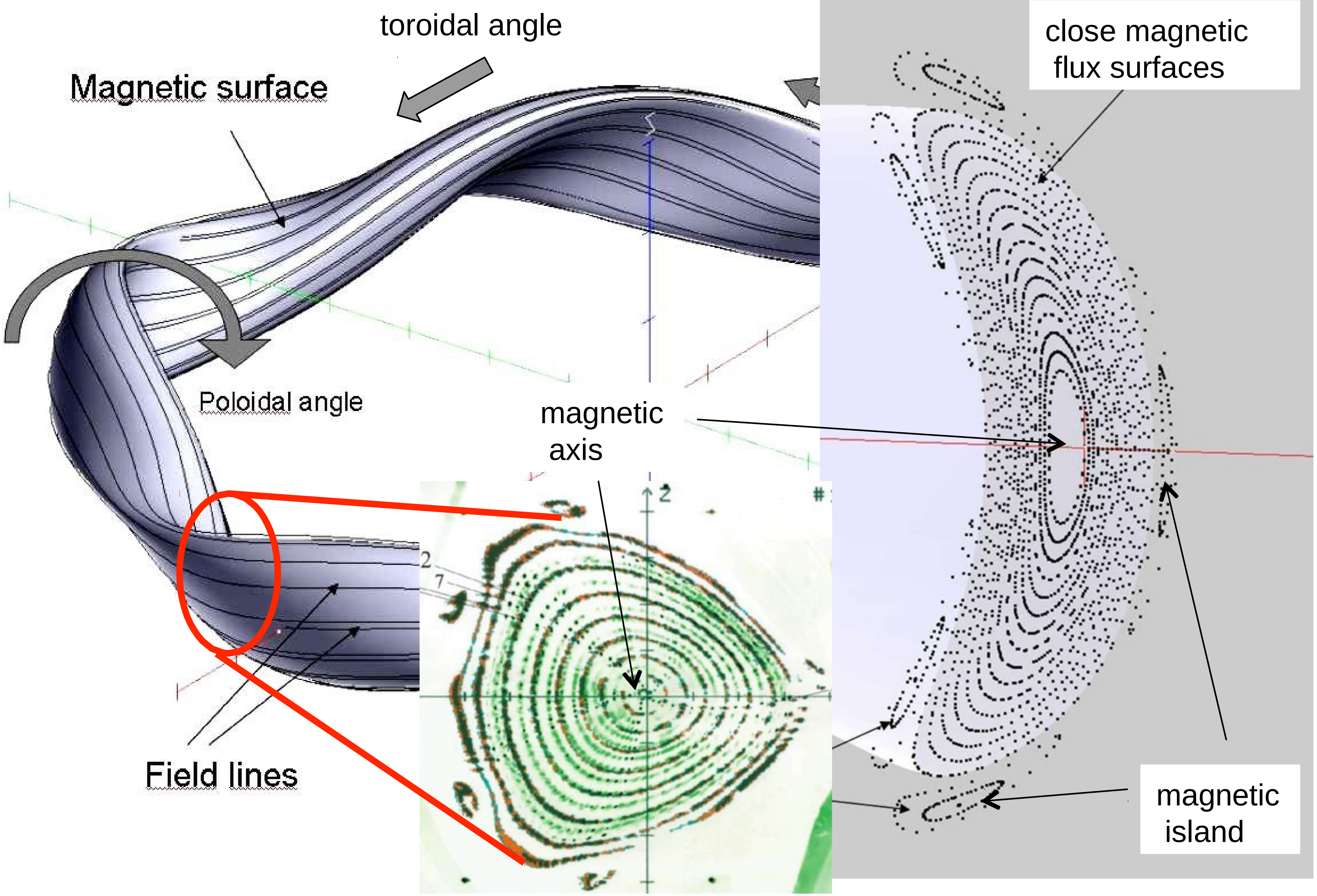
- transform produced by external coils
- transform increases with radius

no externally driven TOTAL toroidal current

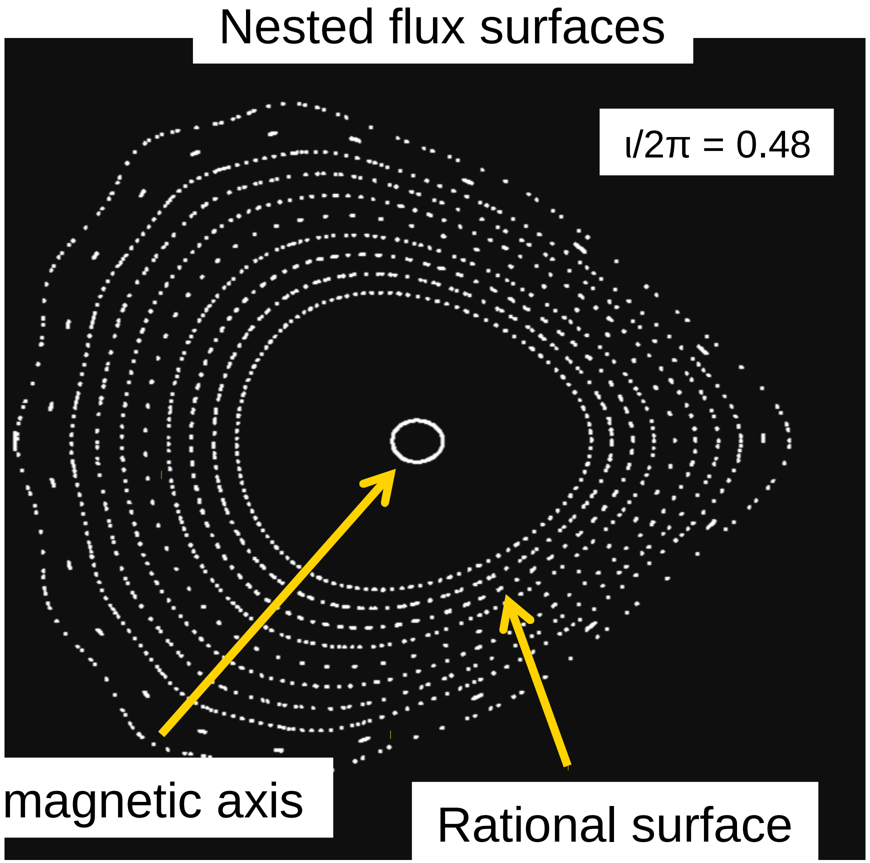
- 3-D configuration by definition ⇒ complex, prone to higher radial transport losses
- No disruptions

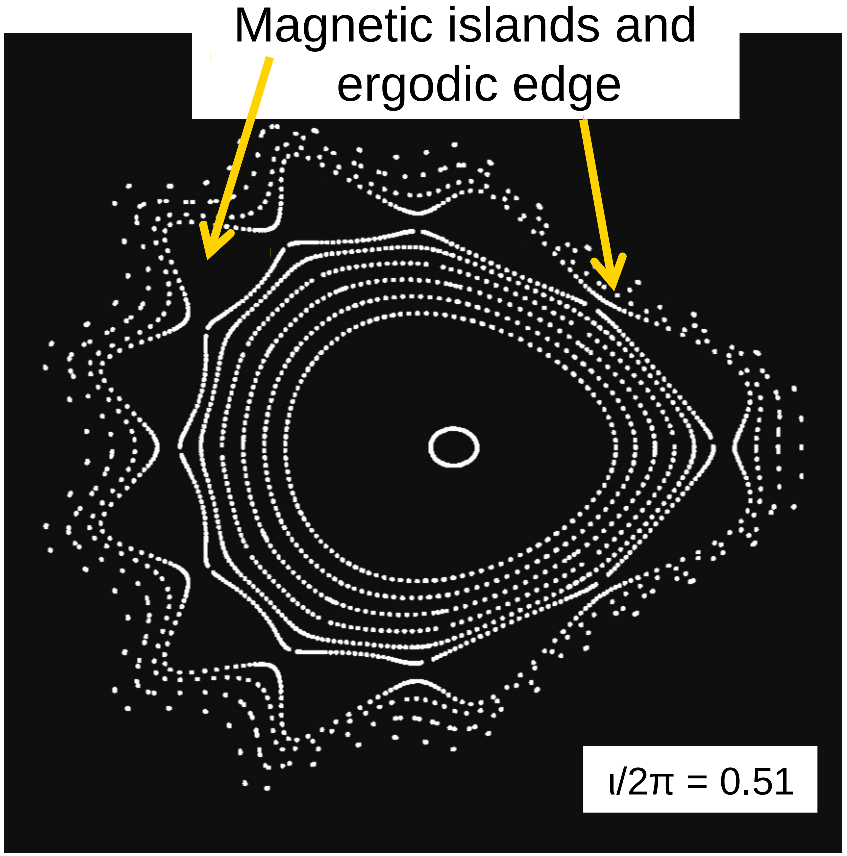
- Steady-state

Magnetic confinement in a stellarator is toroidally asymmetric



Closed field lines exist at rational values of m toroidal and n poloidal transit $\iota/2\pi = m/n$





- **Rotational transform: $R \langle B_\theta \rangle / r_{eff} \langle B_\Phi \rangle$**
  - Local pitch angle may vary strongly on flux surface

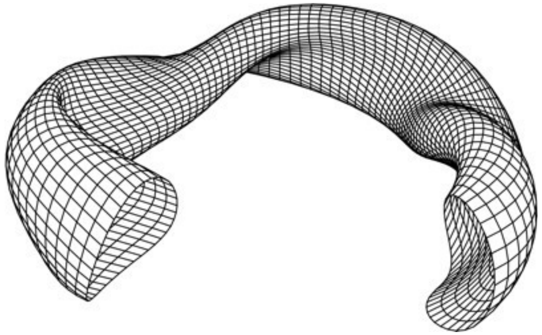
The stellarator equilibrium can be derived from the standard MHD equations

- **Equations (as for tokamaks):**

$$j \times B = \nabla p$$

$$\nabla \times B = \mu_0 j$$

$$\nabla \bullet B = 0$$



- **Equilibrium determined by:**
  - Radial profiles (e.g., pressure, total toroidal current J=0)

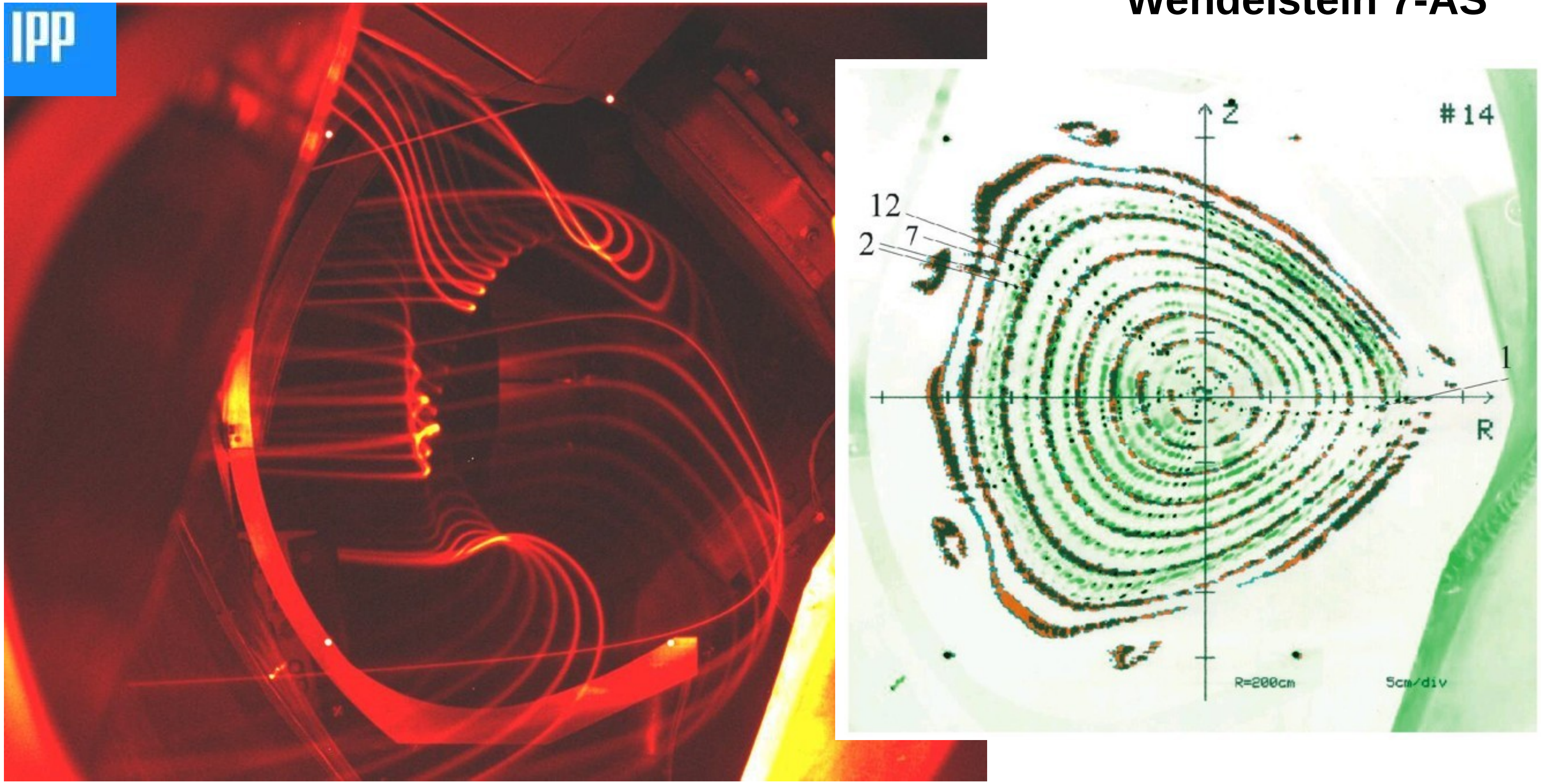
  ⇒ Outer flux surface can be parameterized (in cylindrical coordinates (R, Z, Φ) with periodic conditions

- **Boundary conditions: B tangential to surface**

⇒ Solution of MHD equations inside surface

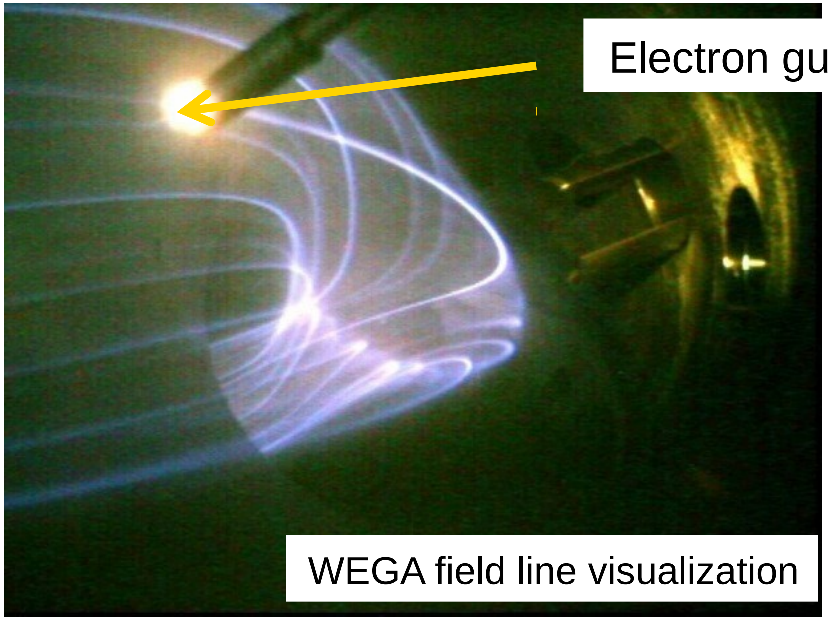
Field lines can be visualized using an electron beam in a hydrogen gas

Wendelstein 7-AS

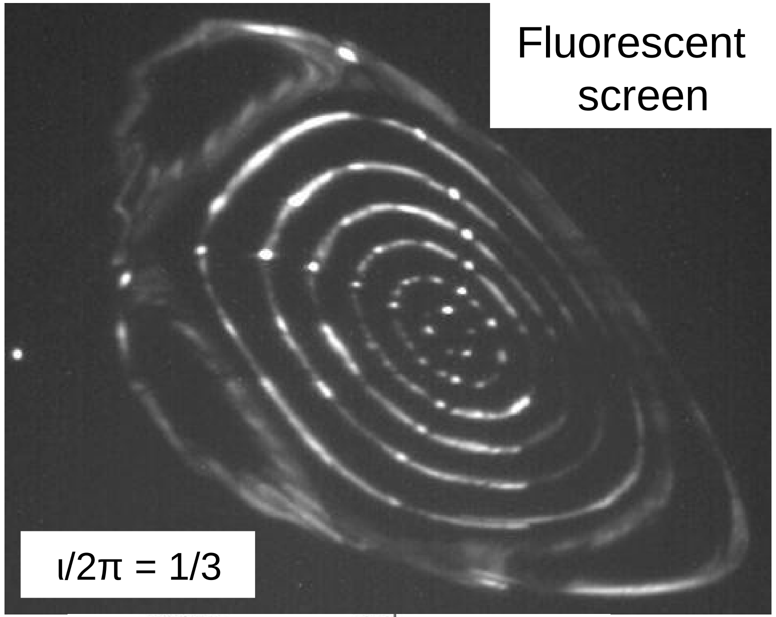


- **Structures of magnetic field: shear, island, ergodic regions ⇒ shortcuts of transport to wall**

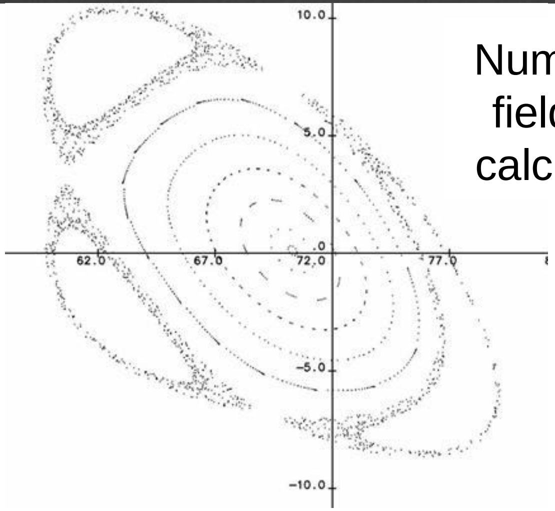
There is generally no analytic proof of existence of flux surfaces in helical devices ⇒ field line tracing


Electron gun

WEGA field line visualization


Fluorescent screen

$\iota/2\pi = 1/3$



Numerical field line calculation

- **Electrons emitted parallel to B in vacuum field without plasma ⇒ fluorescent projector and interaction with (Ar) background gas**

Three basic types of stellarators

All helical confinement concepts revolve around the question of how to build 3D toroidal flux surfaces

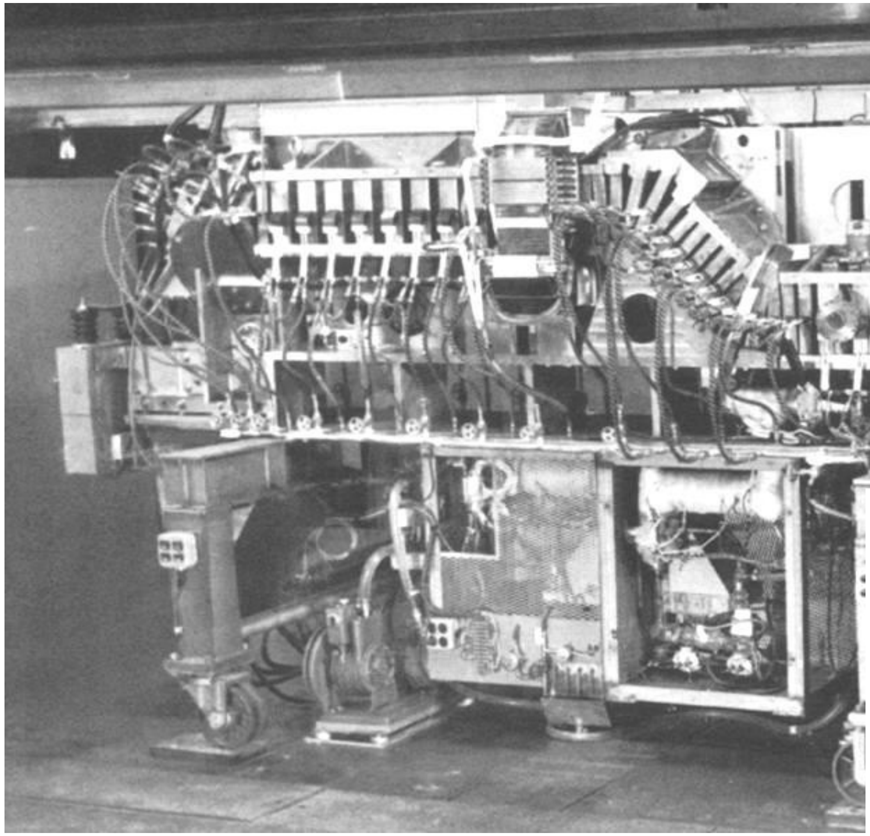
- **Three basic types of systems**
  - Heliotrons, “classical” stellarators, heliacs

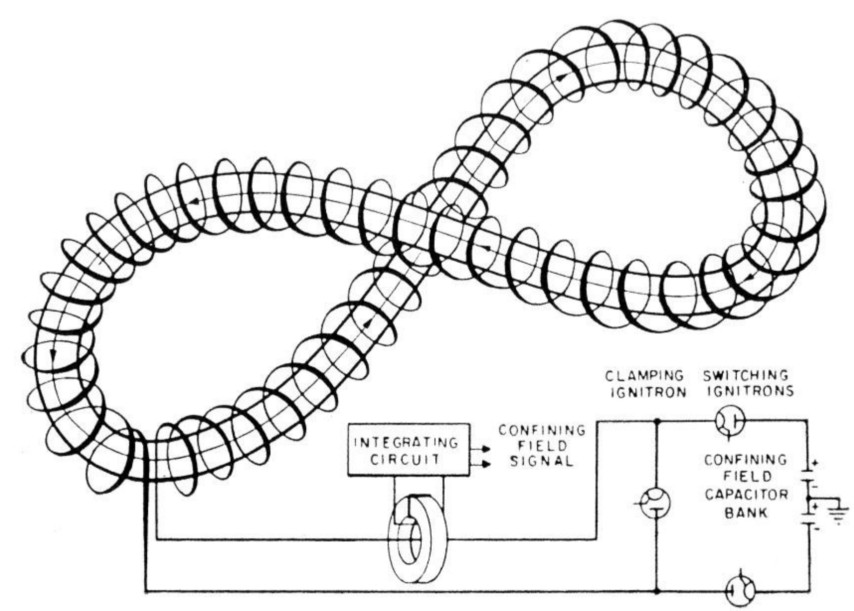
- **Principle research questions are very similar**
  - Design vacuum field (and coils) w/ good flux surfaces
  - Reduce particle losses (drifts) in 3D geometry (fast particles, neoclassical transport, trapped particles) ⇒ similar to tokamaks
  - Operation at maximum density (and pressure)

⇒ For steady-state, additional issues, such as power exhaust and impurity control exist

⇒ Second-generation stellarators include modular coils

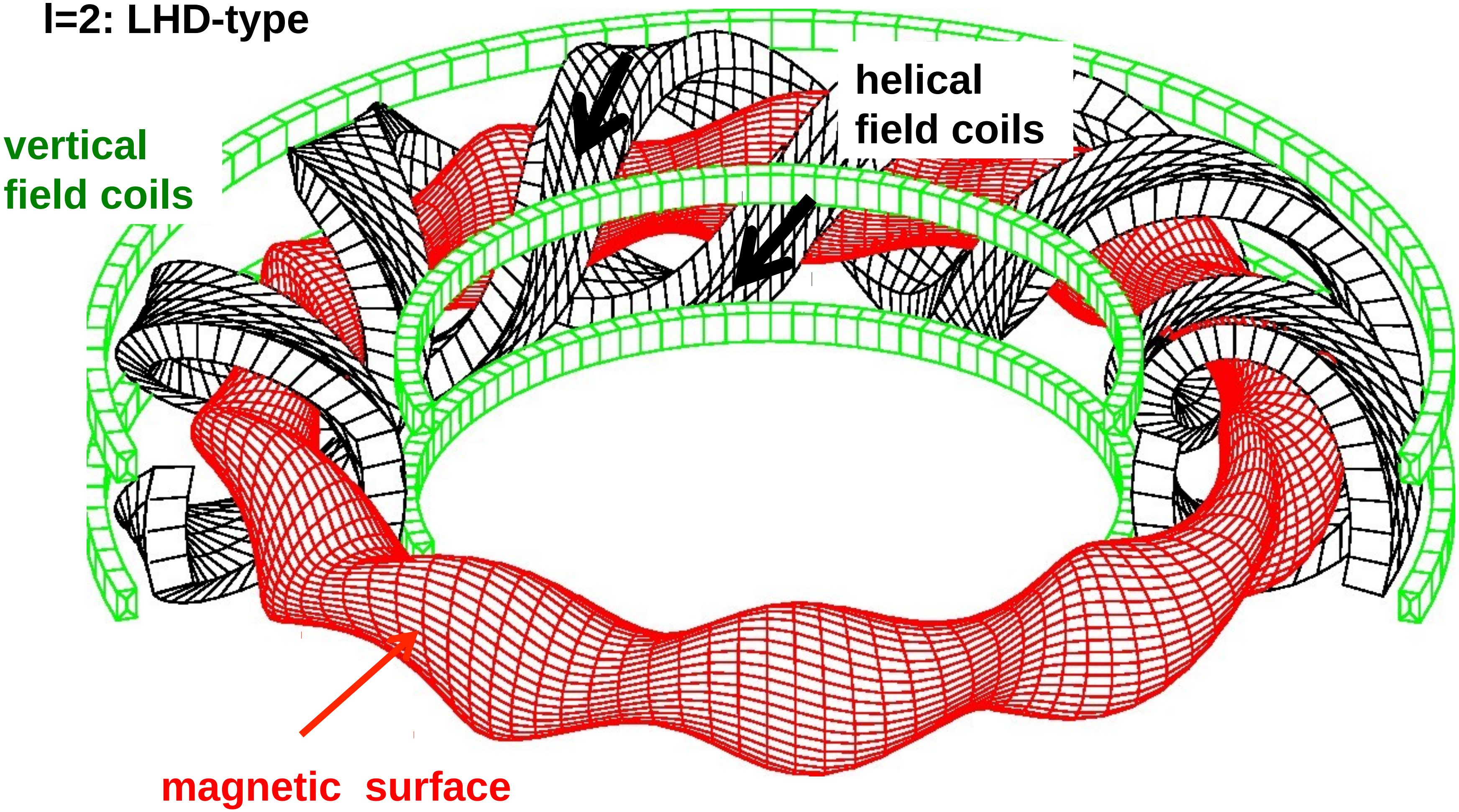
Twisting the torus and hence magnetic field produced helicity (Princeton Figure-8 stellarator)





- **Proposed by Lyman Spitzer in 1951 ⇒ poor confinement**

A heliotron, or torsatron, is a stellarator with a circular axis and helically twisted coils



- **Vertical field needed to counteract helical field**

The Large Helical Device (LHD) is an example of an heliotron

- **LHD dimensions: R=3.5 to 4.1 m, volume= 28 m^3**
- **Primary device and line of stellarator research in Japan**

helical coils, superconducting

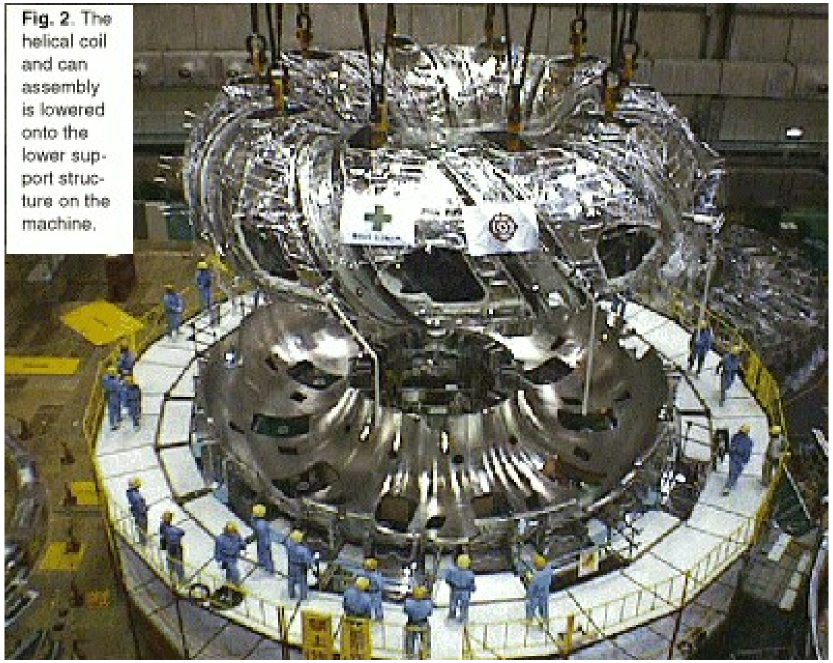
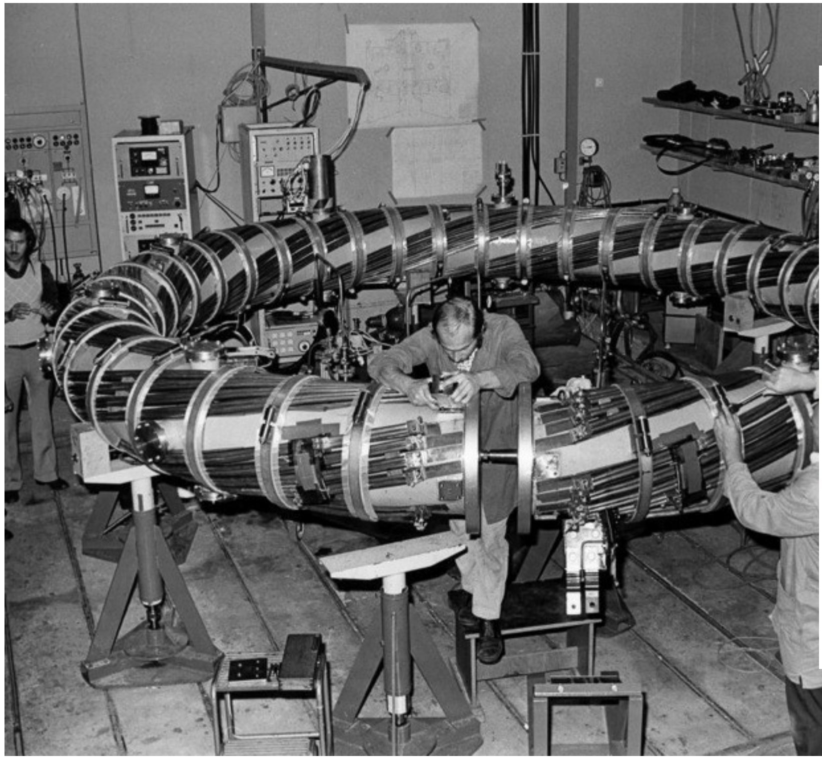


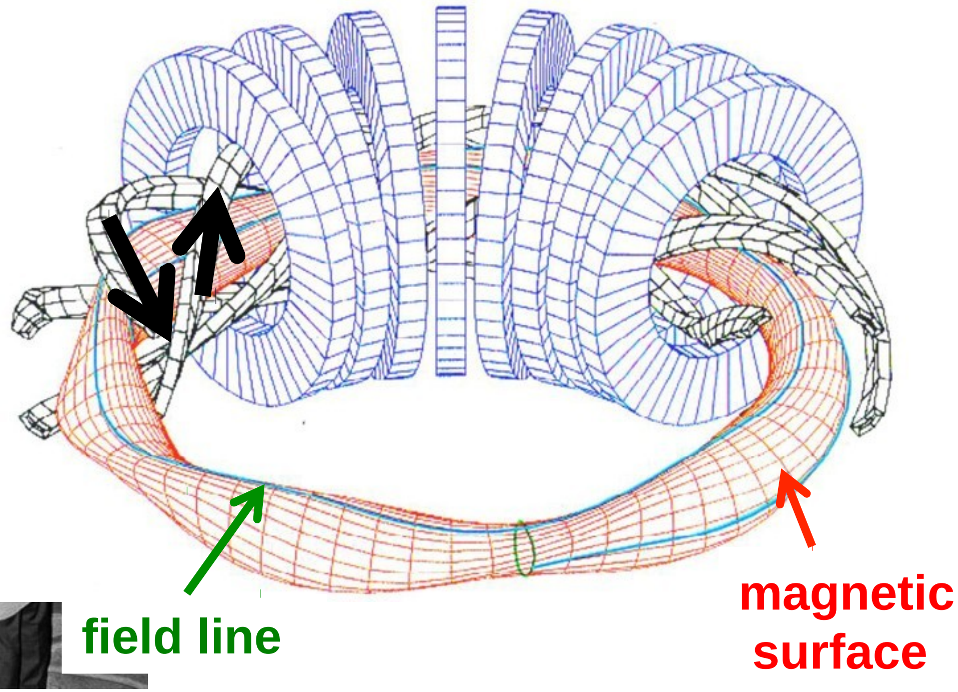
Fig. 2. The helical coil and can assembly is lowered onto the lower support structure on the machine.

The previous Wendelstein 7-A stellarator used both helical and toroidal coils (classic stellarator)

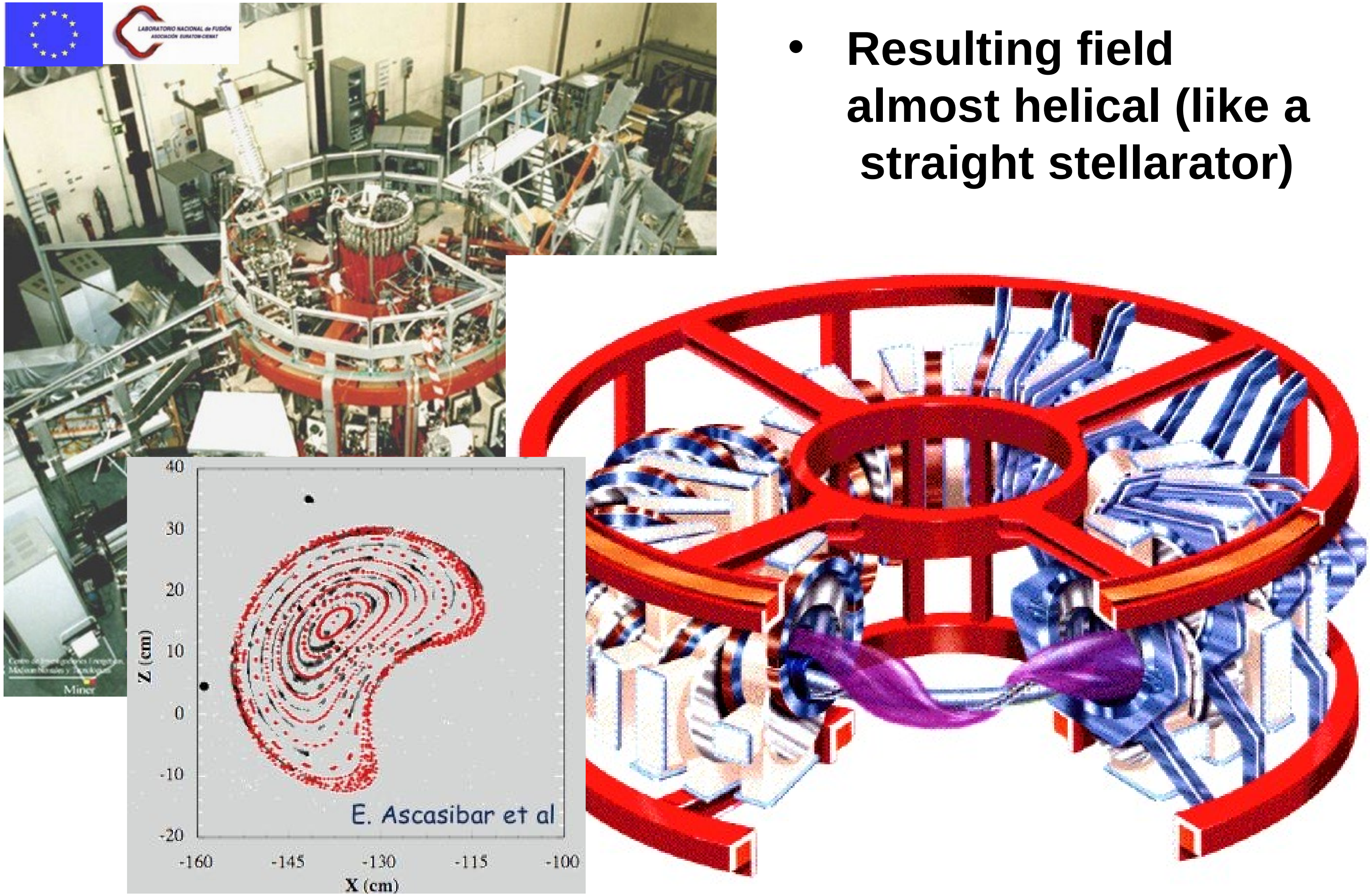
- **Wendelstein 7-A dimensions: R=2 m, a=10cm, l=2, m=5, volume << 1 m^3**

  [Wendelstein family: WEGA, W7-A, W7-AS, W7-X]



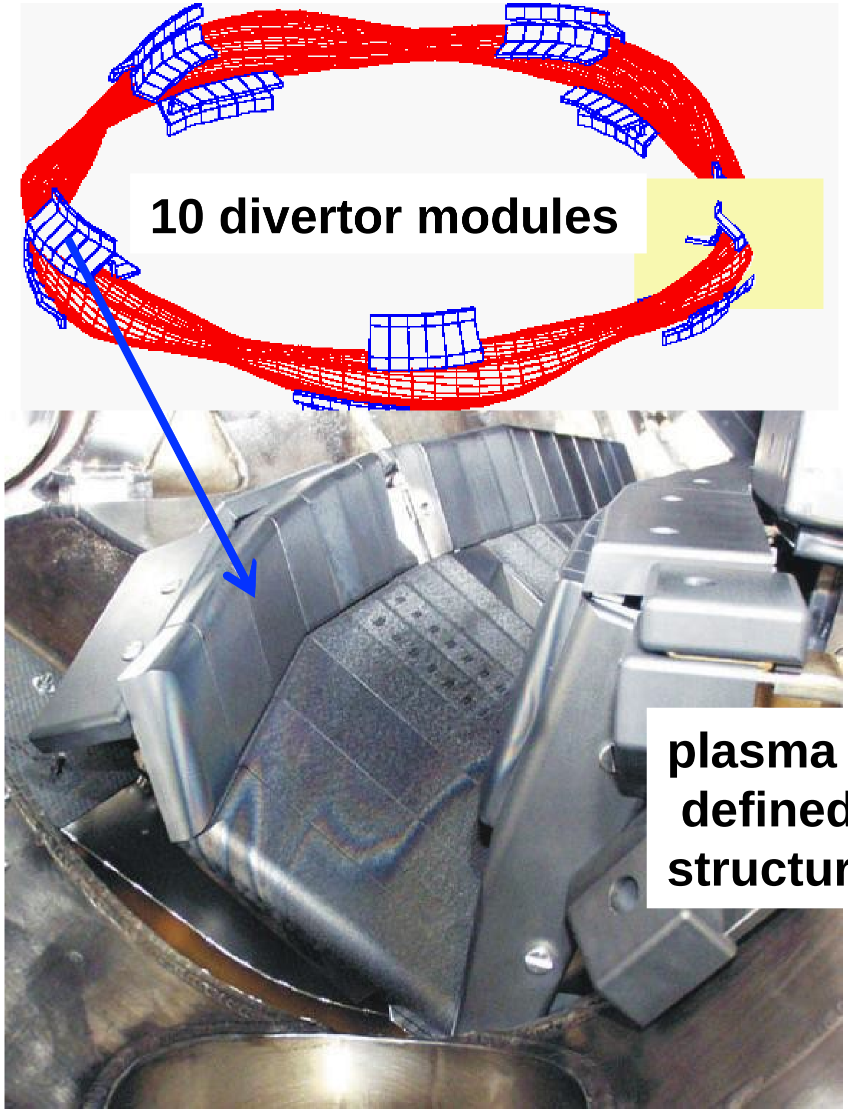


In a heliac (TJ-II, CIEMAT, Spain) the plasma is wound around a single central conductor

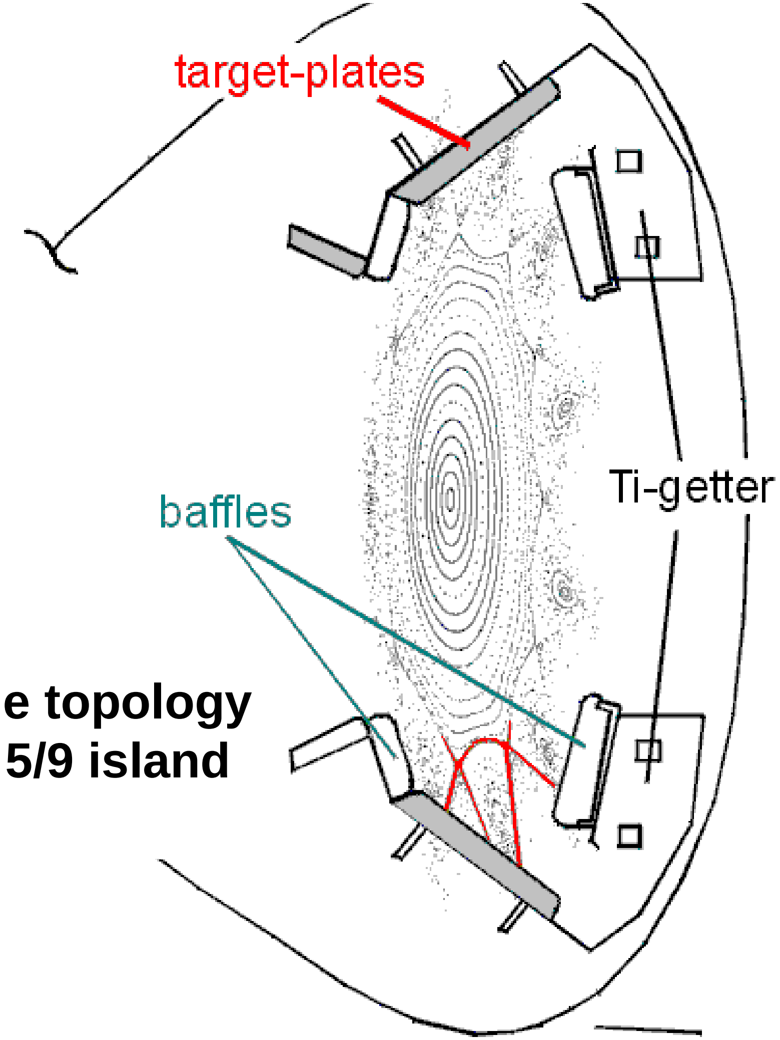


- **Resulting field almost helical (like a straight stellarator)**

Islands in the edge can be used for energy and particle exhaust



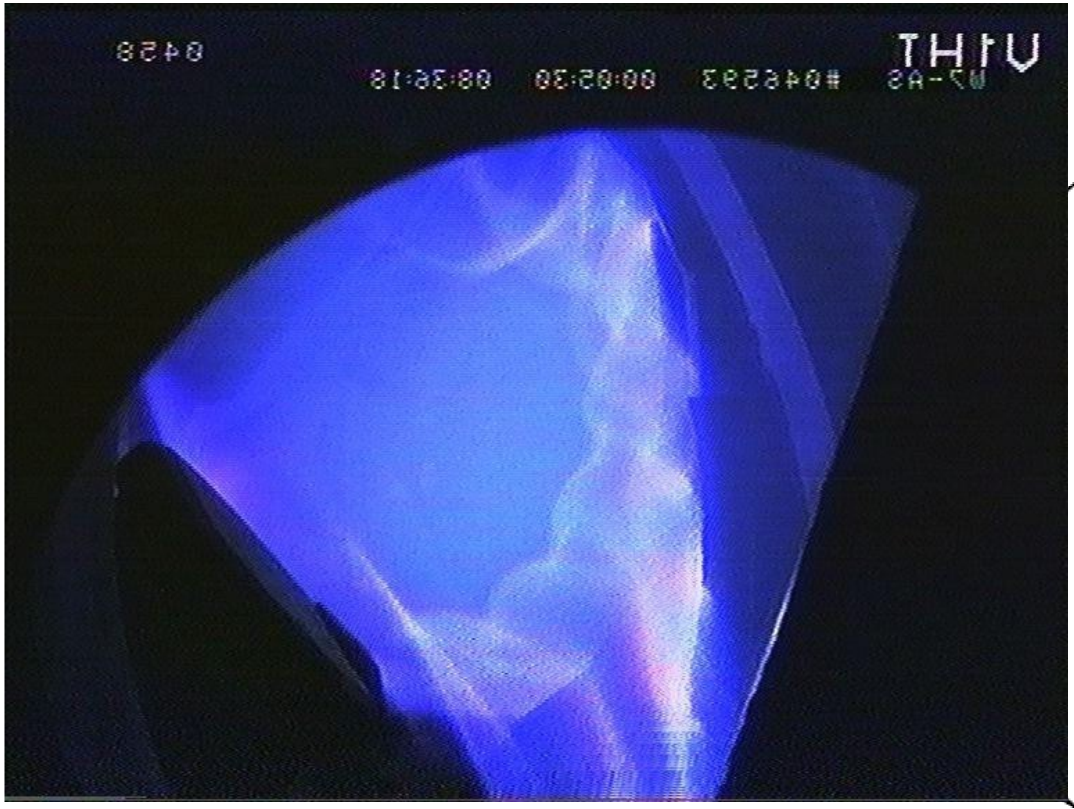
Wendelstein 7-AS

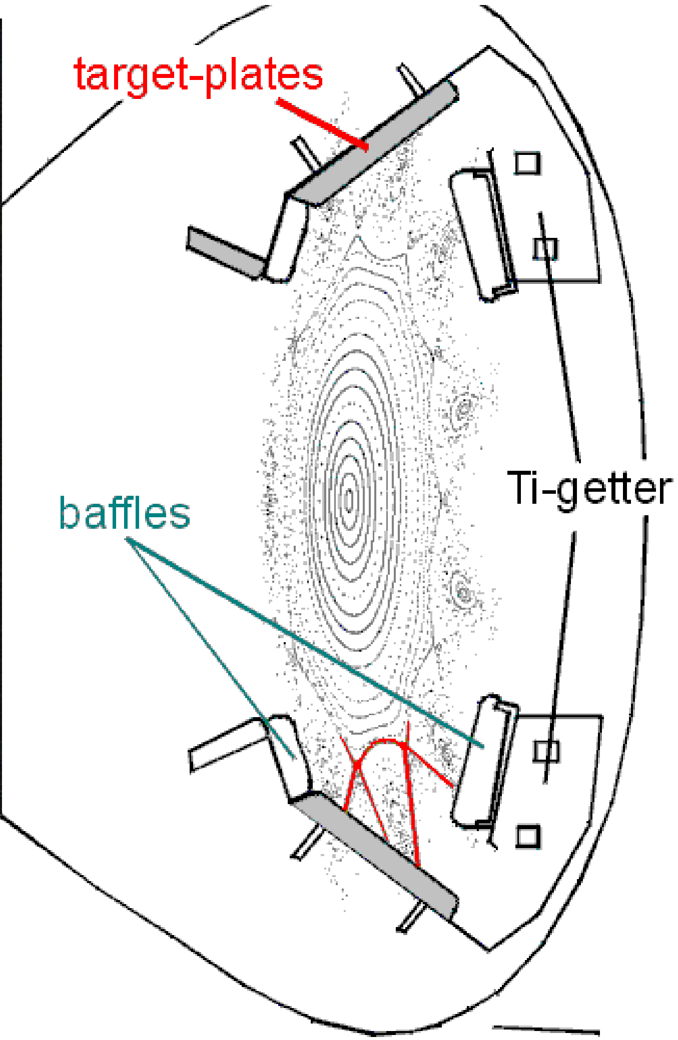


plasma edge topology defined by 5/9 island structures

The island structure was observed with a toroidally viewing camera system

Wendelstein 7-AS

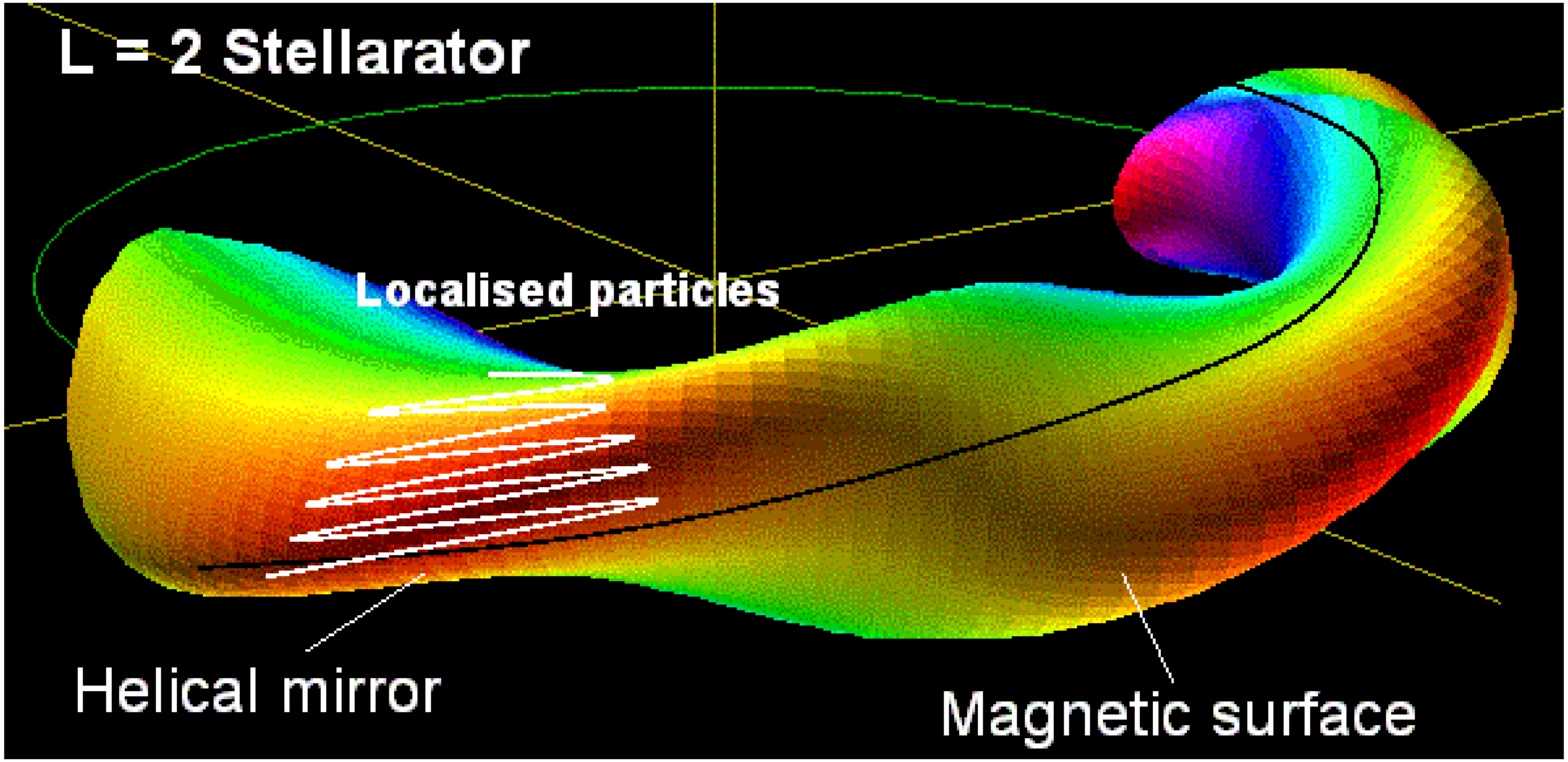




plasma edge topology defined by 5/9 island structures

Transport processes in stellarators

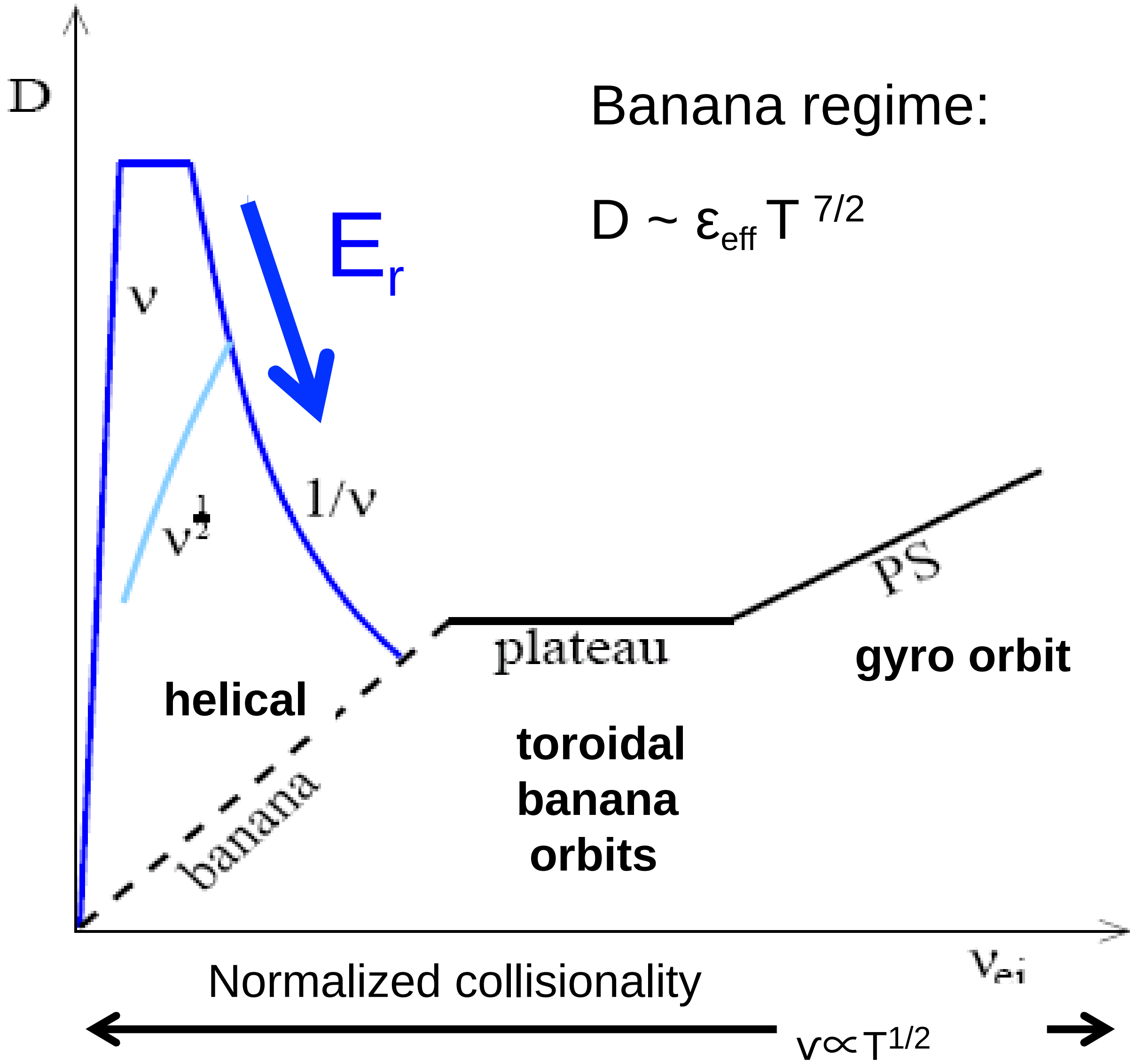
Orbit drifts (in an inhomogeneous magnetic field) leads to losses of particles and energy



- **Stellarators have more classes of trapped particles than tokamaks**

⇒ (Diffusive) neoclassical transport of particles = losses

Stellarators require a strong reduction of radial convective transport to be high-performing



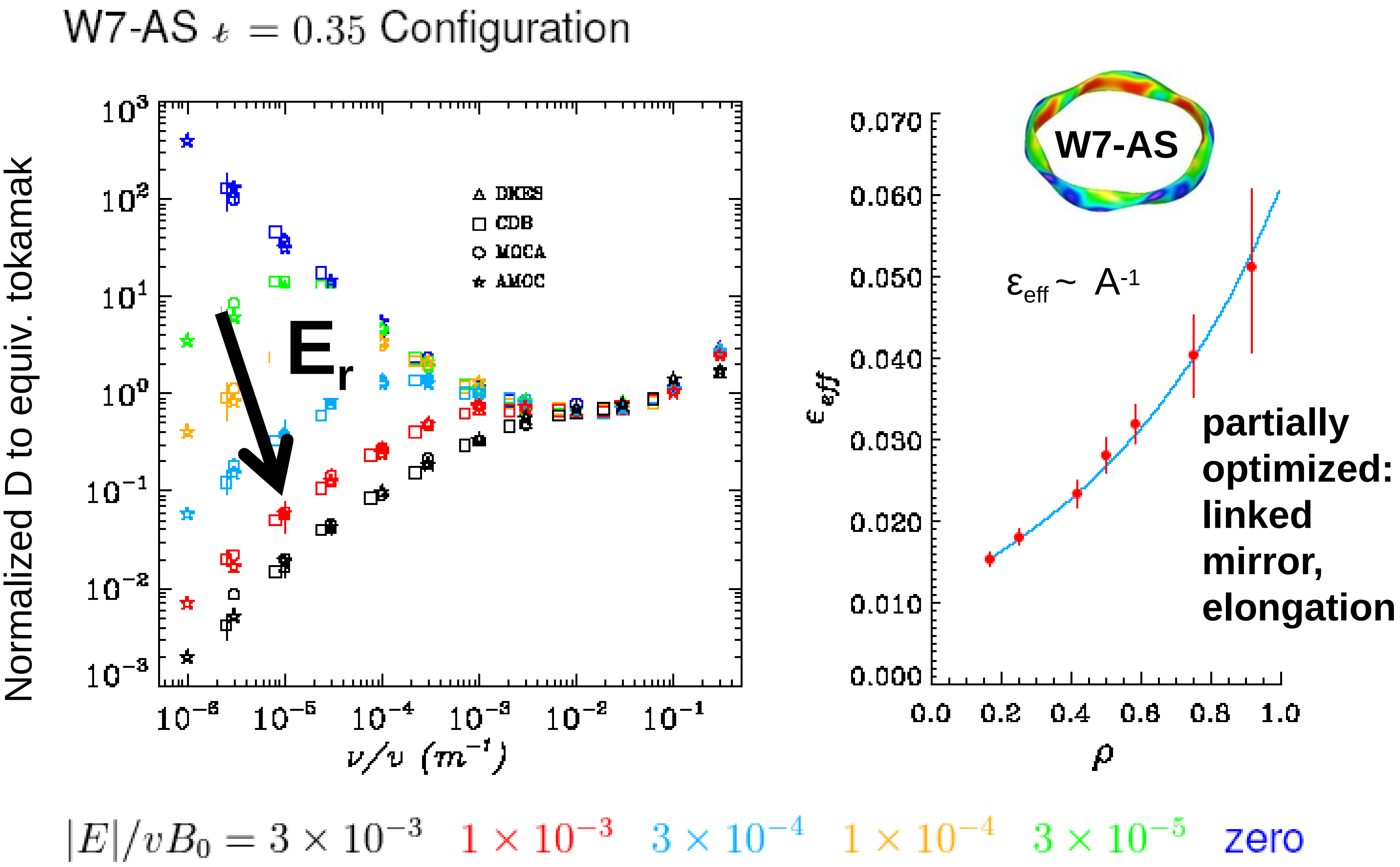
Banana regime:

$D \sim \varepsilon_{eff} T^{7/2}$

- **Diffusion in low collisionality regime is large (ripple trapped particles)**
- **Radial electric field leads to de-trapping of via ExB drifts**

⇒ Optimization of B-field (ε_{eff}) ⇒ linked mirror concept)

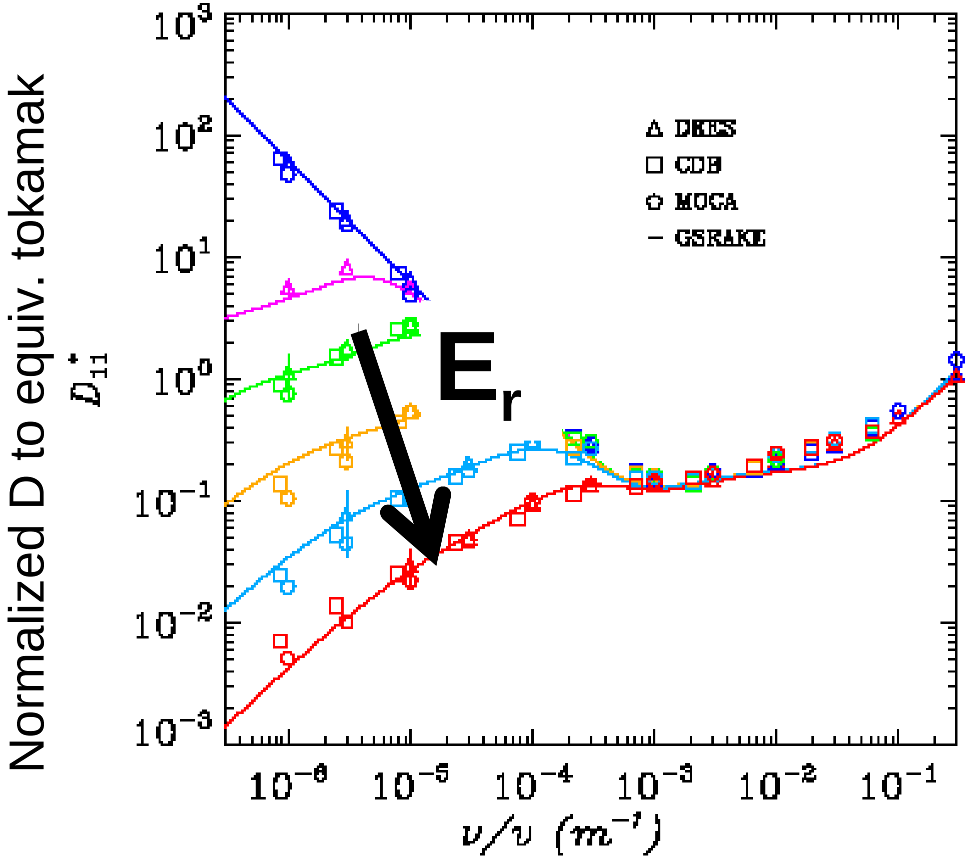
With increasing radial electric field (de-trapping), cross-field transport can be reduced at low collisionality



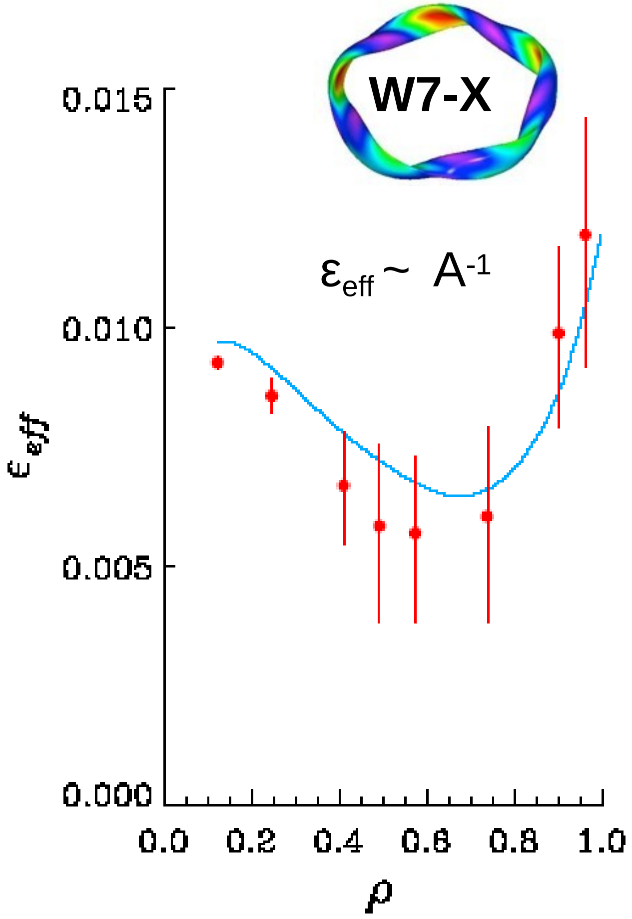
partially optimized: linked mirror, elongation

In a drift-optimized stellarator (Wendelstein 7-X), neoclassical diffusion is significantly reduced

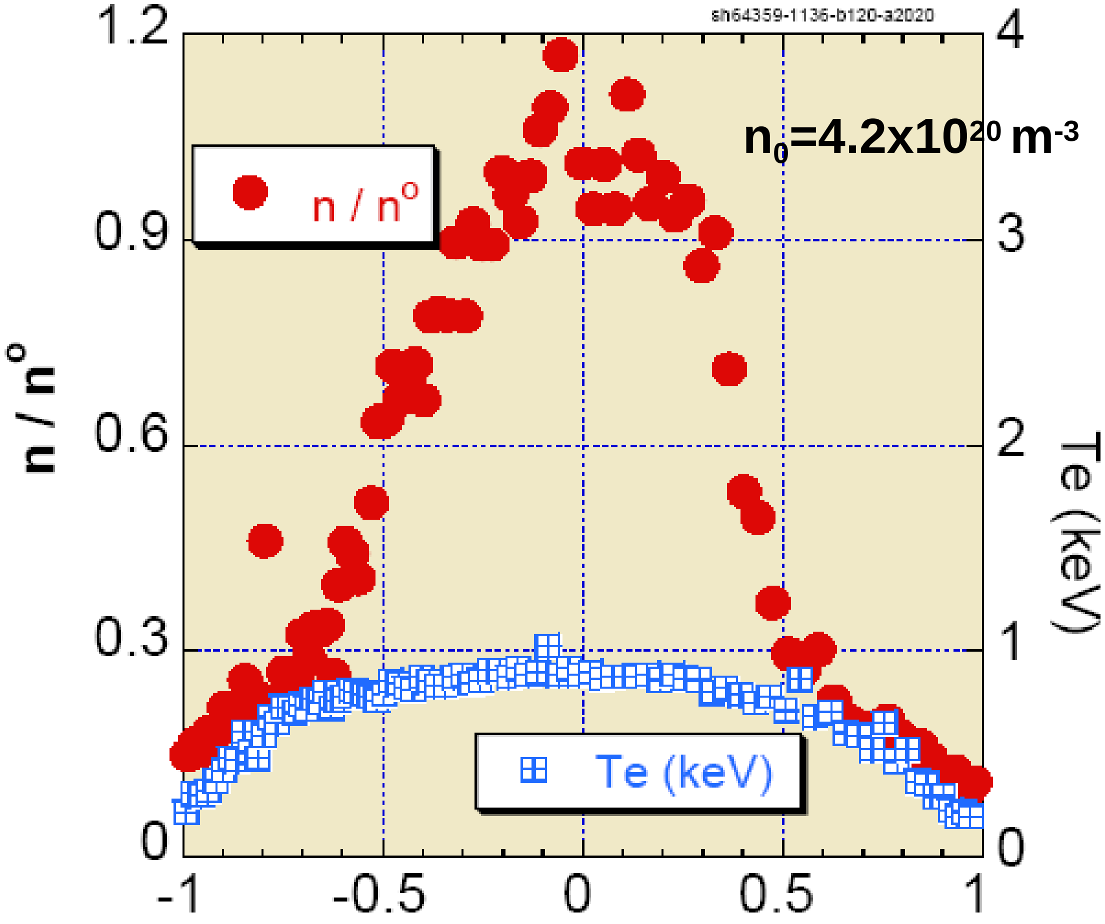
W7-X Standard Configuration



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



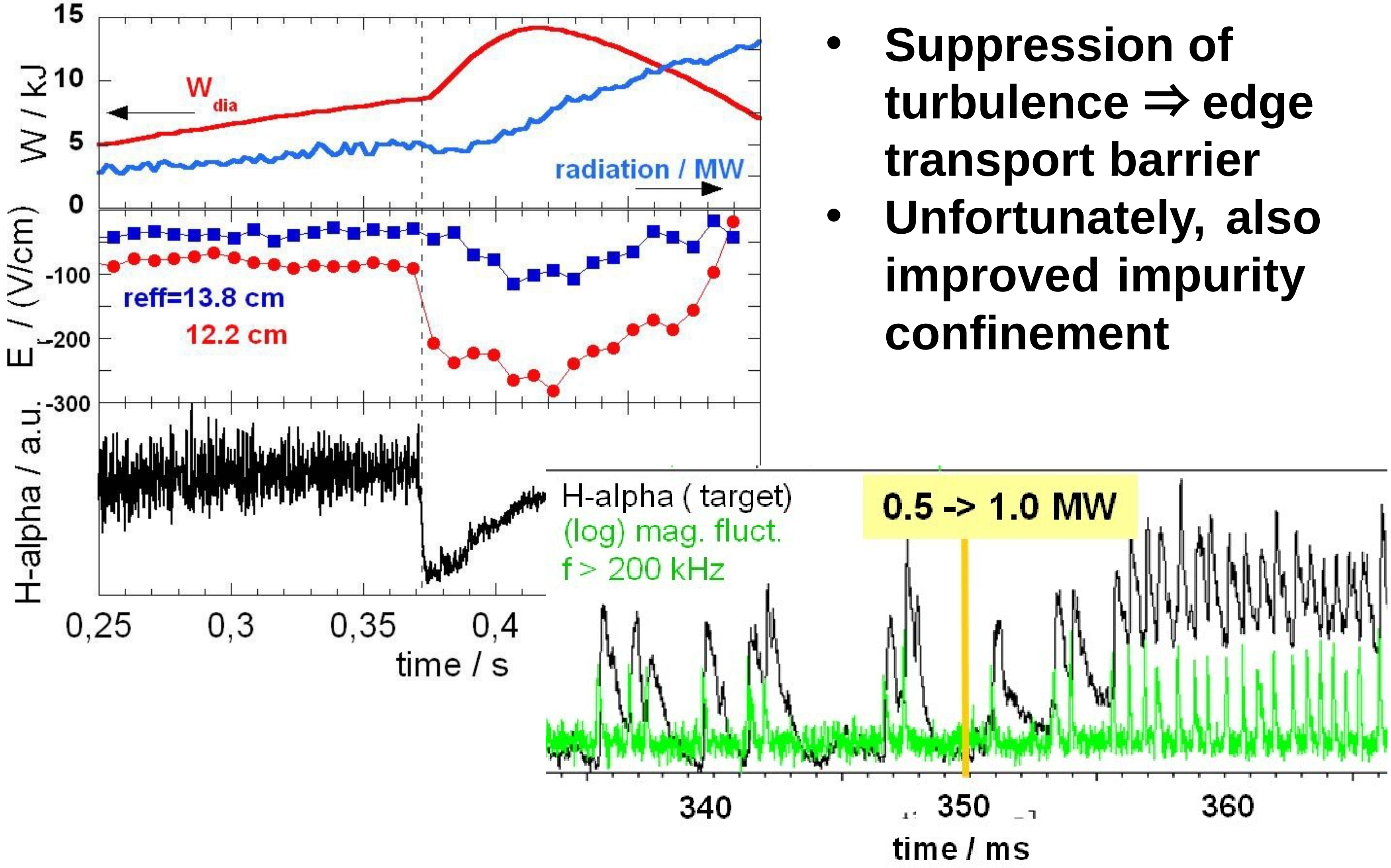
Superdense core plasmas have been obtained in the LHD stellarator



- **High-density operation is preferable also in stellarators:** fusion yield, confinement, low edge temperatures
- **Stellarators have no disruptive density limit**

⇒ Yet, operation still require density and impurity control

H-mode confinement and edge localized modes were also observed in stellarators (W7-AS)



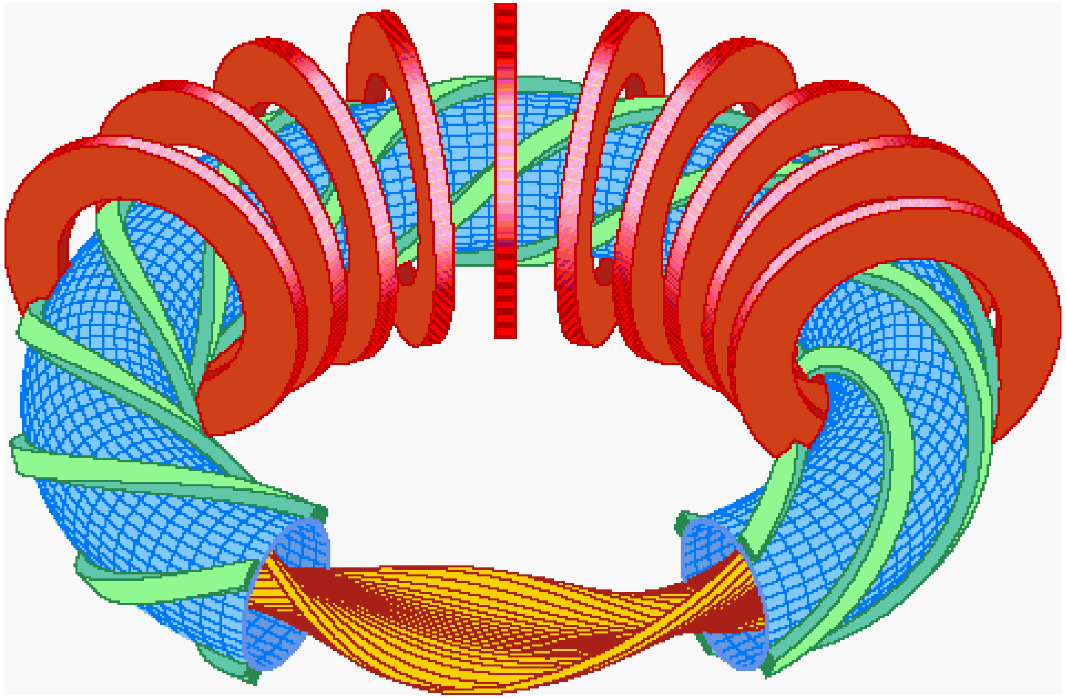
- **Suppression of turbulence ⇒ edge transport barrier**
- **Unfortunately, also improved impurity confinement**

Toward future stellarator reactors

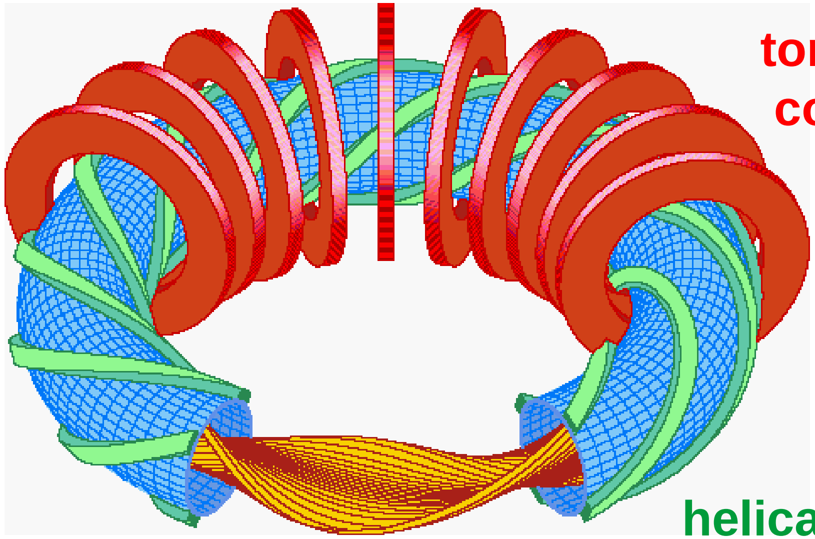
To make stellarators successful, one needs to minimize transport losses

- **Steady-state capability without need for current drive ⇒ no current disruption**
- **Maintain confining field and divertor island structures even at high pressure**
- **High-density operation:** no density limit like in tokamaks
- **Collisional losses:** fast particles, neoclassical transport, turbulence and flows

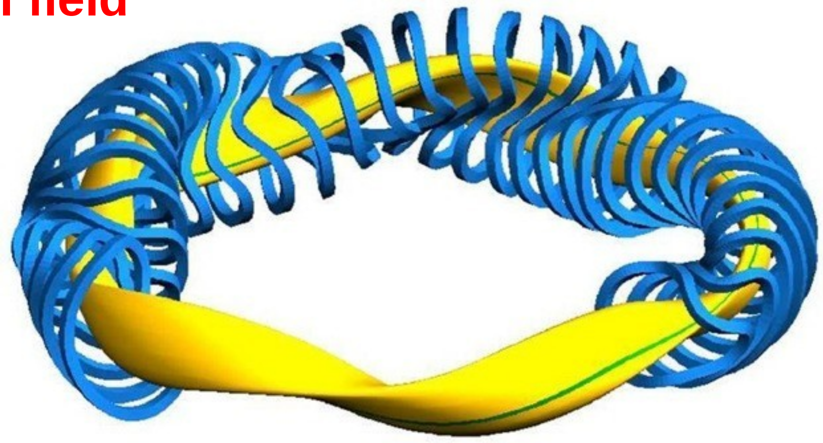
⇒ Option: design an optimized magnetic configuration



Modular coils give wider accessible Fourier distribution of currents, and 3-D shaping of axis

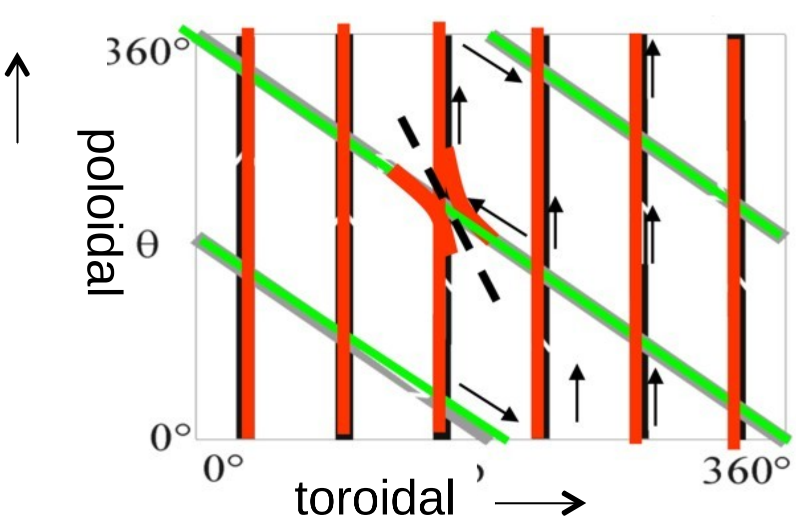


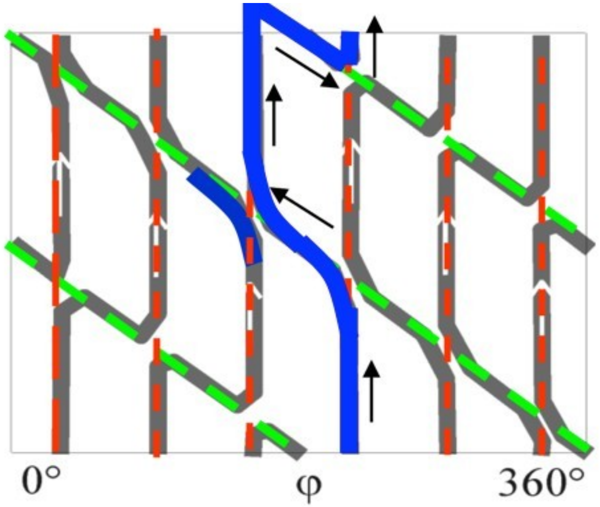
toroidal field coils



helical field coils

modular coils





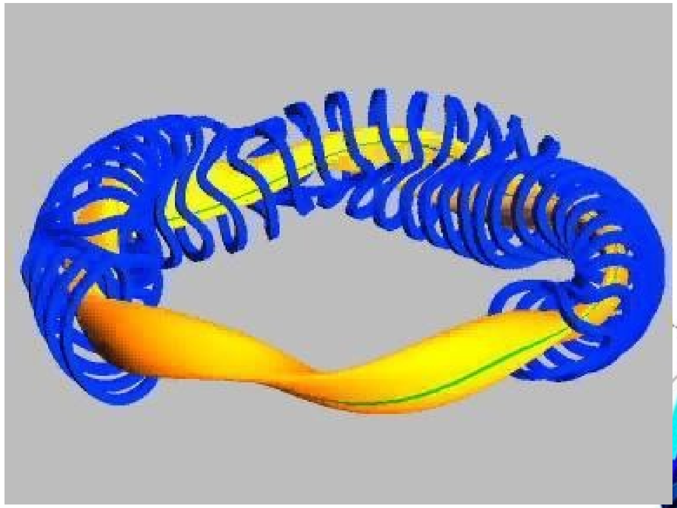
The Wendelstein 7-X is the first optimized superconducting stellarator

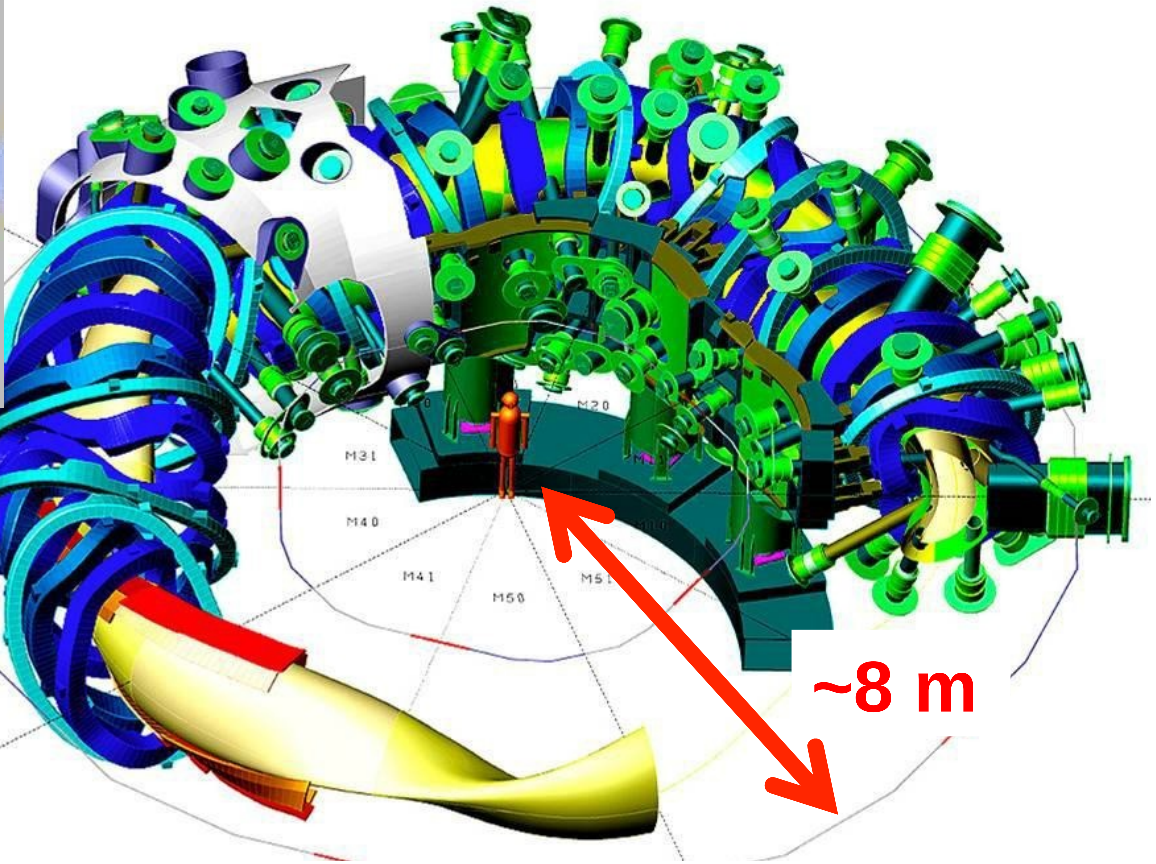
- HELIAS (“pure stellarator”)
  ⇒ drift-optimized

- R=5.5 m, a=0.52 m,

  V_{plasma}~30 m^{-3}

  (vs. JET: 3/1/100 and ITER 6/2/840)





- Fully cooled in-vessel components and island divertor

The Wendelstein 7-X is the first optimized superconducting stellarator

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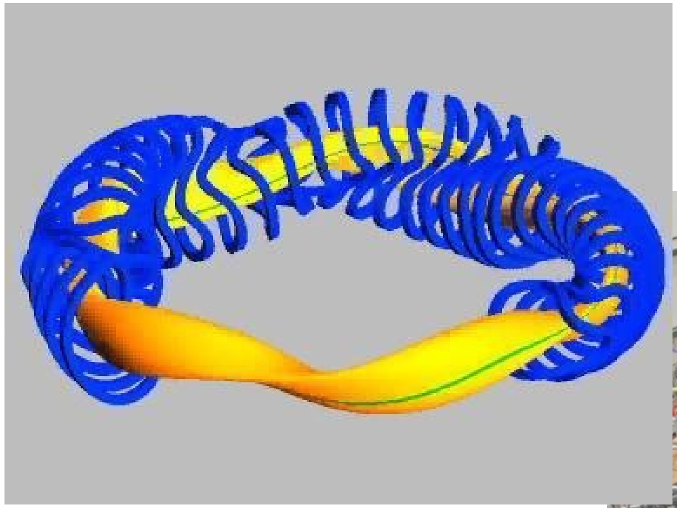
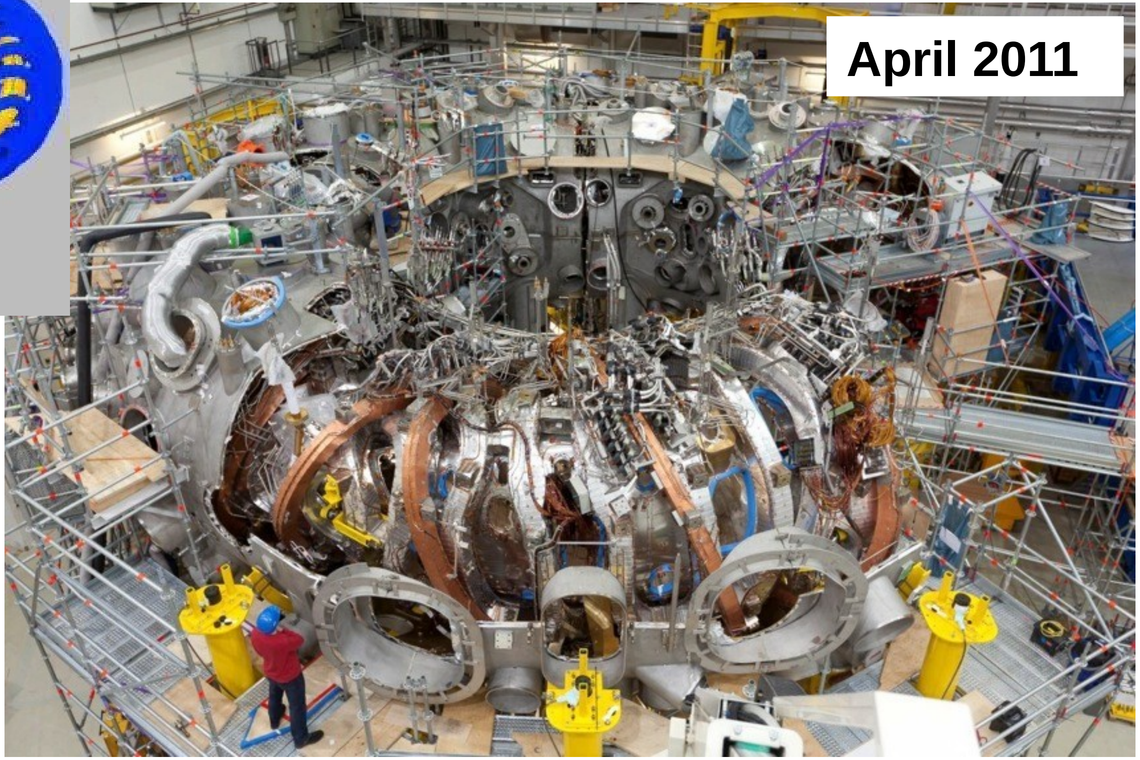
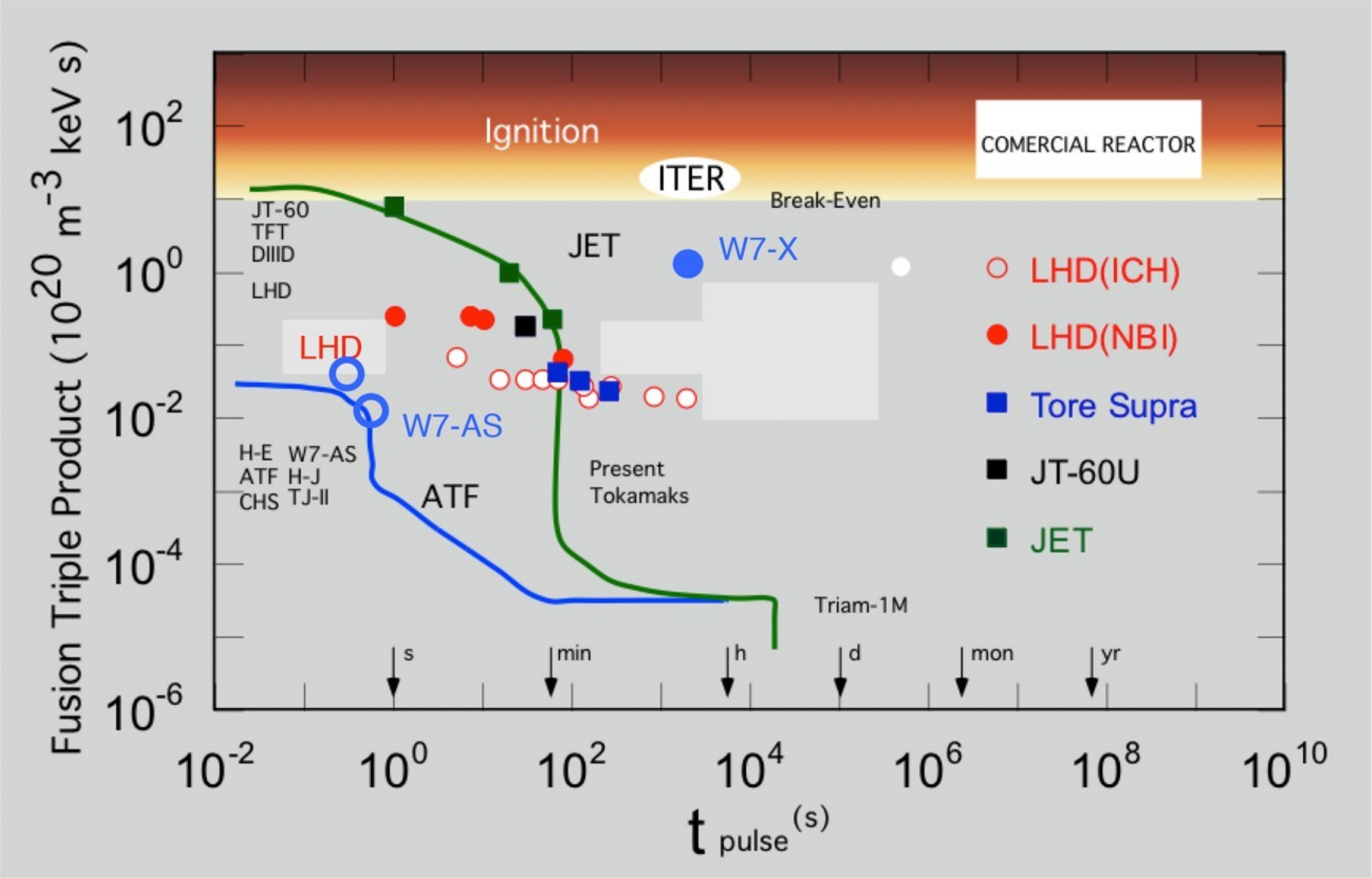




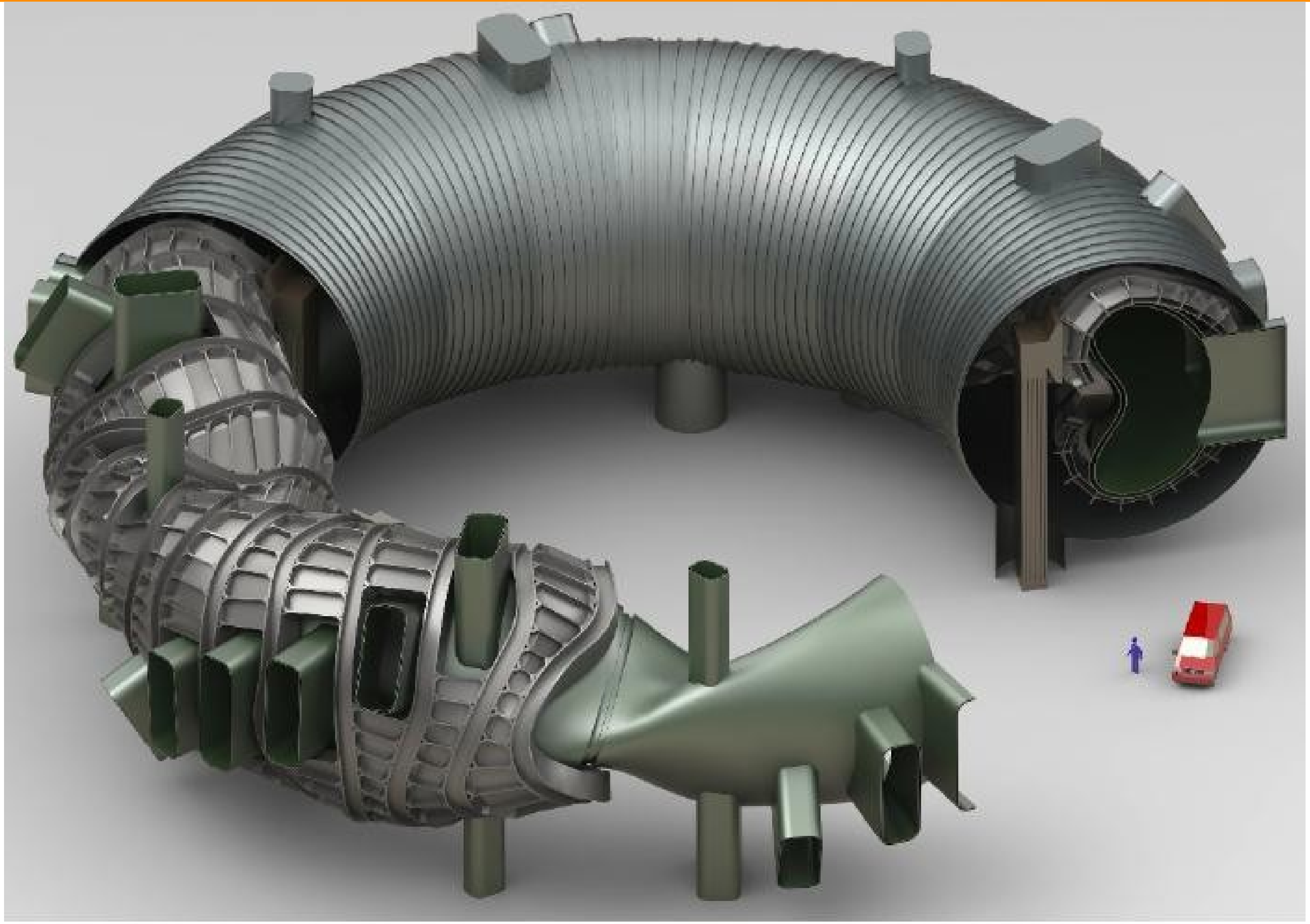
April 2011

- Fully cooled in-vessel components and island divertor

The projected performance (D-T equivalent) of W7-X is an order of magnitude lower than that of ITER

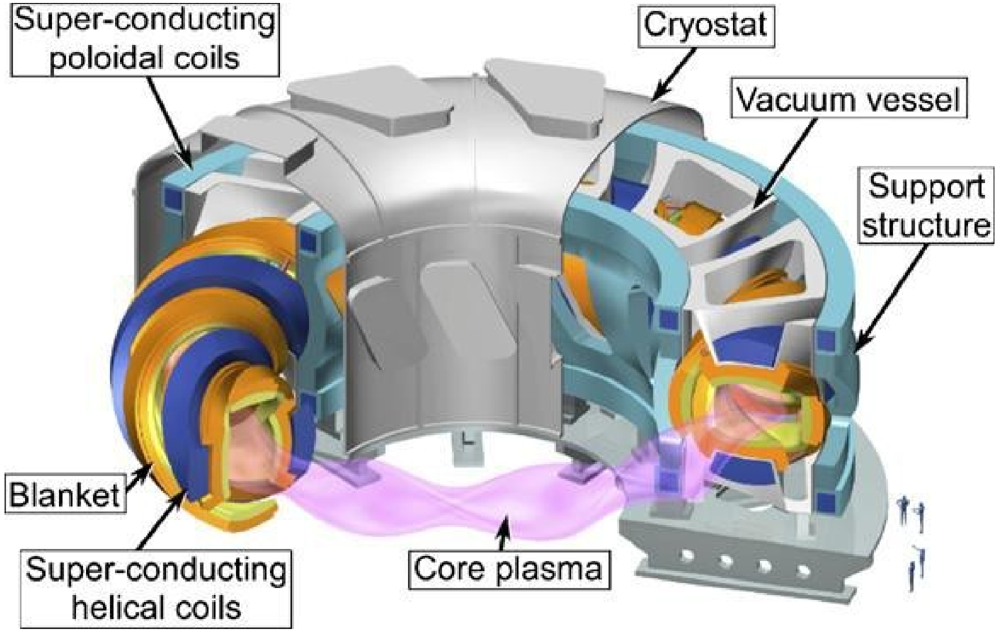


Conceptually, scientists have already been planning for future stellarator reactors

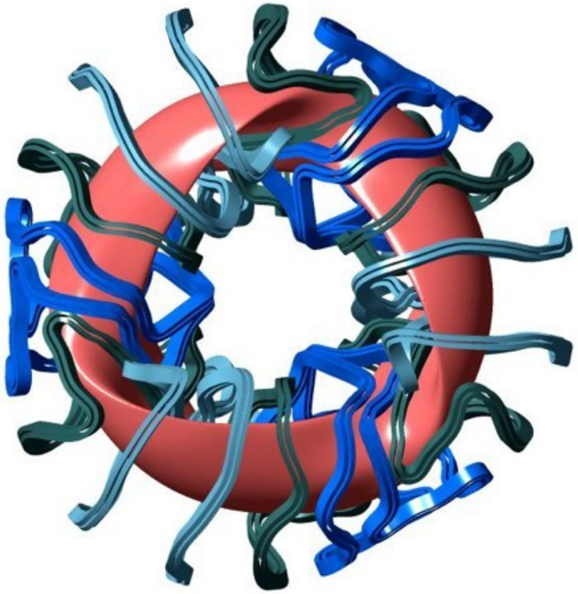


Various extensions of helical devices toward reactors exist

FFHR: R=20 m, P_{th}=3 GW

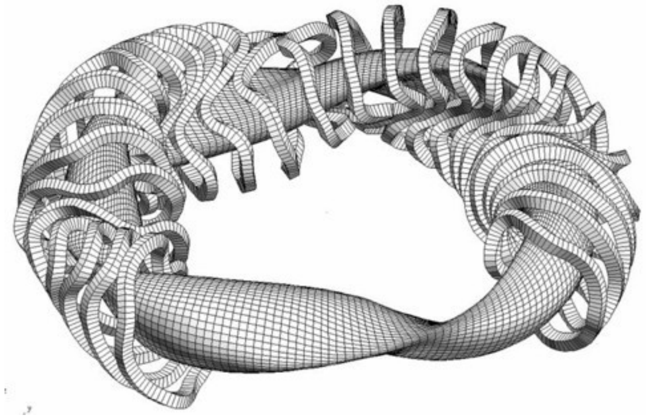


ARIES-CS: R=8 m, P_{th}=4 GW



- FFHR = force free helical reactor (heliotron), based on LHD [Fus. Eng. Design 1995]
- HSR4/18: Helias reactor with four field periods, based on W7-X [Nucl Fusion 2001]
- ARIES-CS: compact stellarator [Fus. Sci and Tech. 2008]

HSR4: R=18 m, P_{th}=3 GW

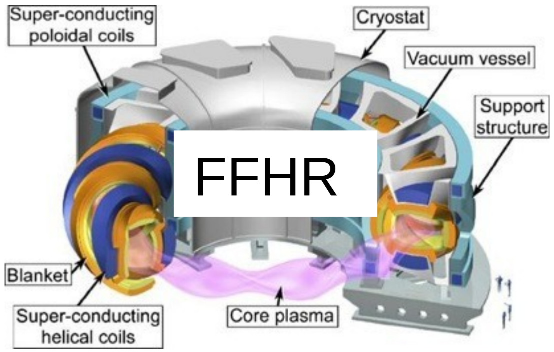


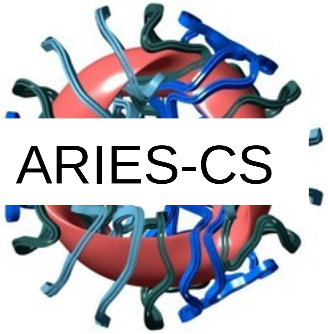
Stellarator specific reactor issues

+ **Steady-state ⇒ reduced fatigue effects**
+ **No current drive ⇒ low recirculating power (CD, SC, pulse length, beta → net electricity)**

- **Mechanical forces between coils requiring heavy support structure**
- **Limited space between plasma edge and coil in certain locations for blanket and shielding**

⇒ Going to larger R usually helps


FFHR


ARIES-CS


HSR4

Is a stellarator reactor better than a tokamak reactor? In other words, who’s winning the race?

- **Costs are significant – why paying twice?**

⇒ Total investment into W7-X (1997-2014) = 1.1 bn €, EU for ITER until 2022 = 8 bn €

⇒ EU fusion strategy for W7-X is not considered relevant for ITER, but for DEMO

- **Will there be more than one DEMO?**

⇒ ITER + JT60-SA + (Chinese study) are steps toward DEMO tokamak

⇒ Korea, Japan and China have built superconducting tokamaks

Is a stellarator reactor better than a tokamak reactor? In other words, who’s winning the race?

⇒ US government stopped National Compact Stellarator Experiment (NCSX), a quasi toroidal LN2 cooled device, but also terminated Alcator C-mod ⇒ focus on ITER

⇒ In Japan, there is not yet a decision on a follow-up device to LHD

- **Can we gain from the synergy between tokamaks and stellarators?**

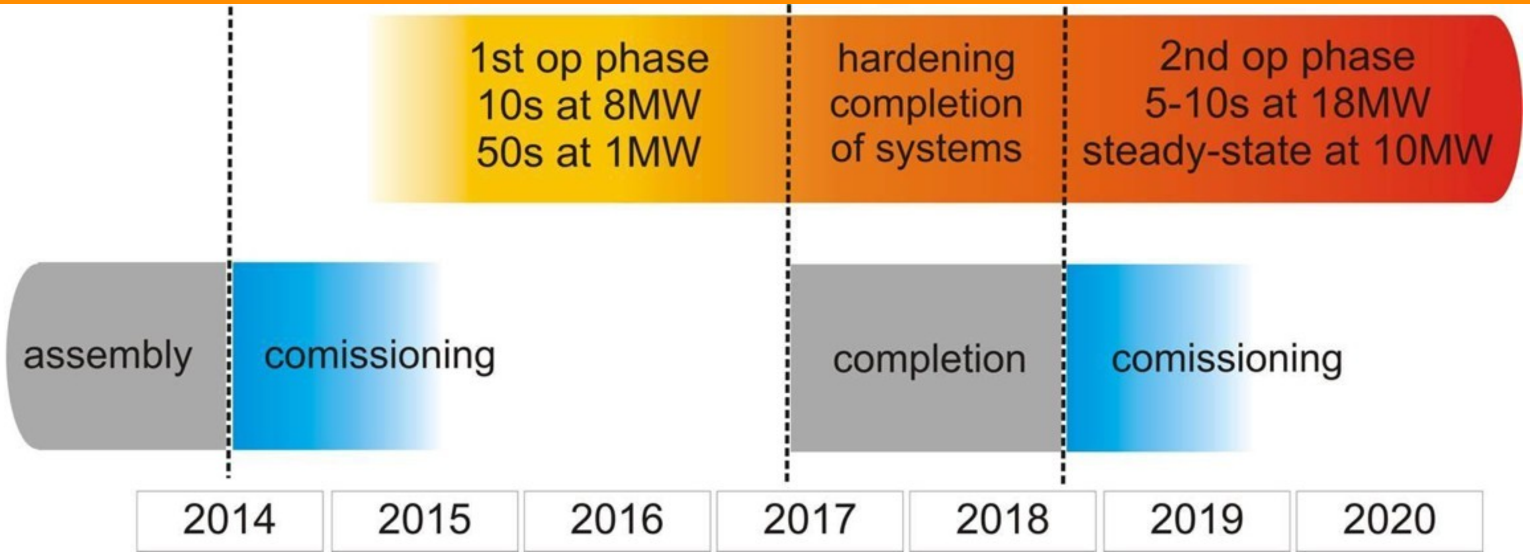
⇒ Tokamak research is better organized, focus on ITER

⇒ Stellarator research need more devices to cover the many concepts

⇒ **Will failure of ITER make way for stellarator?**

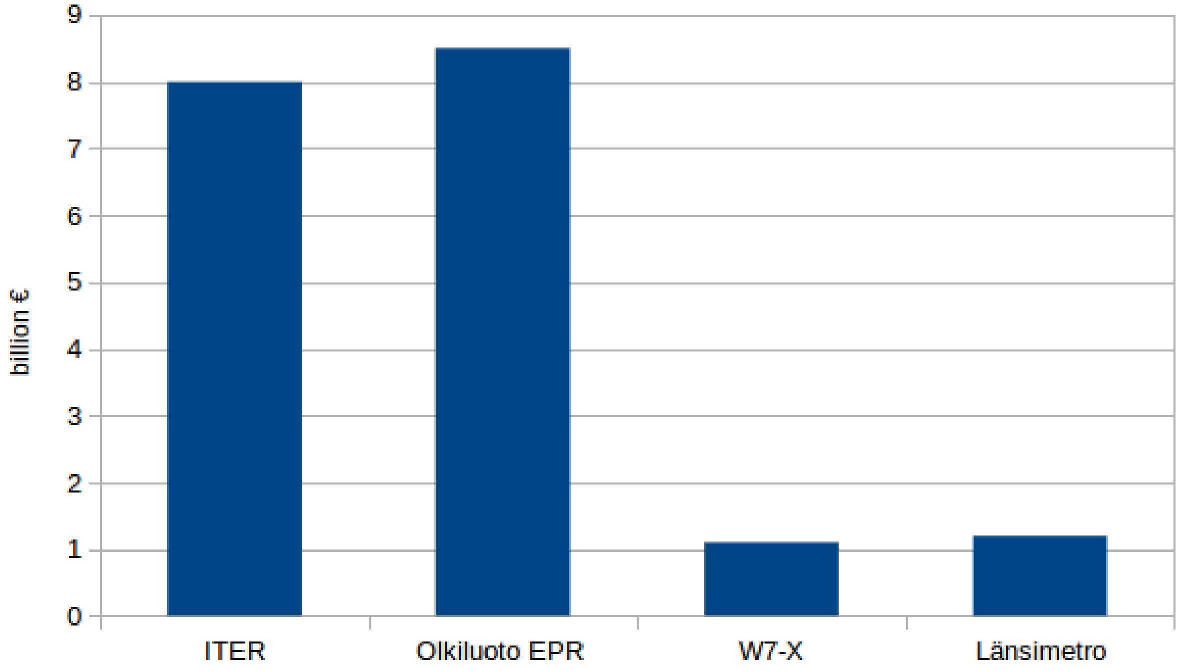
Cost of fusion power plant and electricity

W7-X staff and funding profile



- W7-X team ~380 people, not including visitors and support personnel
- Total investment between 1997-2014 ~1.1 bn € (370 m€
  device, 100 m€ buildings, 310 m€ staff)
- 25% funding from EU, 75% German and regional government

Project costs: ITER and W7-X vs. Olkiluoto and Länsimetro

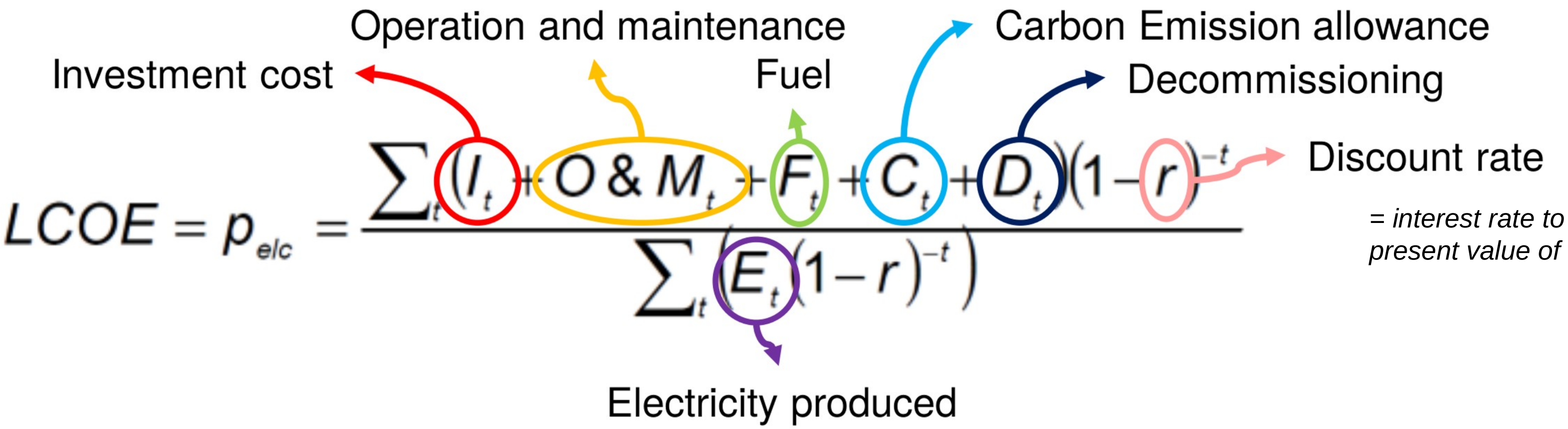


a) EU for ITER until 2022 = 8 bn € (or: total constraction costs 20 bn$ compared to original estimate 5 bn$ and full power 2027 compared to original estimate 2016)
b) Finland: Olkiluoto EPR fission power plant, “first of a kind”: 8.5 billion €, starts 2020 (compare to original estimate 3.2 billion €, starts 2009)
c) Total investment into W7-X (1997-2014) = 1.1 bn€ (0.37 bn€ device, 0.1 bn€ buildings, 0.31 bn€ staff; started 2015, not e.g. 2004)
d) Finland: Länsimetro underground (via Otaniemi): first phase costed 1.2 bn€ (2008 accepted budget 0.7 bn€)

Cost of fusion electricity depends on...

- Investment cost depends on machine size expecially for large reactors
- for <r> ≈ plasma coil spacing further reduce of size does not help much (for a given P_{output})
  - a) higher loads on components
  - b) tighter spaces for maintenance
  - c) other engineering constraints
    - → a larger extrapolation from current technology required
- Cost of electricity also depends on the availability of power plant (→ replacement of components), learning factor, cost of materials and technological development

Cost of fusion electricity depends on...



$$LCOE = p_{elc} = \frac{\sum_t \left(I_t + O\&M_t + F_t + C_t + D_t\right)(1-r)^{-t}}{\sum_t \left(E_t (1-r)^{-t}\right)}$$

= interest rate to determine the present value of future cash flows

In fusion ~ 70 % cost of capital, 3% O&M, 25% blanket and divertor replacement, ~ 1% Fuel, < 1% Decommissioning

Bustreo, ETSAP meeting 2013

Example: ARIES-CS Power-Plant Investment Cost

	Account No.	Account Title	Million dollars
	20	Land & land rights	12.9
	21	Structures & site facilities	336.1
	22	Reactor plant equipment	1,538.9
	22. 1. 1.	FW/blanket/reflector	59.4
	22. 1. 2.	Shield	**228.6**
	22. 1. 3.	Magnets	222.9(b)
	22. 1. 4.	Supplemental-heating/CD systems	66.4
	22. 1. 5.	Primary structure & support	73.1
	22. 1. 6.	Reactor vacuum systems(unless integral elsewhere)	137.1
	22. 1. 7.	Power supply, switching & energy storage	70.6
	22. 1. 8.	Impurity control	6.6
	22. 1.	Reactor equipment	864.7
	22. 2.	Main heat transfer & transport systems	474.8
	23	Turbine plant equipment	**314.6**
	24	Electric plant equipment	**138.8**
	25	Miscellaneous plant equipment	**71.0**
	26	Heat rejection system	56.1
	27	Special materials	151.3
	90	Direct cost (not including contingency)	2,620.0
		Total cost of electricity, COE (c/kWh)	7.76(c)
		No cost penalty has been assumed for manufacturing of complex components. For example, applying a 25% cost penalty to major components (blanket, shield, and coils) increases the COE by 0.37 c/kWh.	
		Assumes coil support structure is fabricated by advanced manufacturing techniques.	
		Assumes an 85% availability (similar to ARIES-AT).	

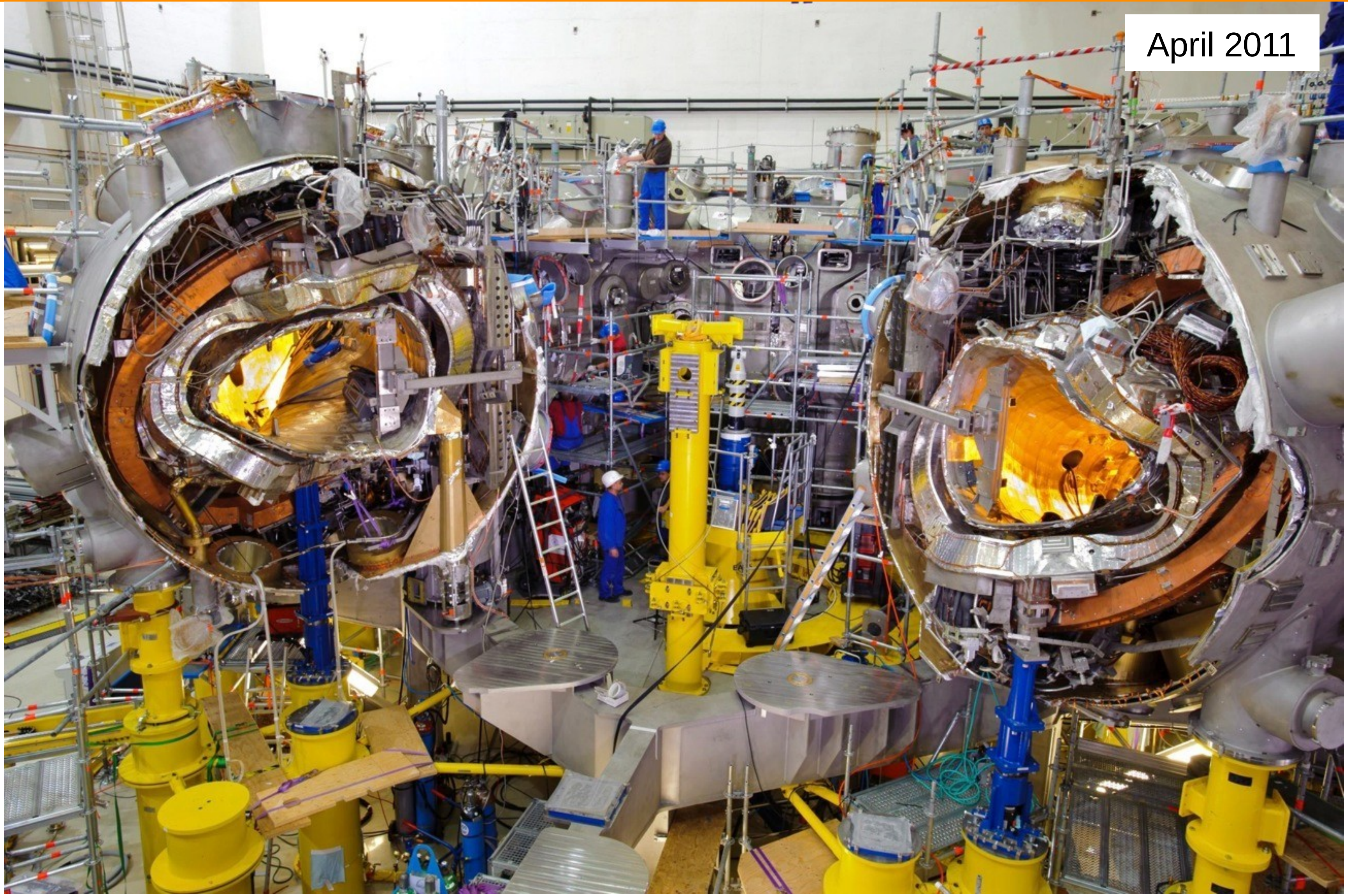
Najmapadi et al, Fusion Science and Technology / Volume 54 / Number 3 / October 2008 / Pages 655-672

The Wendelstein 7-X project

The Wendelstein 7-X project at the Institute for Plasma Physics in Greifswald, Germany

Video: Construction W7-X (1.21 s)

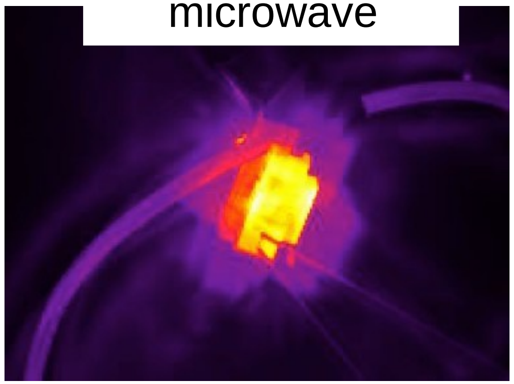
The Wendelstein 7-X project at the Institute for Plasma Physics in Greifswald, Germany



April 2011

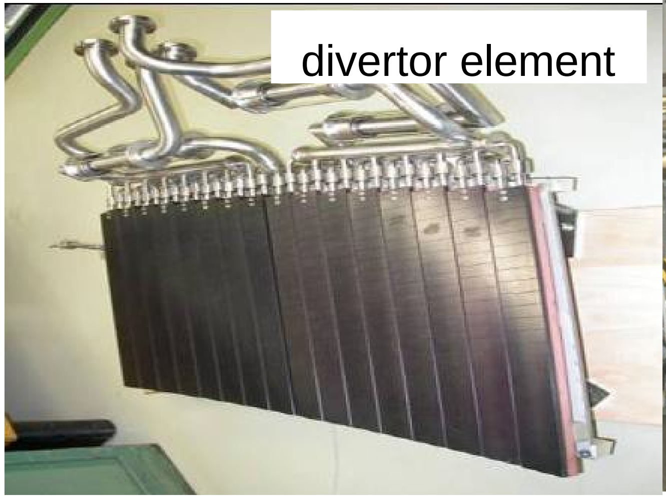
Long-pulse operation requires actively cooled wall elements in the divertor

Mirnov magnetic coil exposed to microwave



microwave launcher

Highly complex device



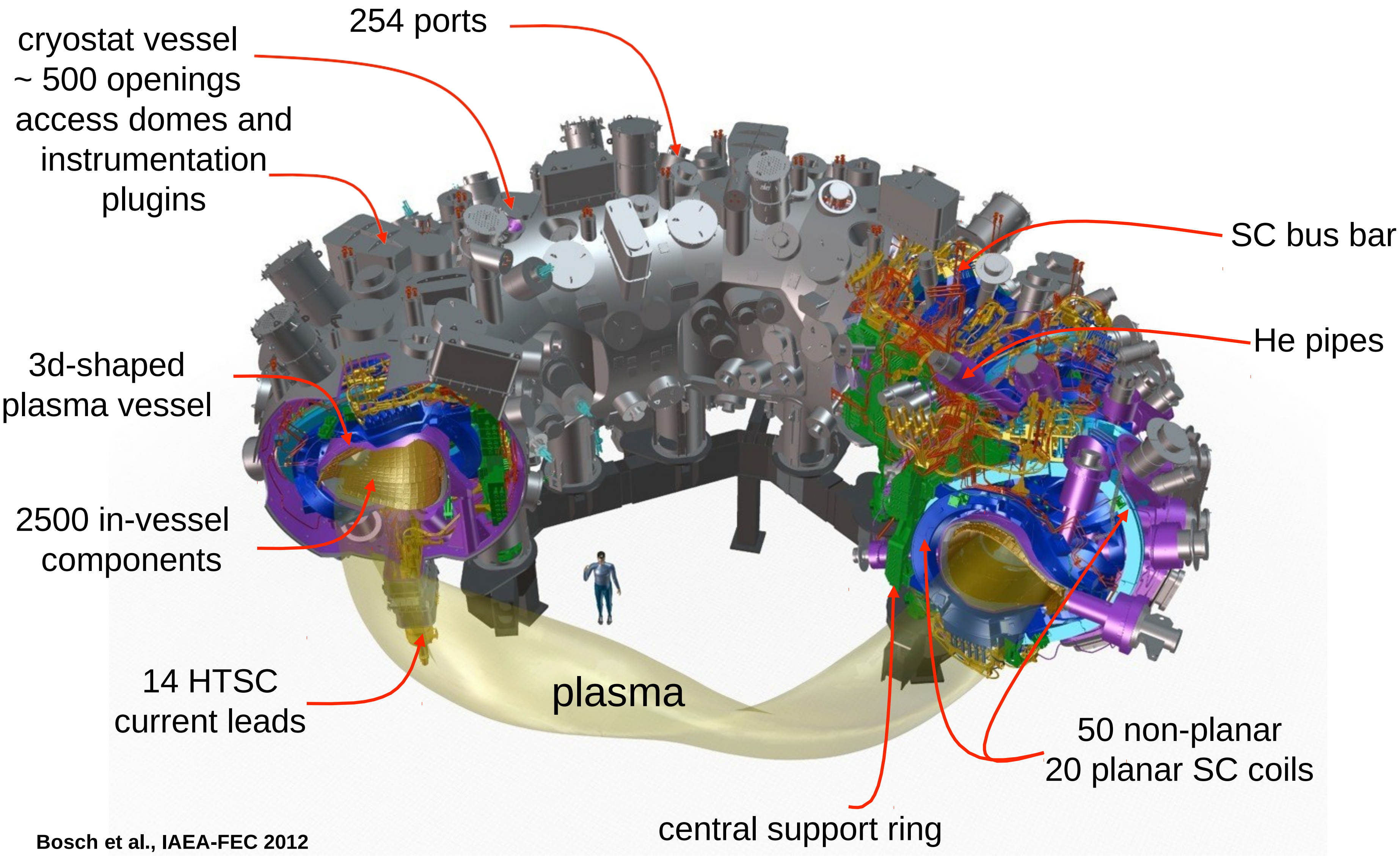
divertor element

1 MW gyrotron

The vacuum vessel follows the twist the desired plasma

Design, fabrication and testing of modular superconducting coils was a major challenge

Integration of the coil / vessel system into a cryostat is a significant engineering challenge



254 ports

cryostat vessel
~ 500 openings

access domes and
instrumentation
plugins

SC bus bar

He pipes

3d-shaped
plasma vessel

2500 in-vessel
components

14 HTSC
current leads

plasma

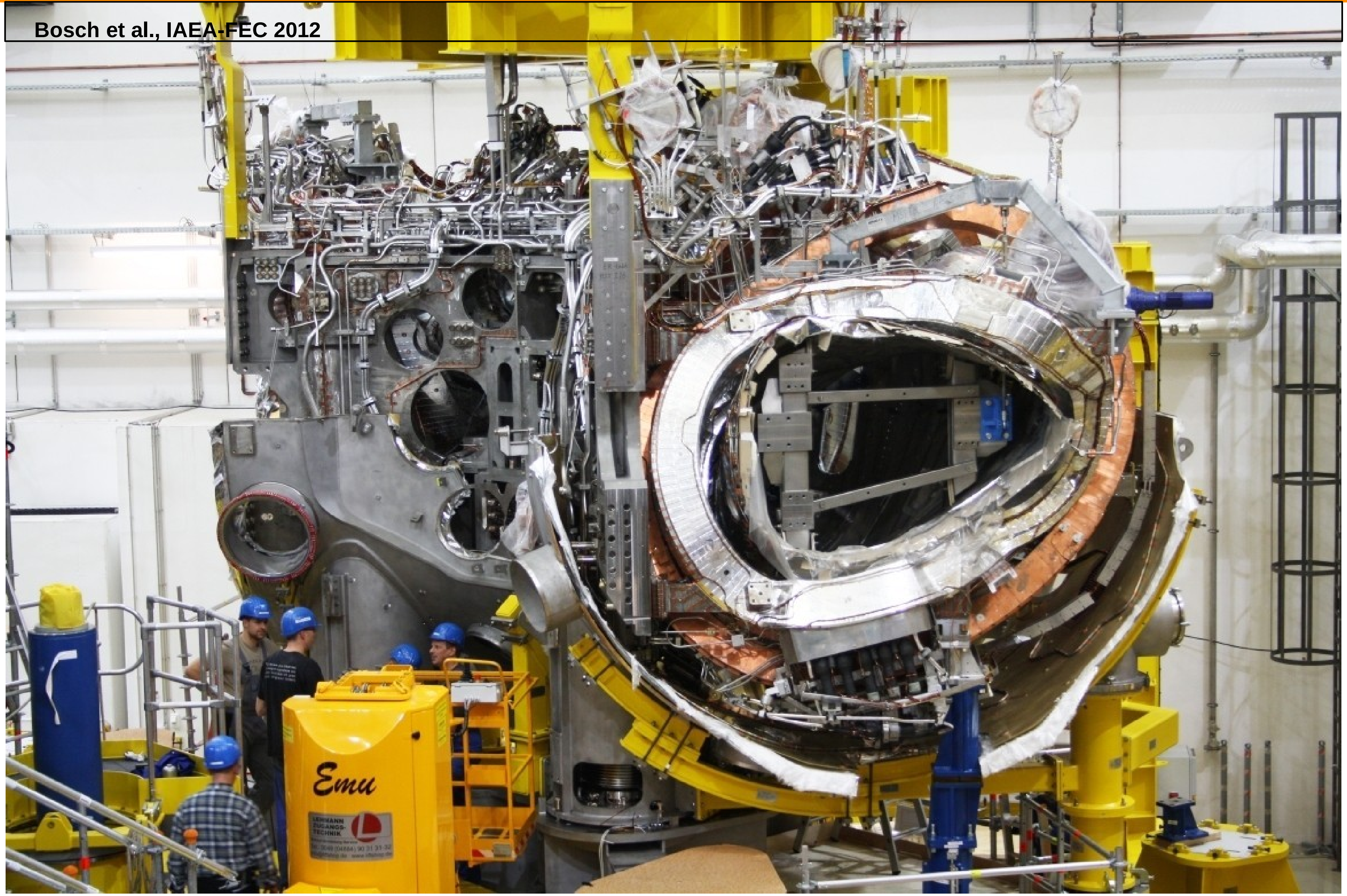
50 non-planar
20 planar SC coils

central support ring

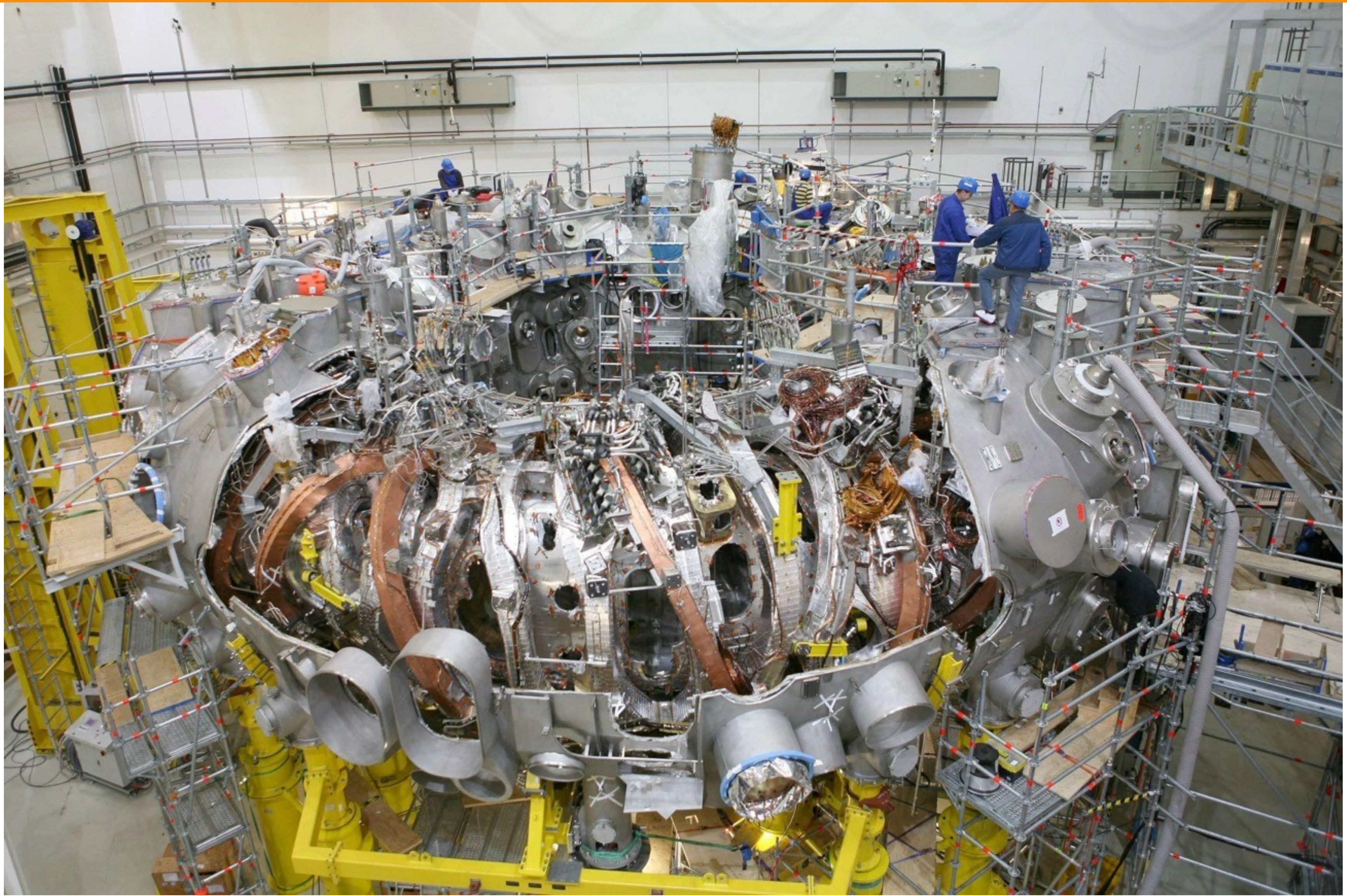
Bosch et al., IAEA-FEC 2012

The Wendelstein 7-X hall in 2006

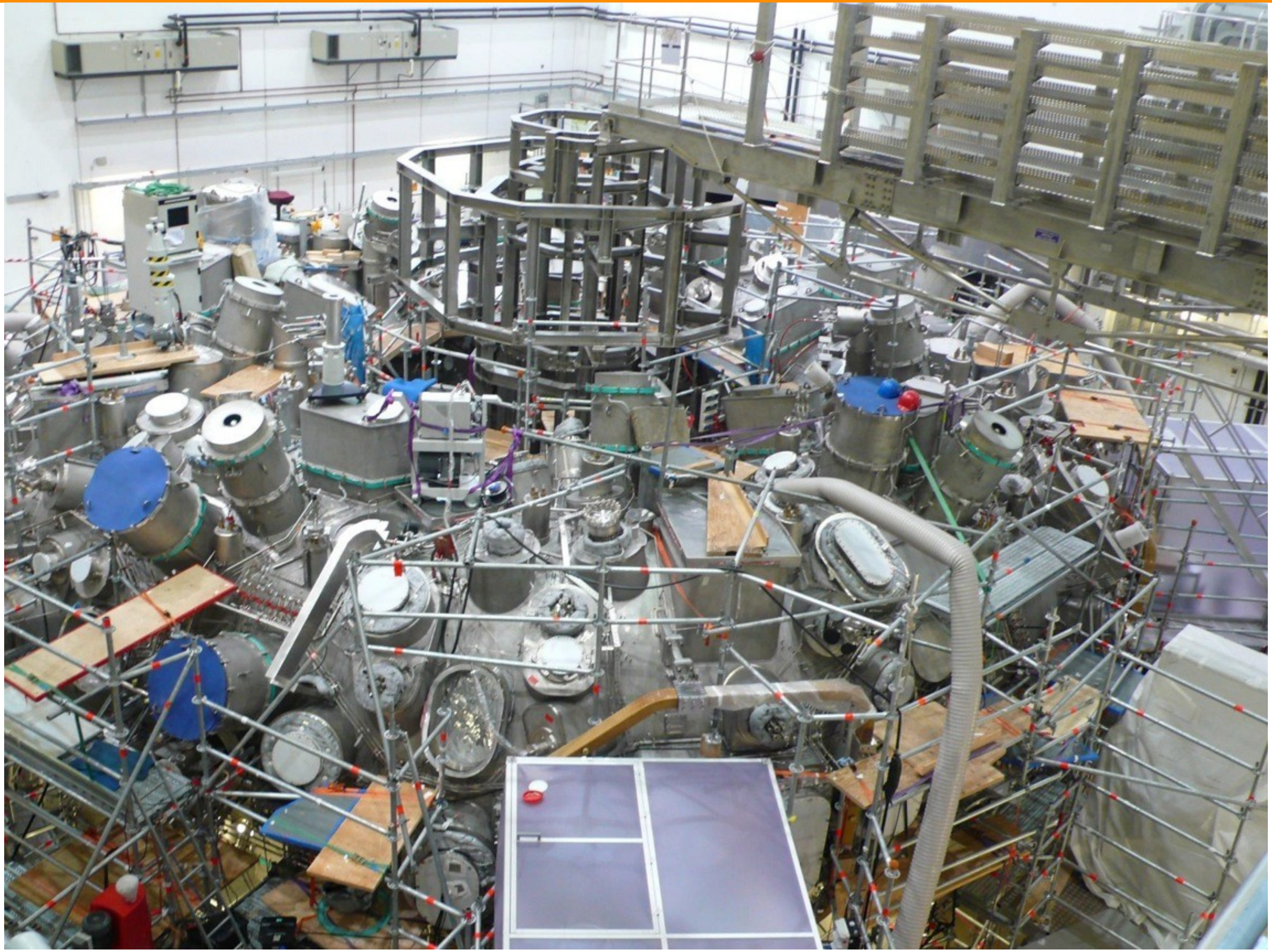
First magnetic assembly in cryostat of the W-7X stellarator started in October 2009


Bosch et al., IAEA-FEC 2012

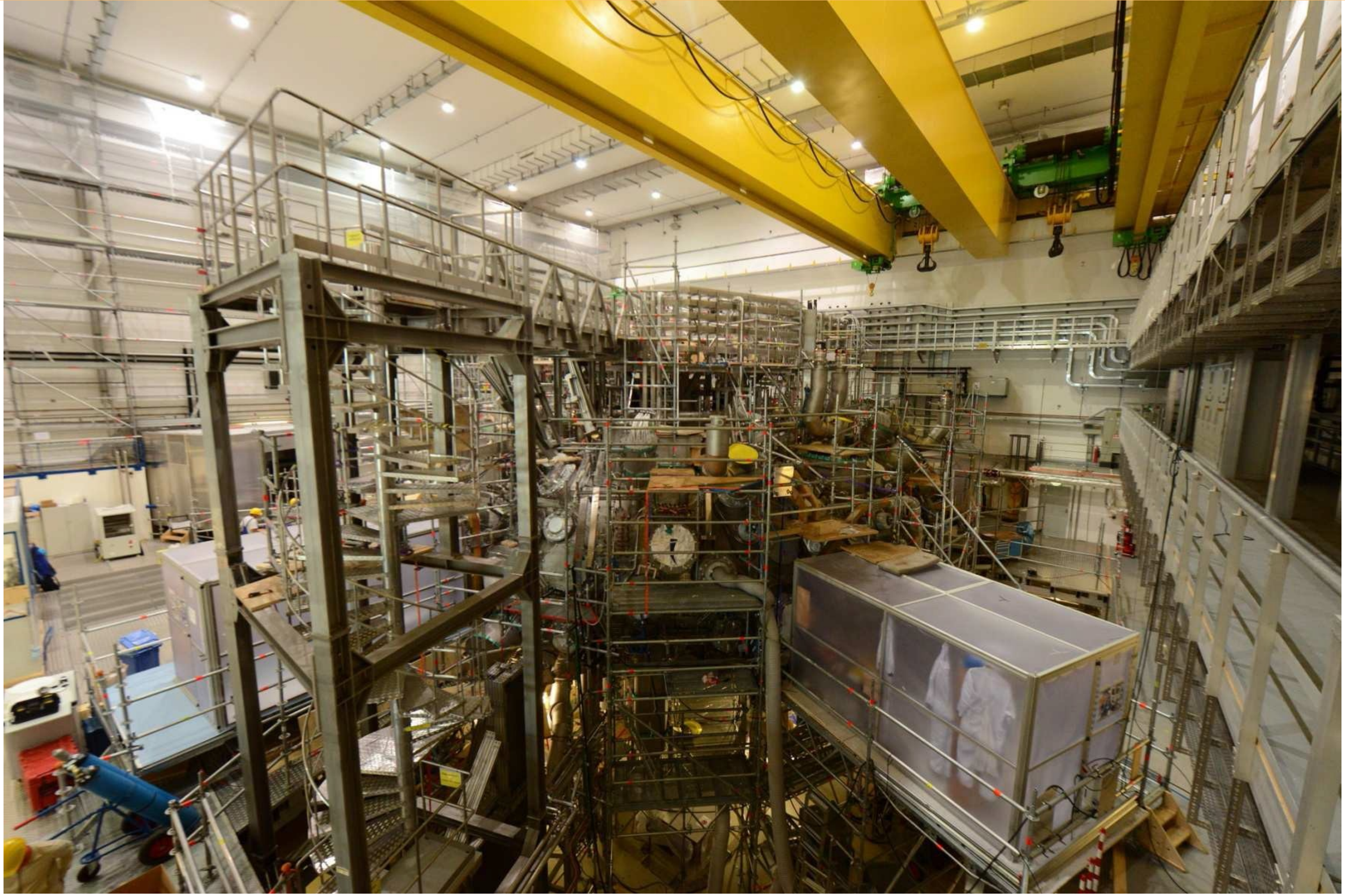
The Wendelstein 7-X hall in early 2013



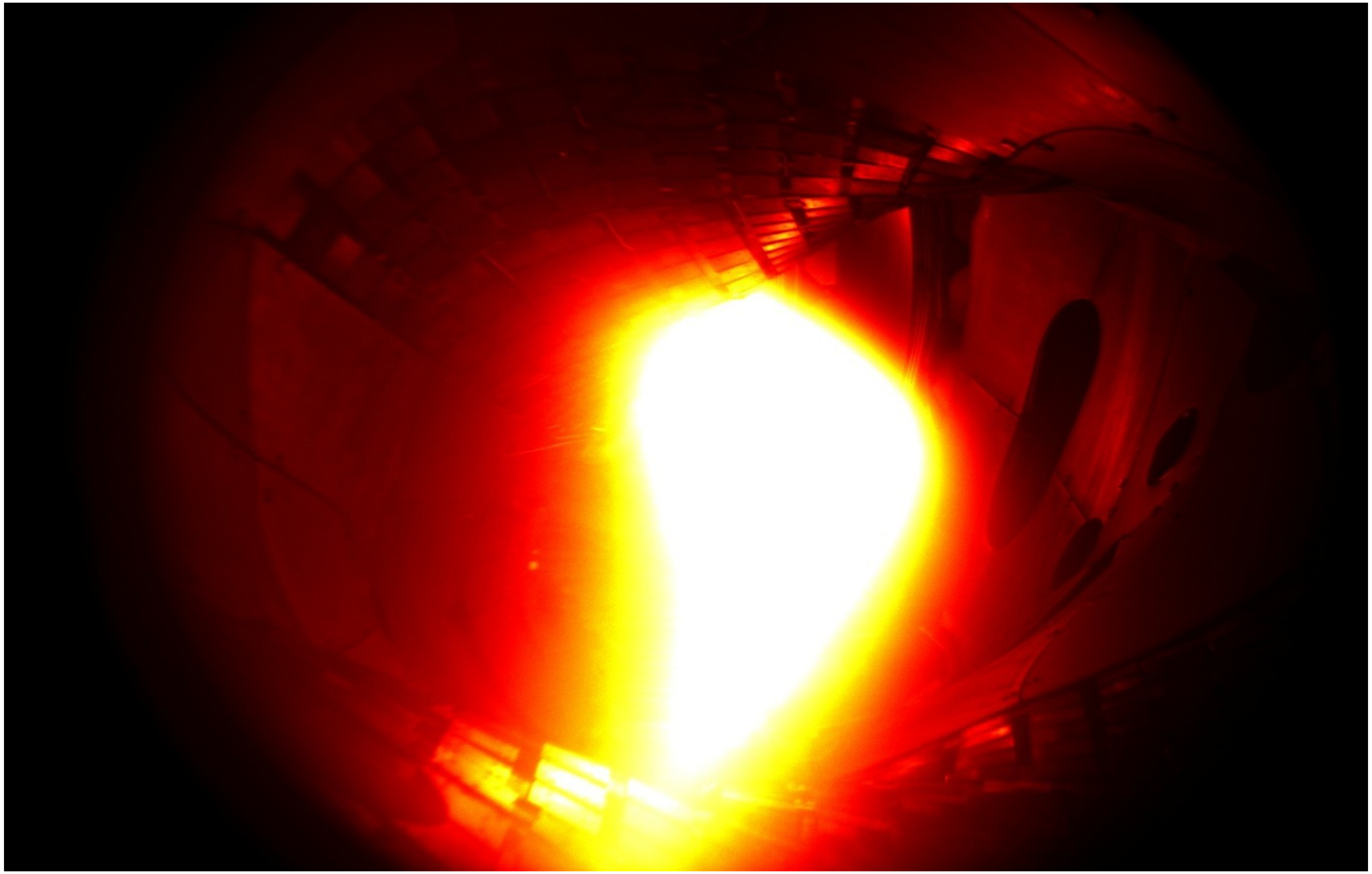
The Wendelstein 7-X hall in August 2013



Assembly of Wendelstein 7-X completed in June 2014 ⇒ start of extensive commissioning

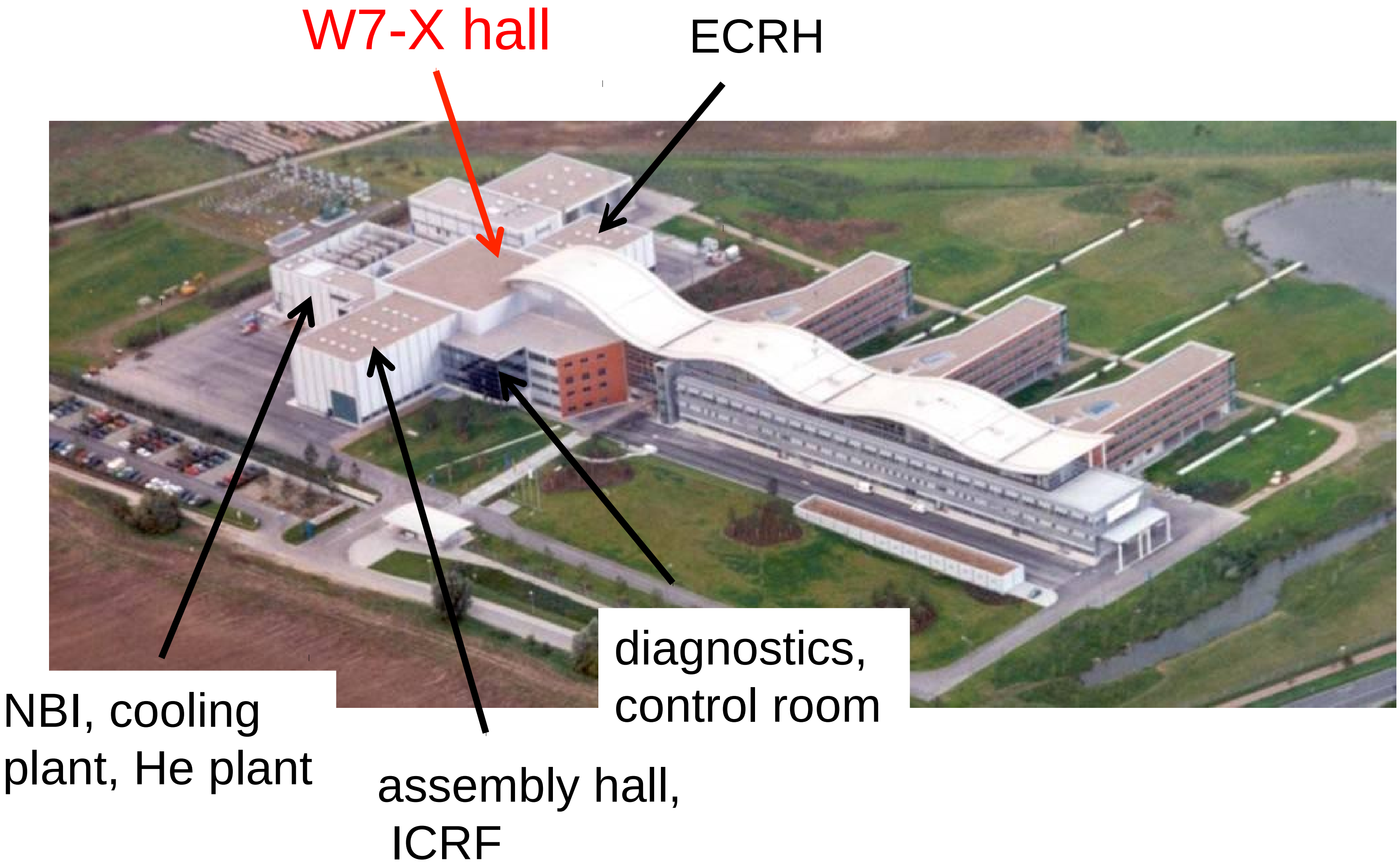


The first (He) plasma in Wendelstein 7-X was obtained on December 10, 2015 (100 ms long)

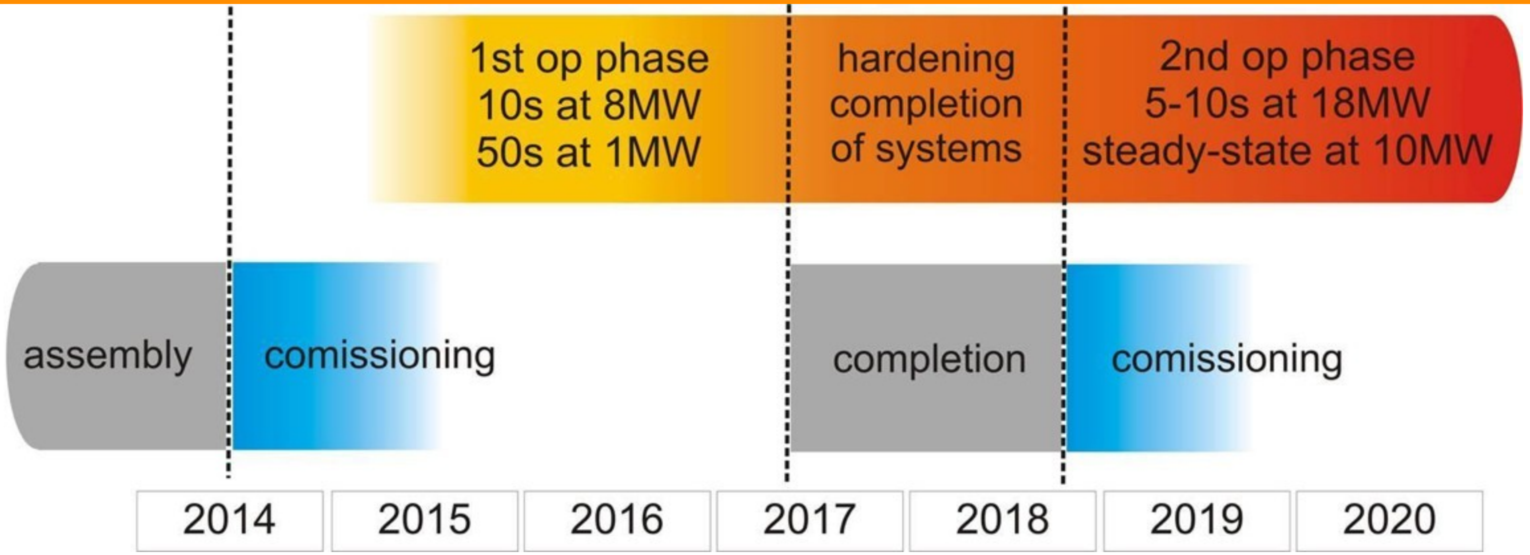


Angela Merkel switches on Wendelstein 7-X fusion device (first hydrogen plasma in Feb 2016)

W7-X is hosted by the Institute for Plasma Physics in Greifswald, Germany (project since 1994)

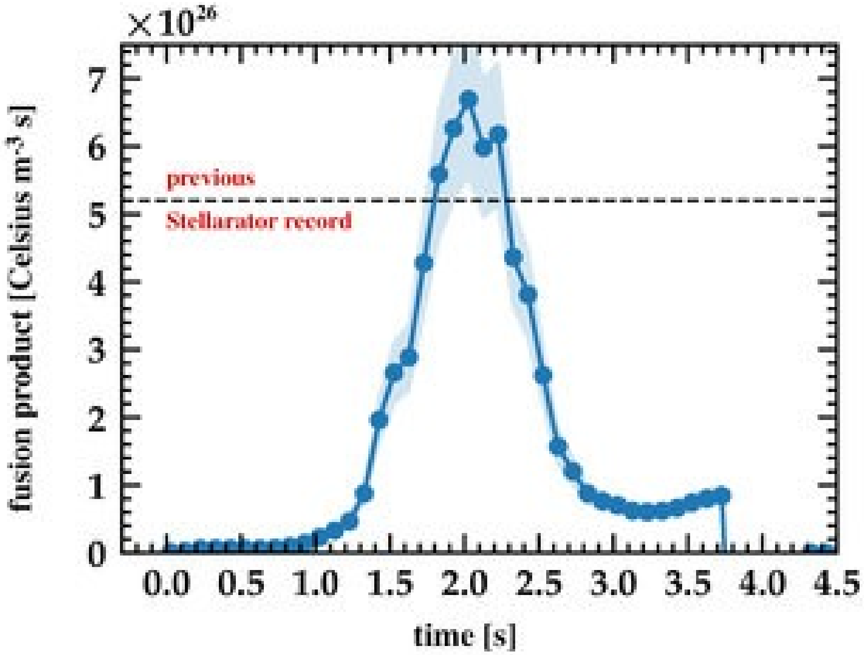


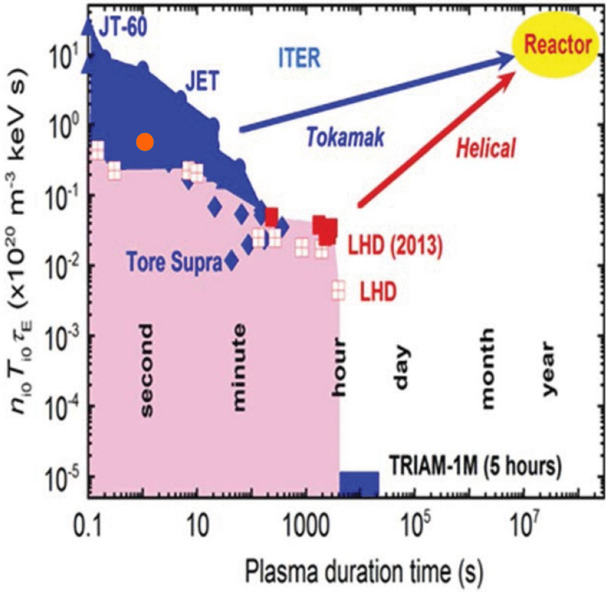
The 1st operation phase of W7-X is to verify the stellarator optimization and develop integrated high-density scenario



- **Commissioning of vacuum vessel, magnetic field, field line tracing, plasma startup ⇒ first plasma Dec-2015**
- **1st operation phase with inertially cooled divertor, some in-vessel components cooled**
- **No provision for D-T operation**

New world record in stellarator fusion product in W7-X (press release 25.6.2018)





Compare to LHD result (Takeiri, IEEE Trans. Plasma Science, 2018)

- Fusion product 6 x 10^{26} Celsius m^{-3} s≈ 0.5 x 10^{20} keV m^{-3} s was received with at Ti = 40000000 K (> 3 keV) and n_i = 0.8 x 10^{20} m^{-3}

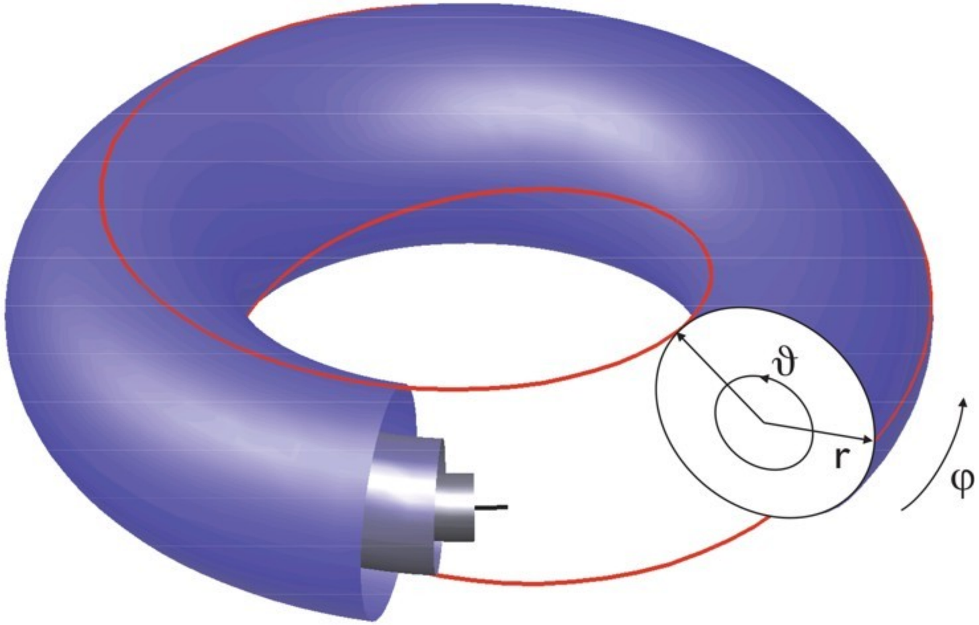
Summary

- **The equilibrium in a stellarator is established by external coils only (3D) ⇒ can naturally be operated in steady-state and no current-driven disruptions**
- **Good nested flux surfaces with small islands can be obtained, even at high plasma pressure ⇒ island divertor for heat exhaust**
- **Loss of axisymmetry results in additional loss mechanism for particles and energy (fast particles, alphas)**
  ⇒ potentially be reduced by field optimization
- **Stellarators can be operated at high-density without impurity accumulation**
- **Wendelstein-7X started plasma operation in Dec-2015**

Reserve material

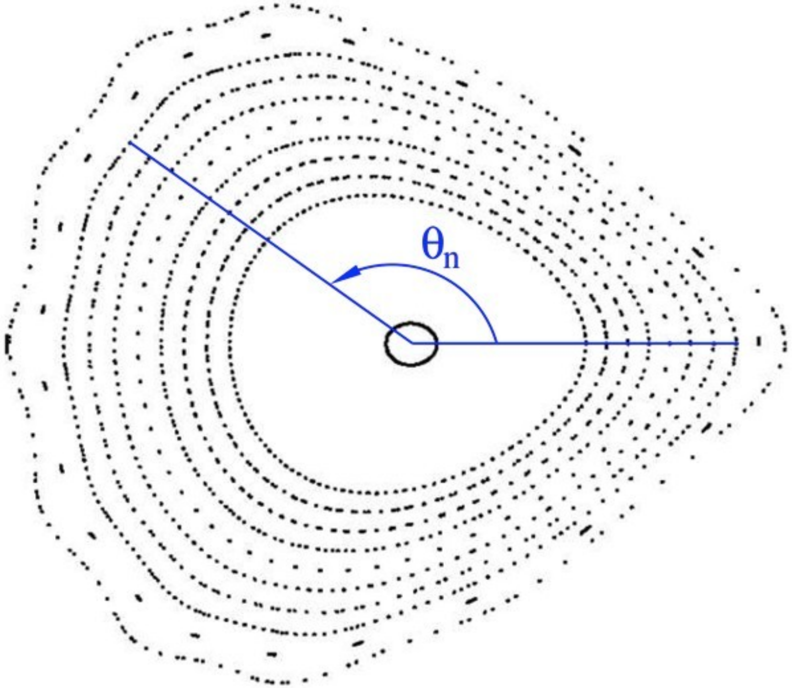
Rotational transform defines closed field lines (rational iota) and ergodic regions (irrational iota)

Tokamak



Stellarator

(Poincare plot)



$$q(r) \equiv \frac{rB_T(r)}{R_{major}B_P(r)}$$

inverse

$$\Leftrightarrow$$

$$\iota \equiv \lim_{n\to\infty} \frac{\theta_n}{2\pi n}$$

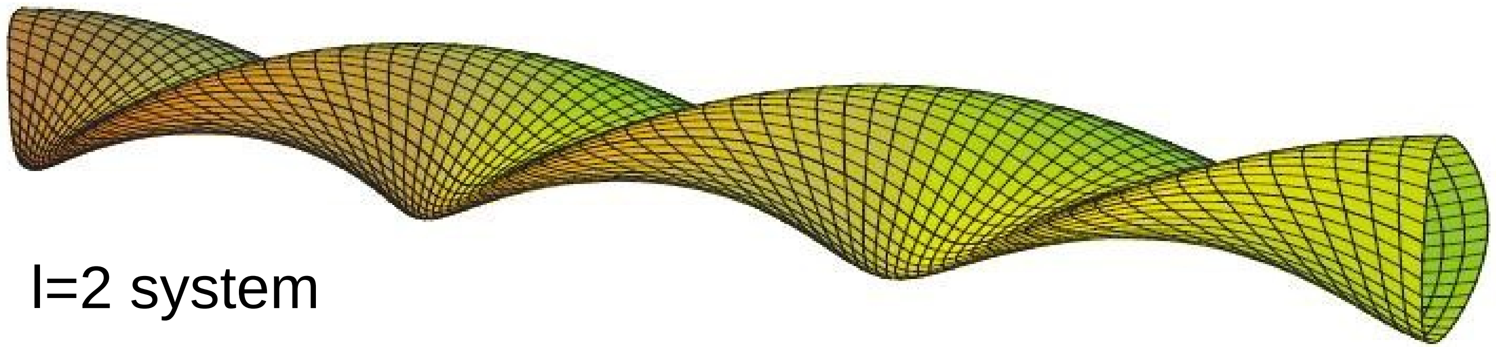
Parameterize magnetic geometry in a straighten-out stellarator of pitch k

- **Assume helical symmetry:** $$\vec{B} = \vec{B}(r, \vartheta - kz)$$

- **Vacuum field only: (pressure = 0)** $$\Phi = B_0 z + \frac{1}{k} \sum_{l=1}^{\infty} b_l I_l(lkr) \sin l\,(\vartheta - kz)$$

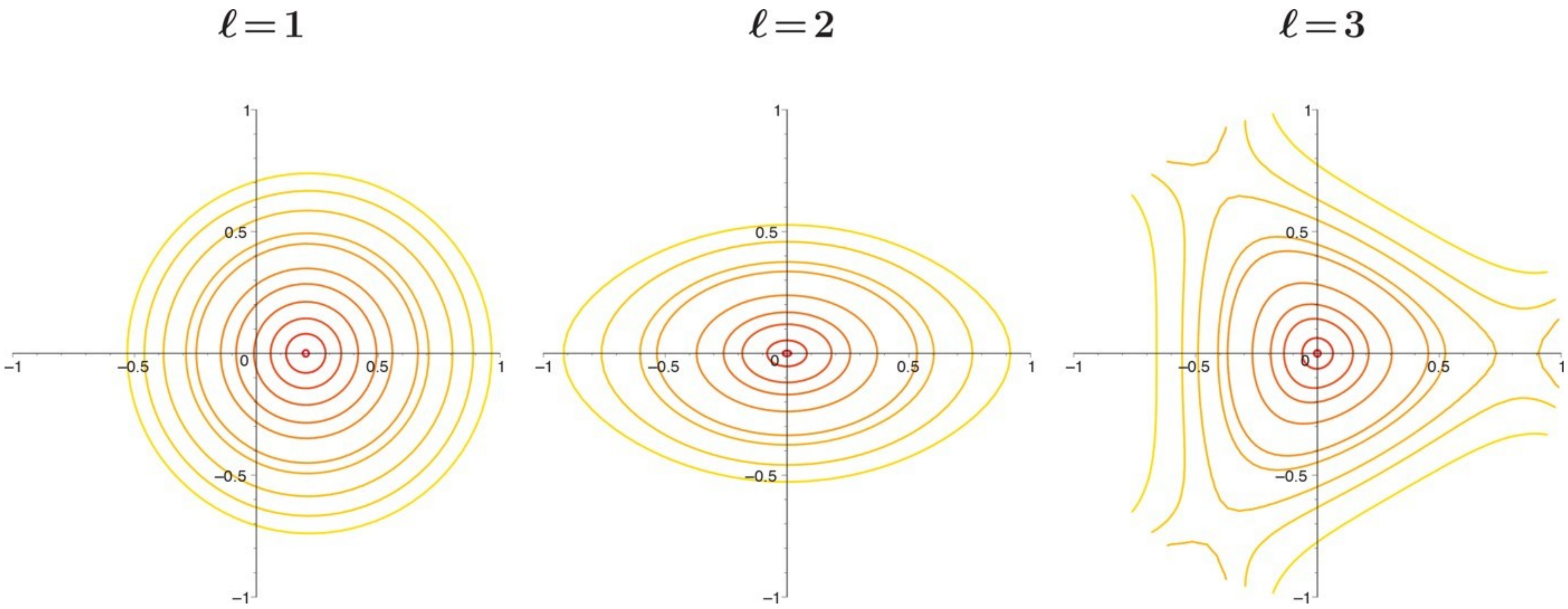
Mod. Bessel function $I_l(lkr)$

⇒ **Flux surfaces:** $$\Psi = B_0 \frac{kr^2}{2} - r \sum_{l=1}^{\infty} b_l I_l'(lkr) \cos l(\vartheta - kz) = const.$$



l=2 system

The Bessel function parameter l determines the dominant helical harmonic



- **l=1 systems: shifted circles**
- **l=2 systems: elliptical with the center on-axis**
- **l=3 systems: triangular shape**