39th Annual Meeting of the FPA 2018

First operation of the Wendelstein 7-X stellarator and expectations for the future

Hans-Stephan Bosch

Max-Planck-Institut für Plasmaphysik Greifswald, Germany

on behalf of the Wendelstein 7-X Team

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Content of the talk

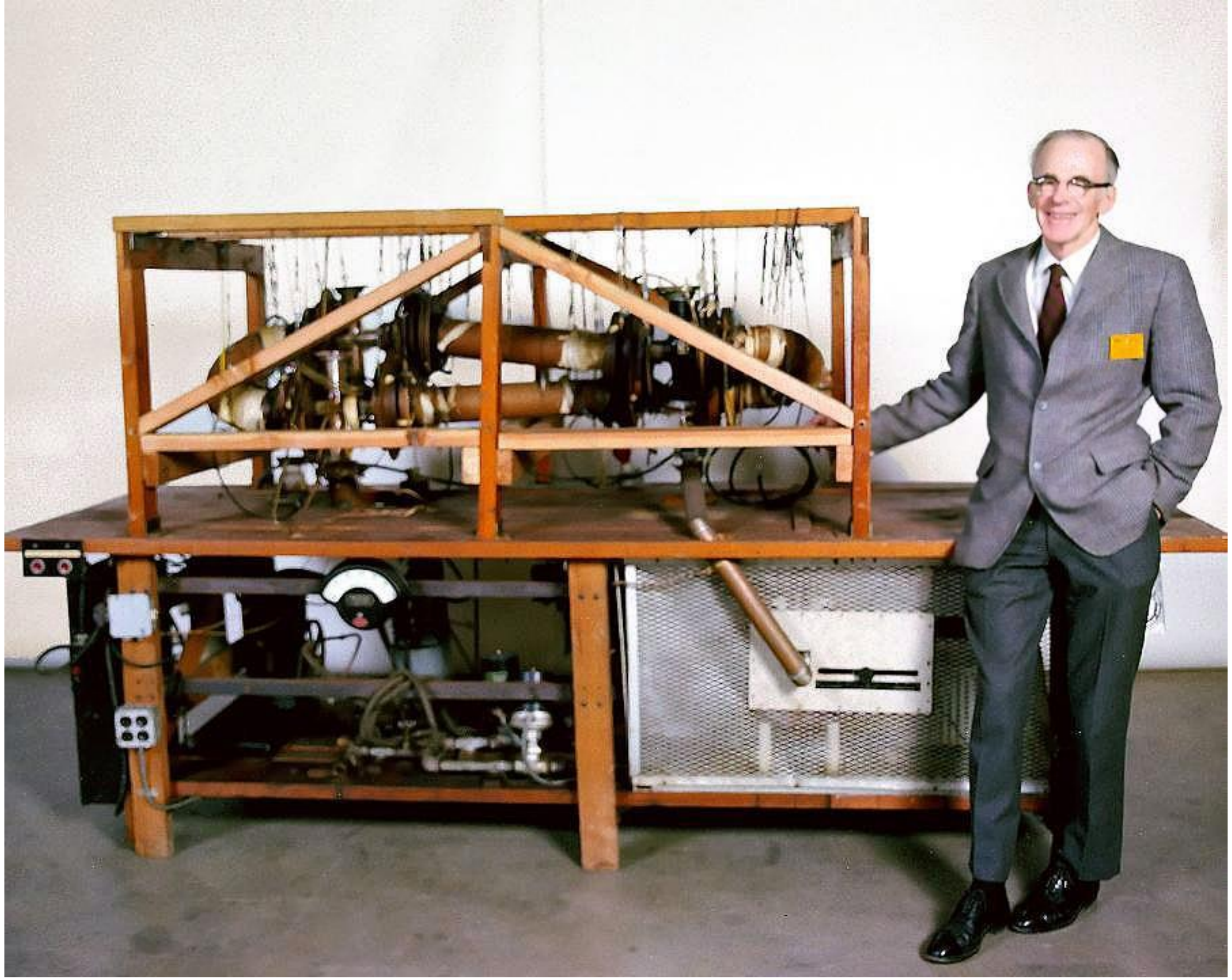
One-team approach for operation:

- The W7-X Team includes researchers and engineers from IPP, Euratom, US (DoE), and Japan.
- It will be published in the Nuclear Fusion paper of the IEAE FEC 2018.

Content of the talk

1. **Why stellarators?**
2. **Technologies of W7-X**
3. **Performance**
4. **Conclusions and the future**

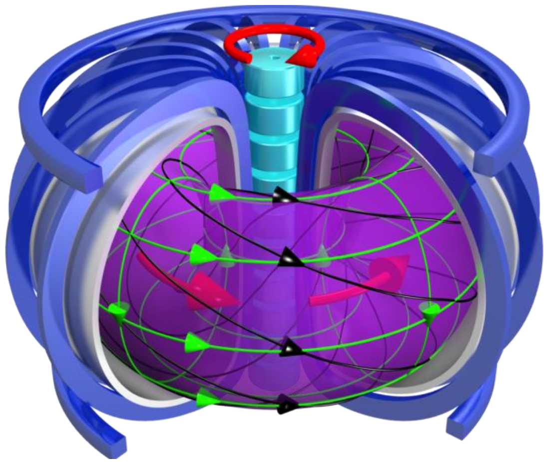
Stellarators



- 1951 invented by Lyman Spitzer jr. in Princeton in a classified report.
- Project Matterhorn
- First stellarator operated in early 1953, as figure-8 or racetrack.
- This picture in 1983, just before donated to the Smithsonian.
- Plasma confinement was rather bad and PPPL "switched" to tokamaks.

Magnetic confinement schemes

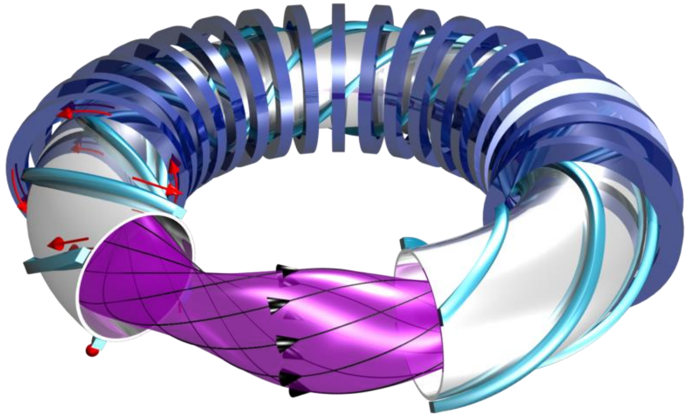
Tokamak



currents in coils and plasma

- **+ good heat isolation**
- **+ highly symmetric**
- **- pulsed operation**
- **- free energy can drive instabilities**

Stellarator

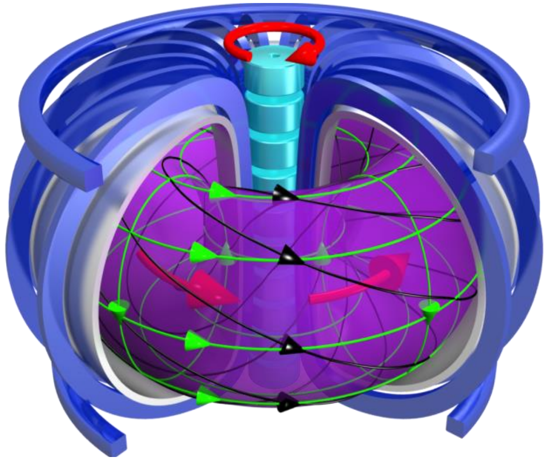


current in coils only

- **- bad heat isolation**
- **- not obviously symmetric**
- **+ steady state operation**
- **+ no current-driven instabilities**

Improved Magnetic confinement schemes

Tokamak



optimized Stellarator

currents in coils and plasma

+ good heat isolation

+ highly symmetric

○ steady state operation with current drive

○ active control of instabilities

current in coils only

○ good heat isolation

○ quasi-symmetric

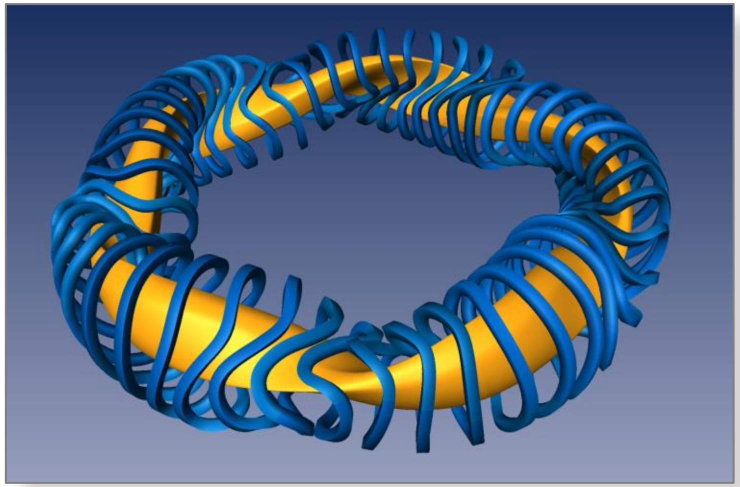
+ steady state operation

+ no current-driven instabilities

Physics optimisation of stellarators

seven optimisation criteria:

1. high quality of vacuum magnetic surfaces
2. good finite equilibrium properties @ $\langle\beta\rangle = 5\%$
3. good MHD stability properties @ $\langle\beta\rangle = 5\%$
4. reduced neoclassical transport in $1/\nu$ -regime
5. small bootstrap current in lmfp-regime
6. good collisionless fast particle confinement
7. good modular coil feasibility



J. Nührenberg et al.

Several 3d computer codes

- vacuum field and coils
- MHD equilibrium/stability
- neoclassical transport
- Monte Carlo test particle
- edge and divertor

the mission for W7X

- Confirm the numerical optimisation
- Perform high power discharges, steady-state, with an energy confiment like a equivalent tokamak

Content of the talk

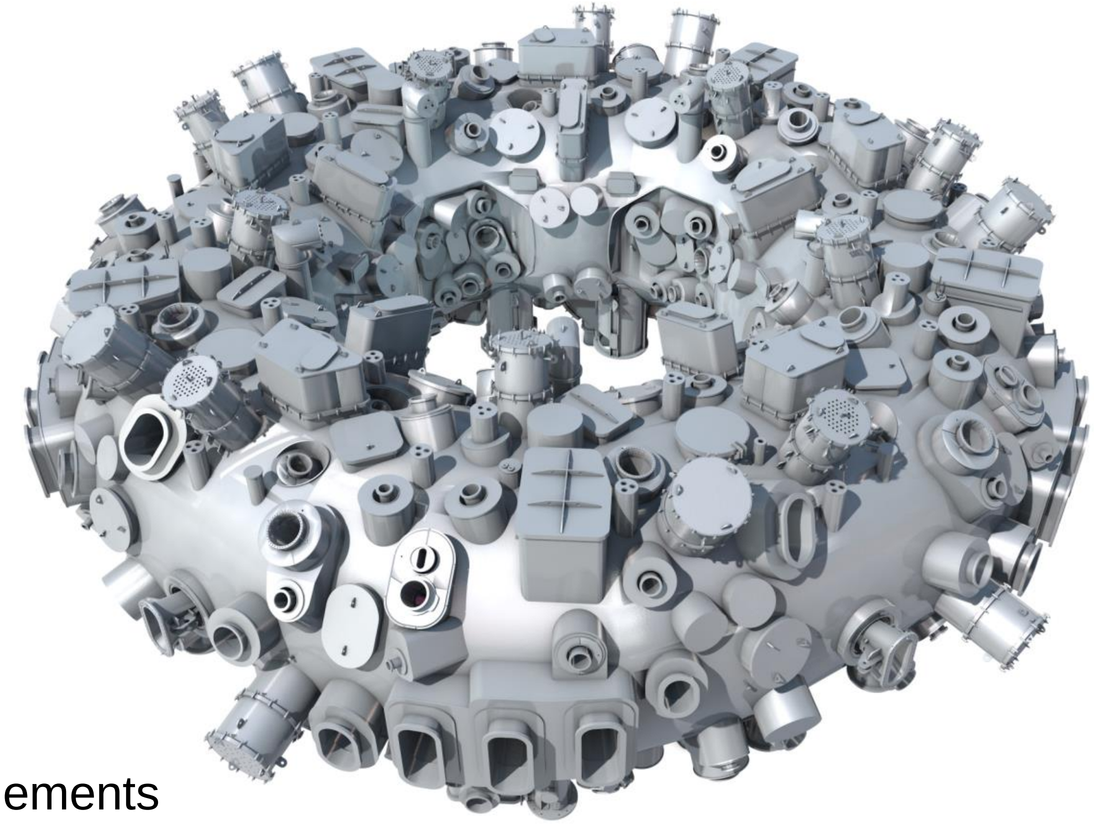
1. Why stellarators?
2. **Technologies of W7-X**
3. Performance
4. conclusions and the future

Major elements and parameters of Wendelstein 7-X

50 non-planar NbTi coils
5 types DC <18 kA

plasma 30 m^3

20 planar NbTi coils
2 types DC <16 kA



254 ports 120 shapes

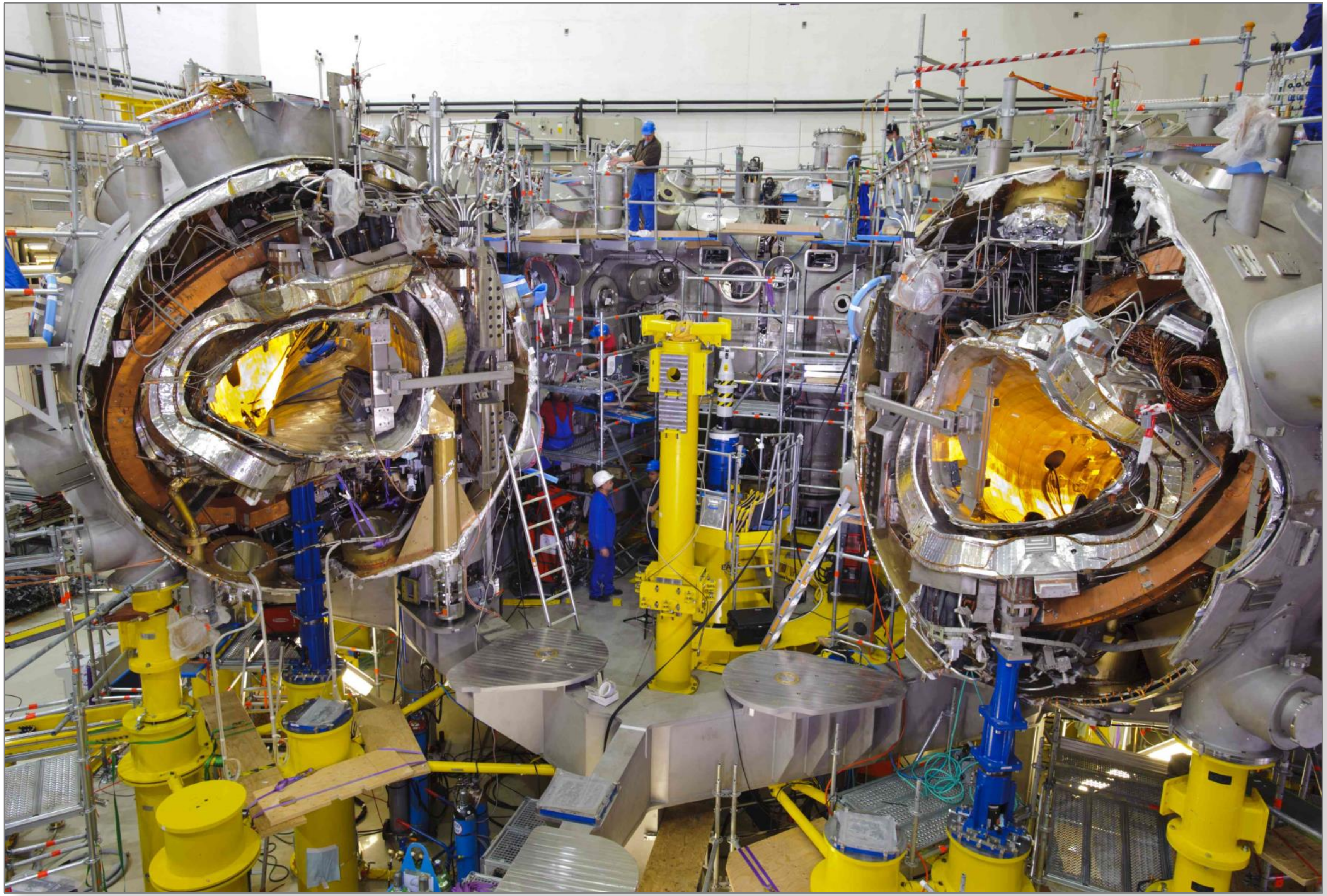
plasma vessel 80 m^3
265 m^2 in-vessel components

4.5 m machine height
16 m machine diameter
735 t device mass
435 t cold mass 3.4 K

113 NbTi bus bars
14 HTSC current leads
about 1000 helium pipes
10 central support ring elements

cryostat vessel 420 m^3
thermal insulation

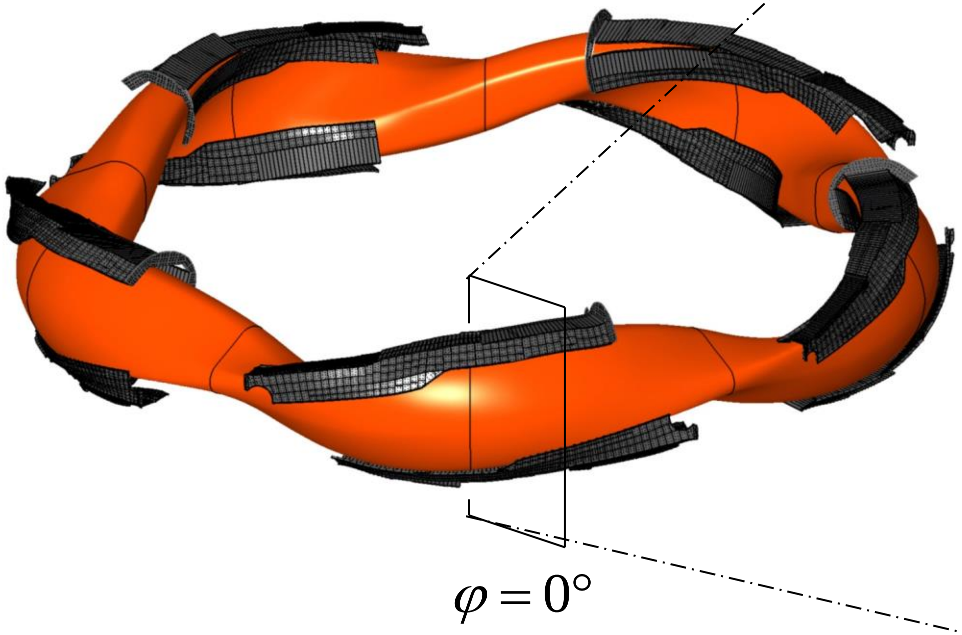
Construction of W7-X, 4/5 modules in the final position

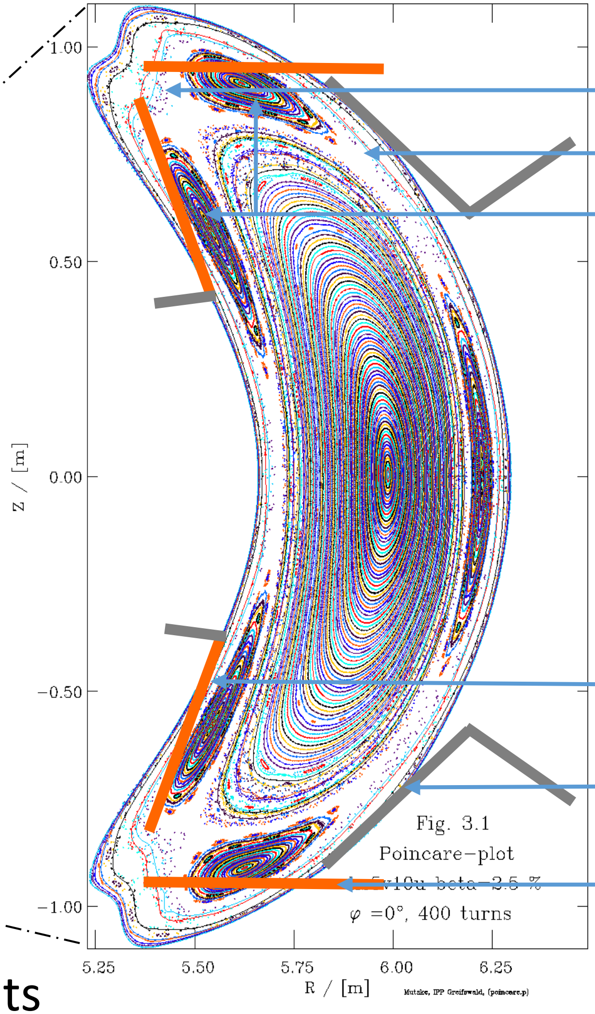


The island divertor concept

10 island divertor modules

@ bean-shaped cross sections





X-point

ergodic region

islands

total target area 19 m^2

heat flux ≤ 10 MW/m^2

connection lengths ≤ 500 m

incidence angles 2-3°

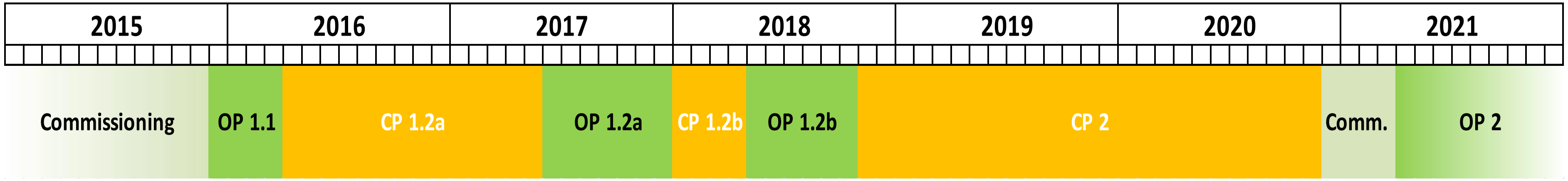
vertical target

baffle

horizontal target

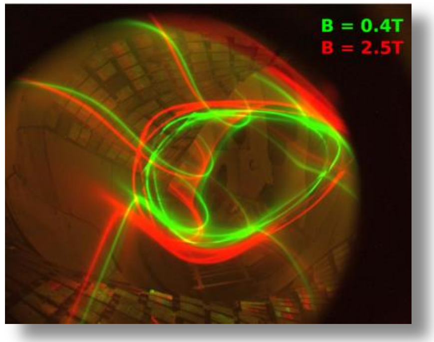
initial setup with un-cooled graphite elements
future setup with water-cooled CFC elements

staged approach to the finalization of W7X



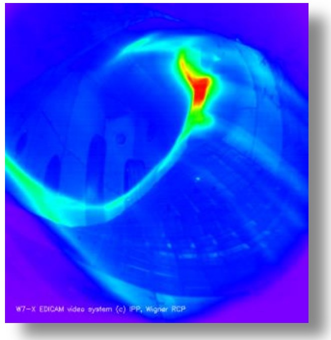
	OP 1.1	**OP 1.2a/b**	**OP 2**
Device configuration	• 5 limiters	• Test Divertor elements (TDU) • Baffle elements • Divertor closures • Graphite wall tiles • 2 TDU scraper elements (OP 1.2b)	• High-Heat-Flux divertor (steady-state Water cooling) • Port protection liners • Cryo pumps (10 units)
Operation parameters	$P < 5$ MW $\int P\,dt \leq 2$ MJ **$\tau_{pulse} \sim 1$ s**	$P \leq 10$ MW $\int \quad P\,dt \leq 80$ MJ $\Rightarrow$ **200 MJ** $\int \quad$ **$\tau_{pulse} \sim 10$ s** (... 60 s @ reduced power)	$P_{cw} \sim 10$ MW $P_{pulse} \sim 20$ MW (10 s) $P/A \leq 10$ MW/m^2 Technical limit: **30 minutes** @ 10 MW

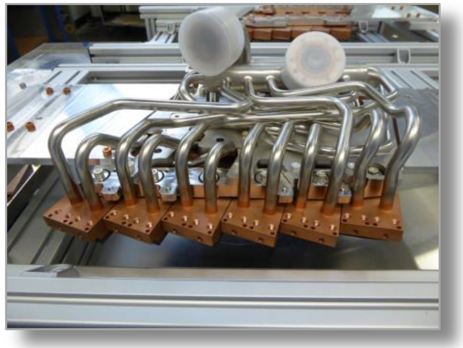
Time line of the project Wendelstein 7-X




FUSION IN EUROPE

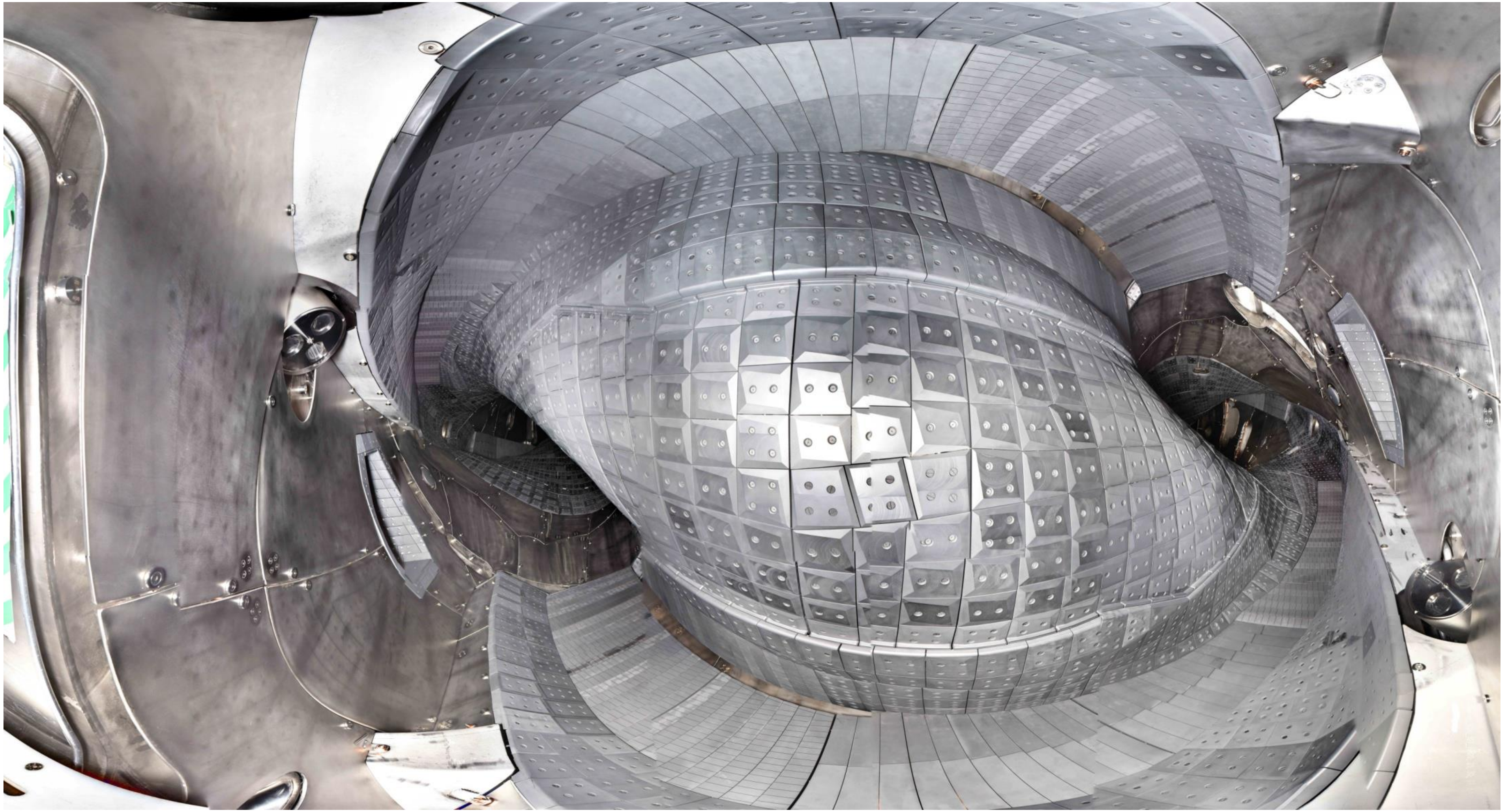
NEWS & VIEWS ON THE PROGRESS OF FUSION RESEARCH





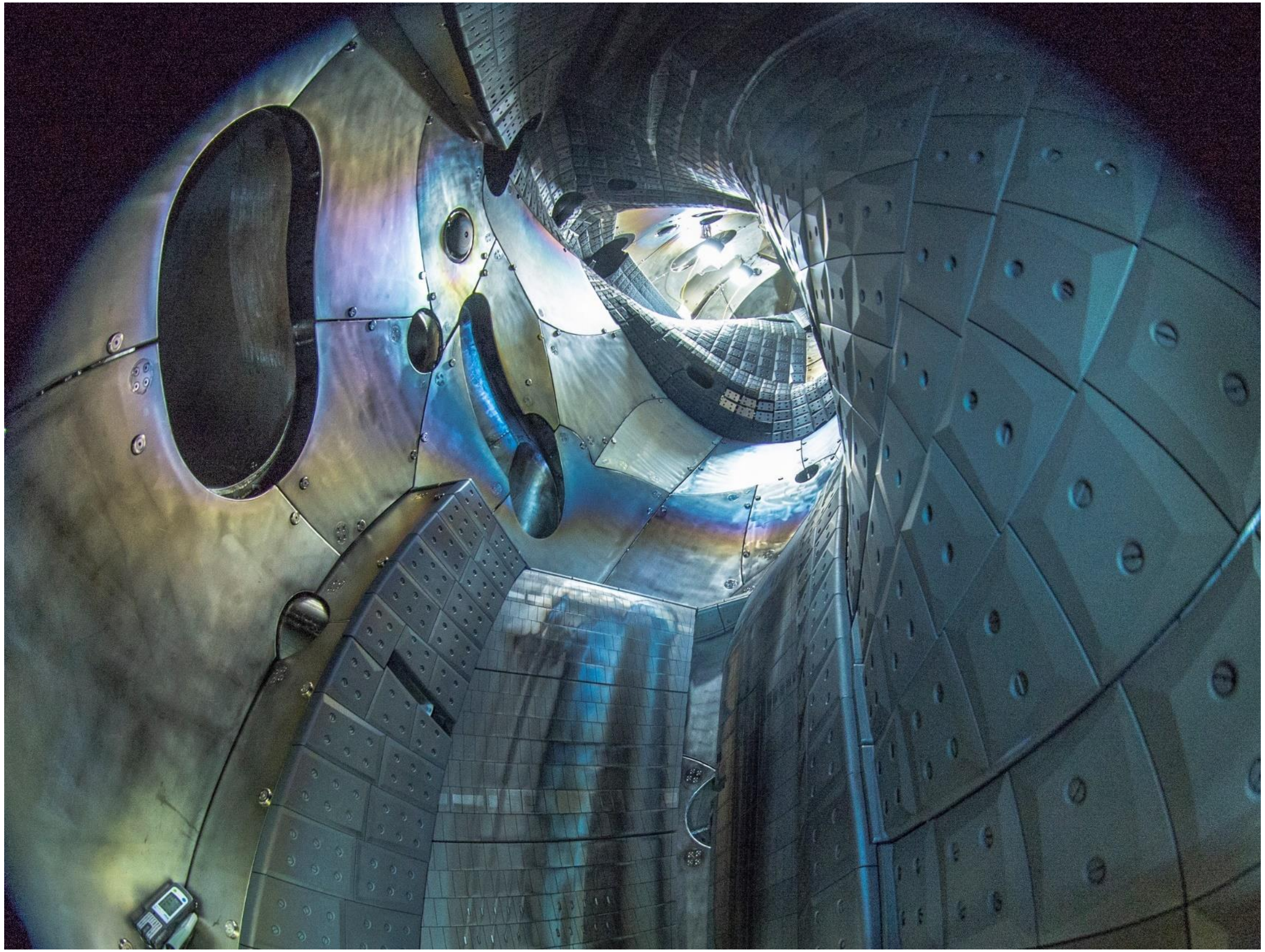
2004-2014	2014	2015	2016	2017-2018	2018-2020	2021 …
10 years assembly	pump-down cool down magnet ramp-up flux surfaces	1st plasmas 5 limiters $E_h \leq 4$ MJ $T_p \sim 1$s	18 months assembly	inertial cooling 10 divertors $E_h \leq 200$ MJ $T_p \sim 10\text{-}100$ s	24 months assembly	water cooling 10 HHF divertors $E_h \leq 18000$ MJ $T_p \sim 100\text{-}1800$s

View into the plasma vessel (May 2017, module 2)

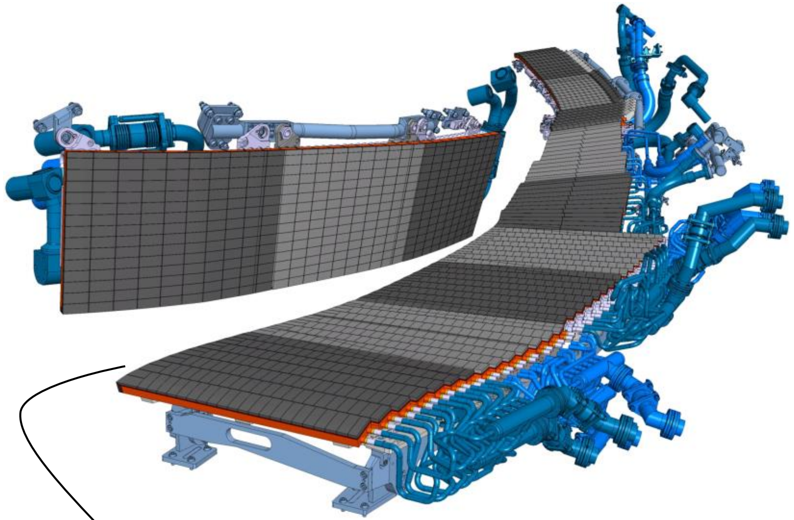


View into the plasma vessel (December 3, 2018!)

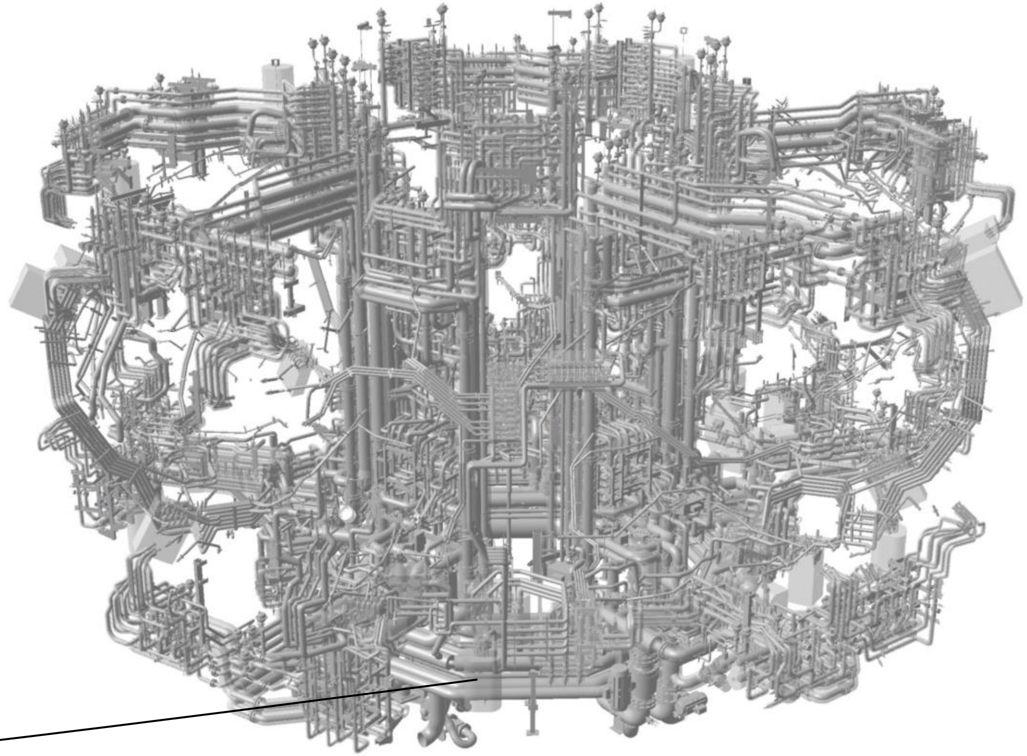
- First inspection after OP 2.1b
- Module 4, seen from M5First



Completion phase 2018-2020



- 10 HHF Divertors
- 10 Cryo pumps with LHe
- complex cooling water system

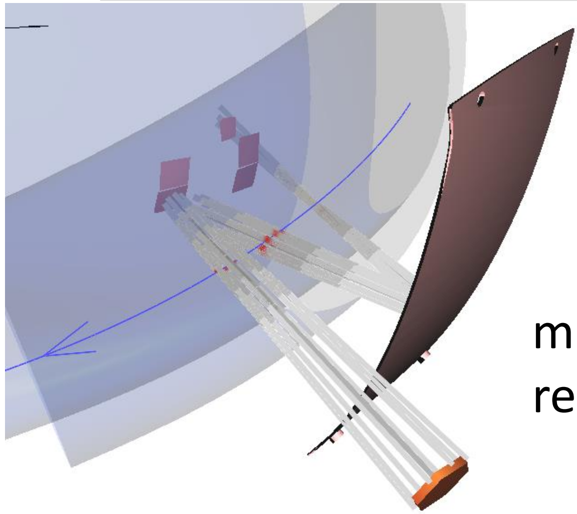


- pulse lengts up to 30 min possible
- full heating power
  10 MW ECRH, steady state
  10 MW NBI, for 10 s

Content of the talk

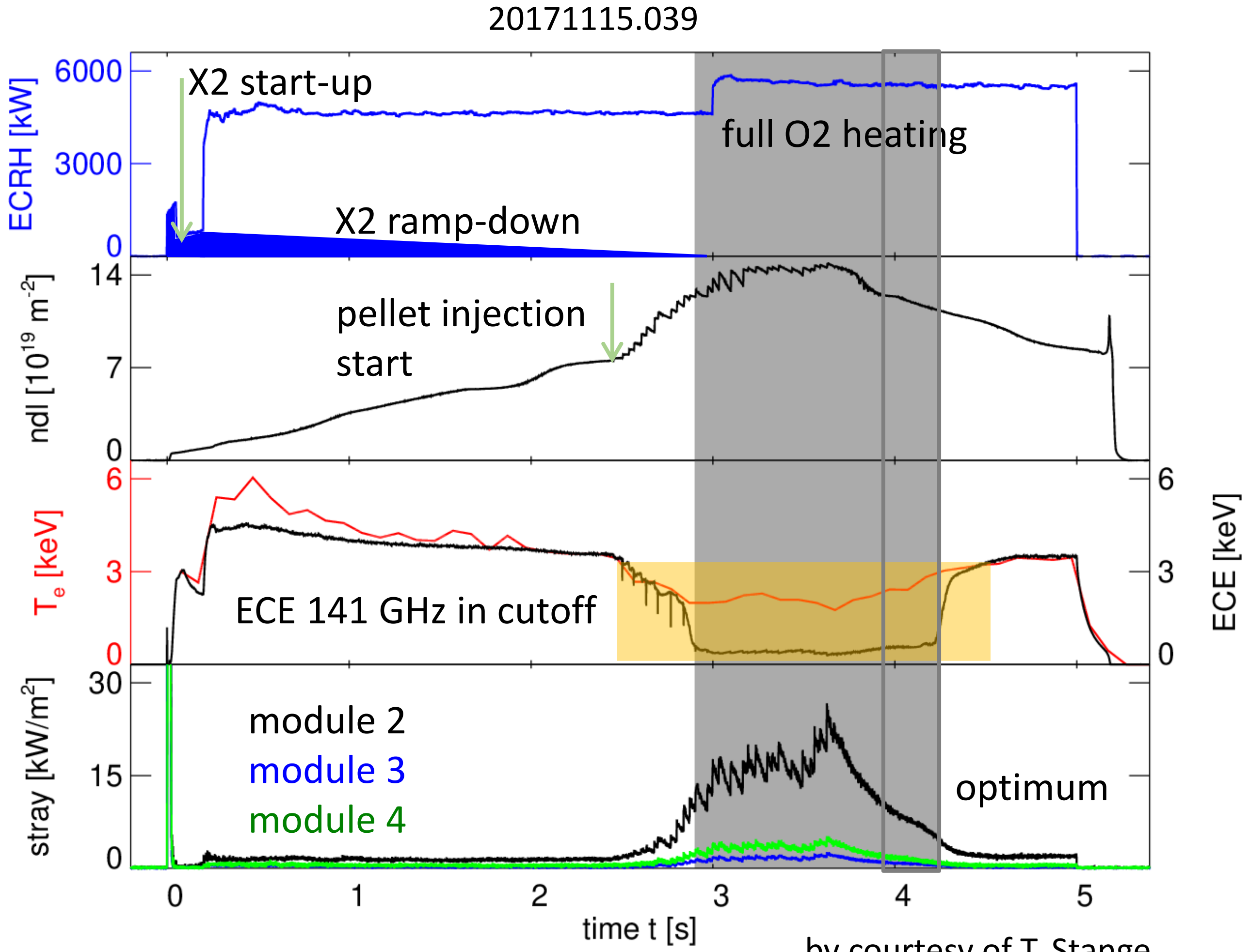
1. Why stellarators?
2. Technologies of W7.X
3. **Performance**
4. conclusions and the future

Electron cyclotron resonance heating above X2 cut-off



only 70% single beam O2 absorption (at 10^{20} m^{-3} and 3 keV)

multi-pass absorption/w reflection at tiles & panel



by courtesy of T. Stange

cut-off limits

ECR X2 heating $n_e < 1.2 \cdot 10^{20}$ m^{-3}

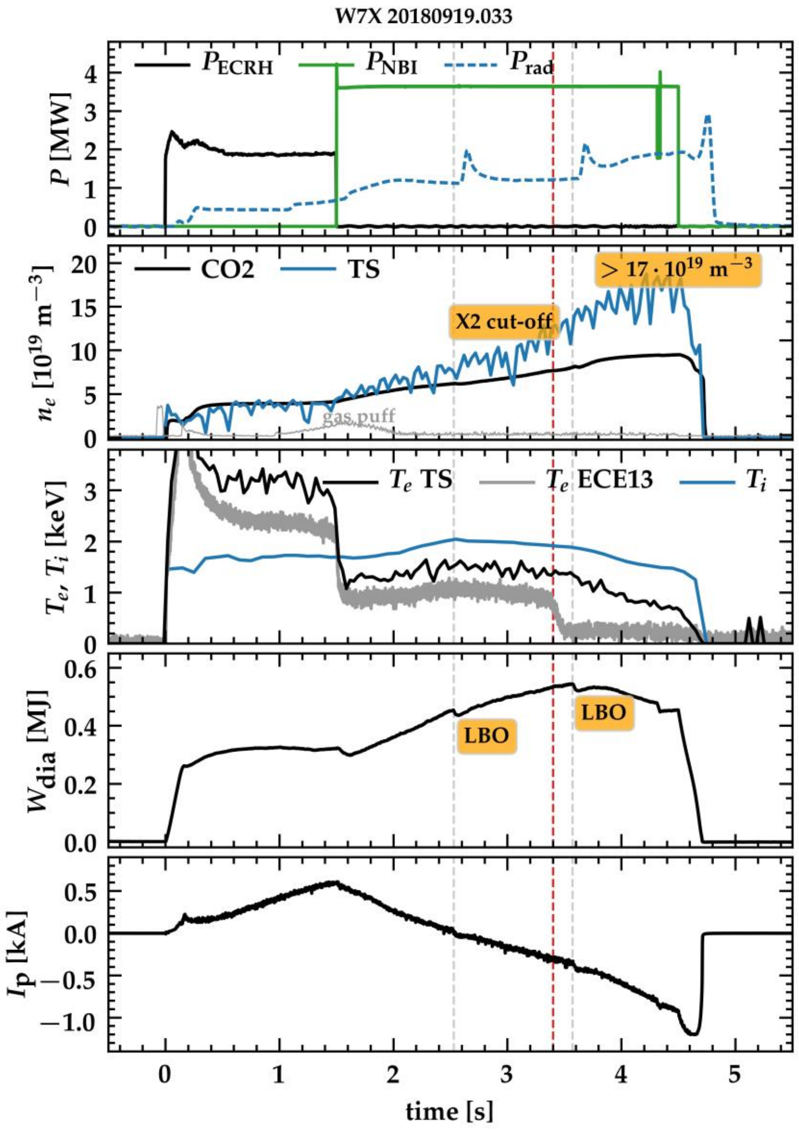
ECR O2 heating $n_e < 2.4 \cdot 10^{20}$ m^{-3}

ECE and TS

stray radiation < 20 kW/m^2

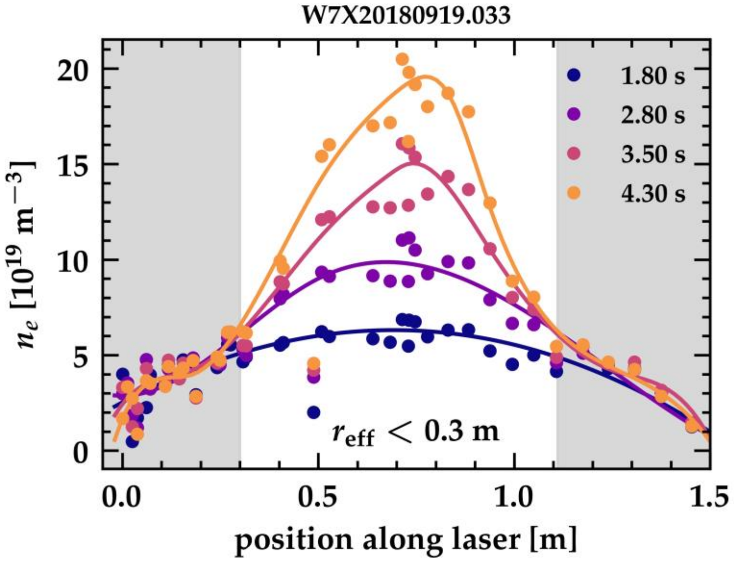
OXB mode conversion schemes under development

NBI heated discharges with high density



- pure NBI heating can sustain plasma
- stabilization of ion heating w/o ECRH

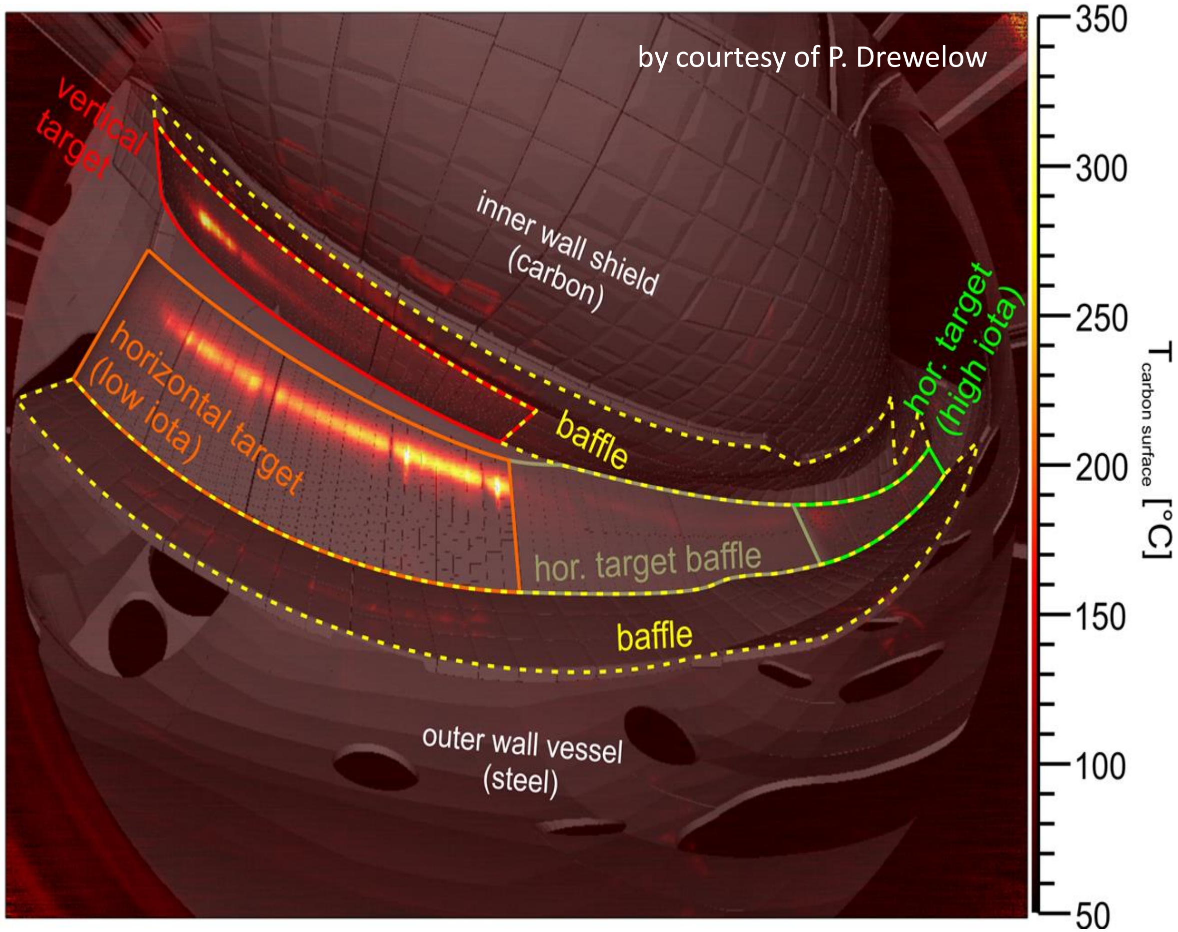
by courtesy of D. Hartmann



- centrally peaked high density plasma core
- density peaking can be controlled by additional ECRH
- pure NBI heating with $n_{peak}(0)=2 \cdot 10^{20}$ m^{-3} demonstrated

Divertor heat flux

10 high resolution IR cameras

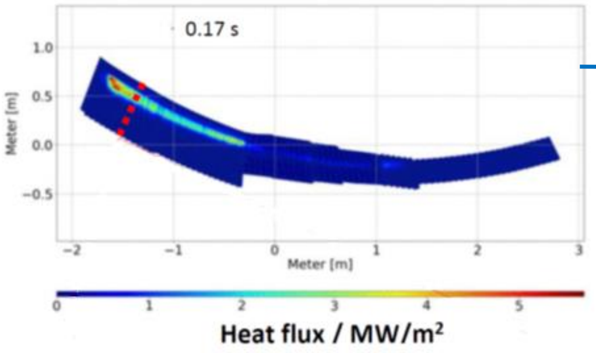


asymmetry toroidal ≤ 40 / 25% (trim coils)
up/down ~ 10-20%

power fall-off length $\lambda_q = 15 \ldots 30$ mm
power wetted area $A_{wet} \sim 1.2\ m^2$

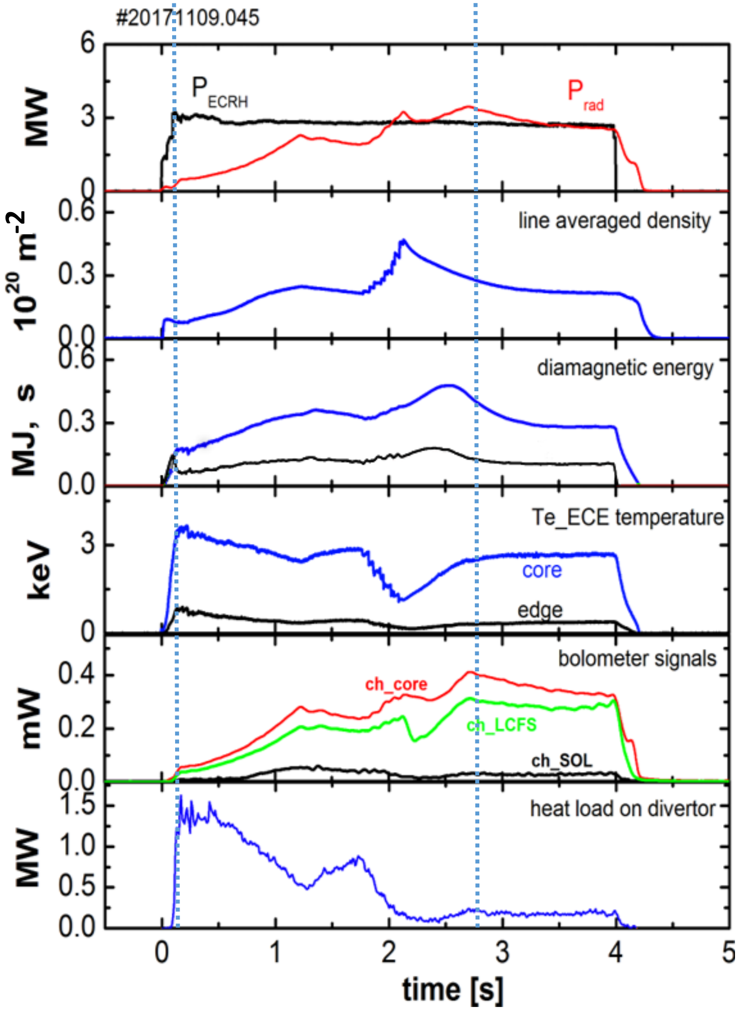
strike lines defined by the long connection lengths ranging between 200 m and 600 m

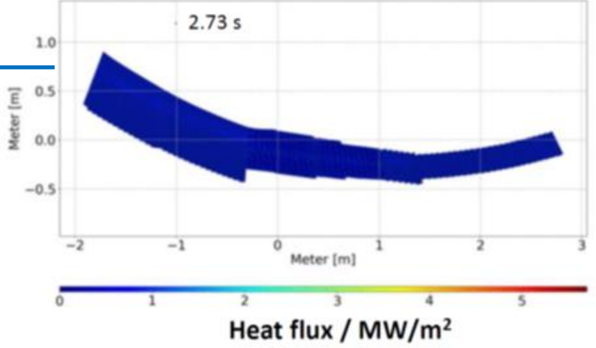
Power detachment



before detachment

detachment by pellet injection

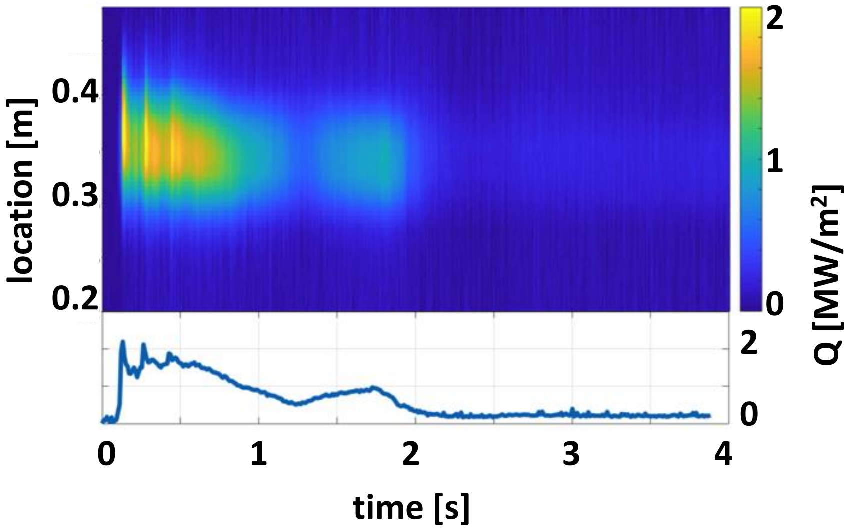




after detachment

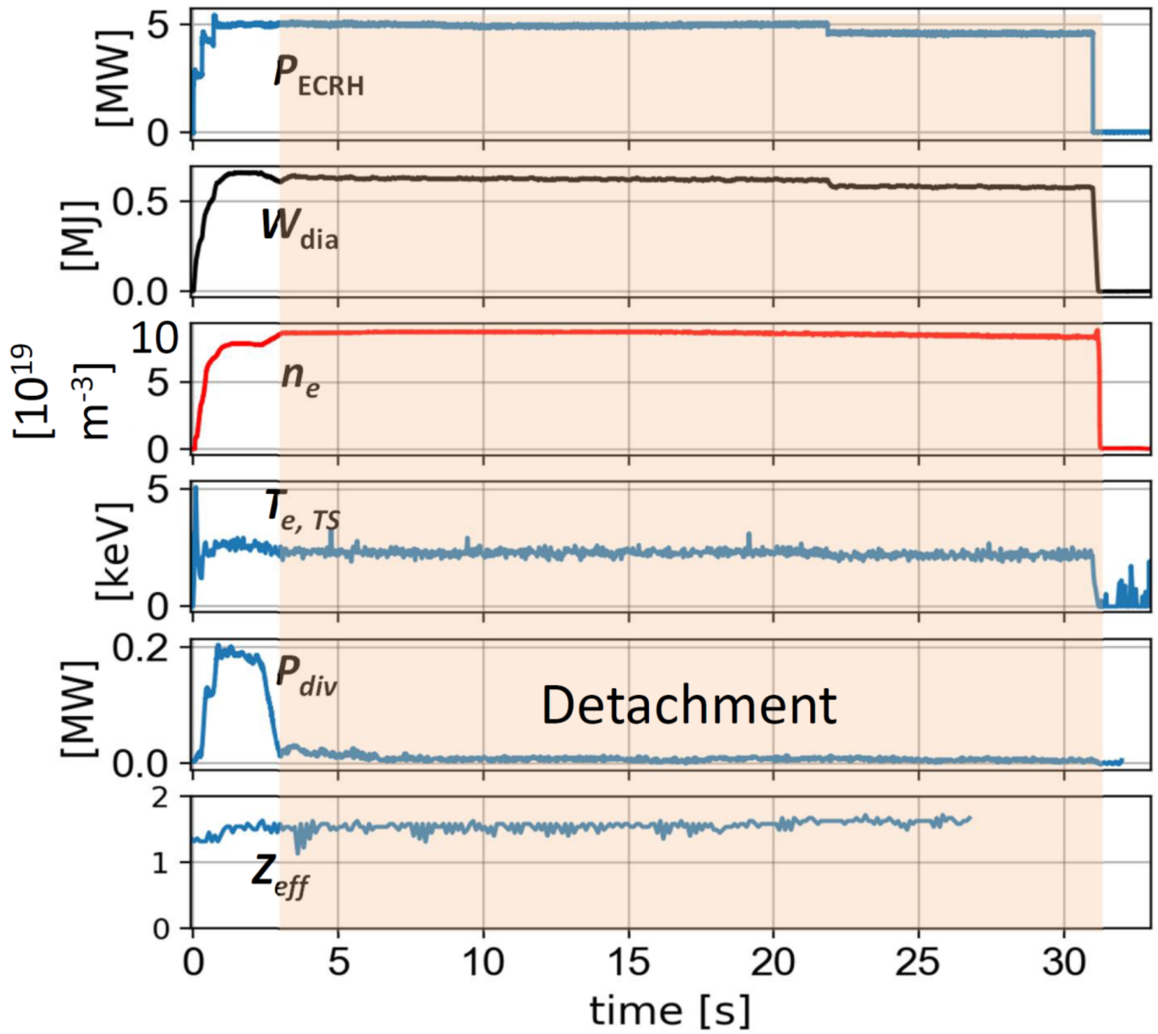
H pellet injection into He density ramp-up to $4\text{-}5\cdot 10^{19}\,\mathrm{m}^{-2}$

time evolution of heat flux



power load drops to ~ zero

Long pulse discharge with divertor detachment



O2 ECR heating with 5 MW power 150 MJ

0.6 MJ diamagnetic energy $\tau_E \approx 0.1$ s

$0.9 \cdot 10^{20}$ m^{-2} line-integrated density
30 s flat-top density feedback control

2.5 keV central electron/ion temperature

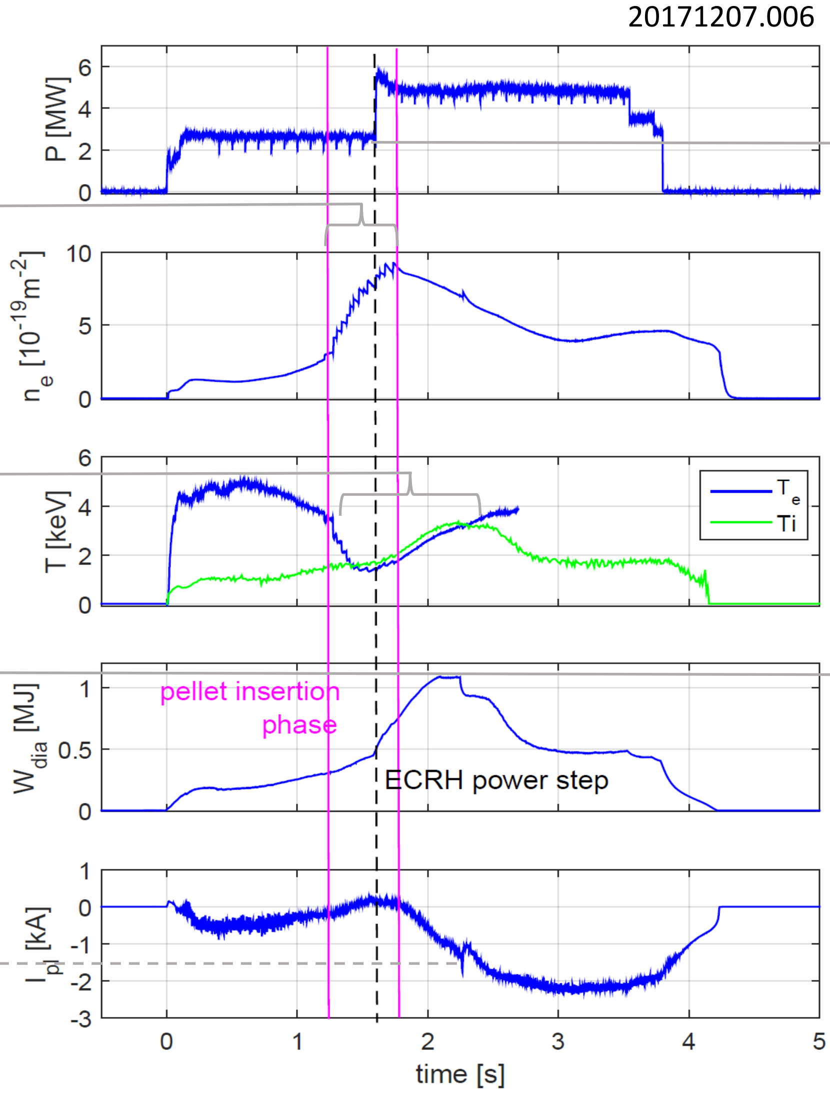
full detachment – div. pressure 0.07 Pa

Z_{eff} constant over the full discharge

reference scenario

A record high performance plasma

by courtesy of S. Bozhenkov



X2 ECRH power doubled

hydrogen pellet injection
into helium target plasma

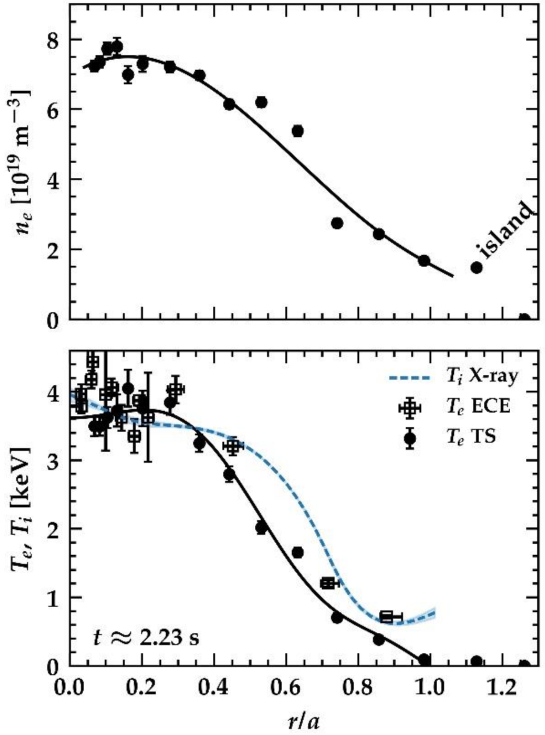
record triple product
$\geq 0.6 \cdot 10^{20}$ keVs/m^3

thermalization $T_e = T_i$

diamagnetic energy ≥ 1 MJ

MHD event

by courtesy of S. Bozhenkov



Content of the talk

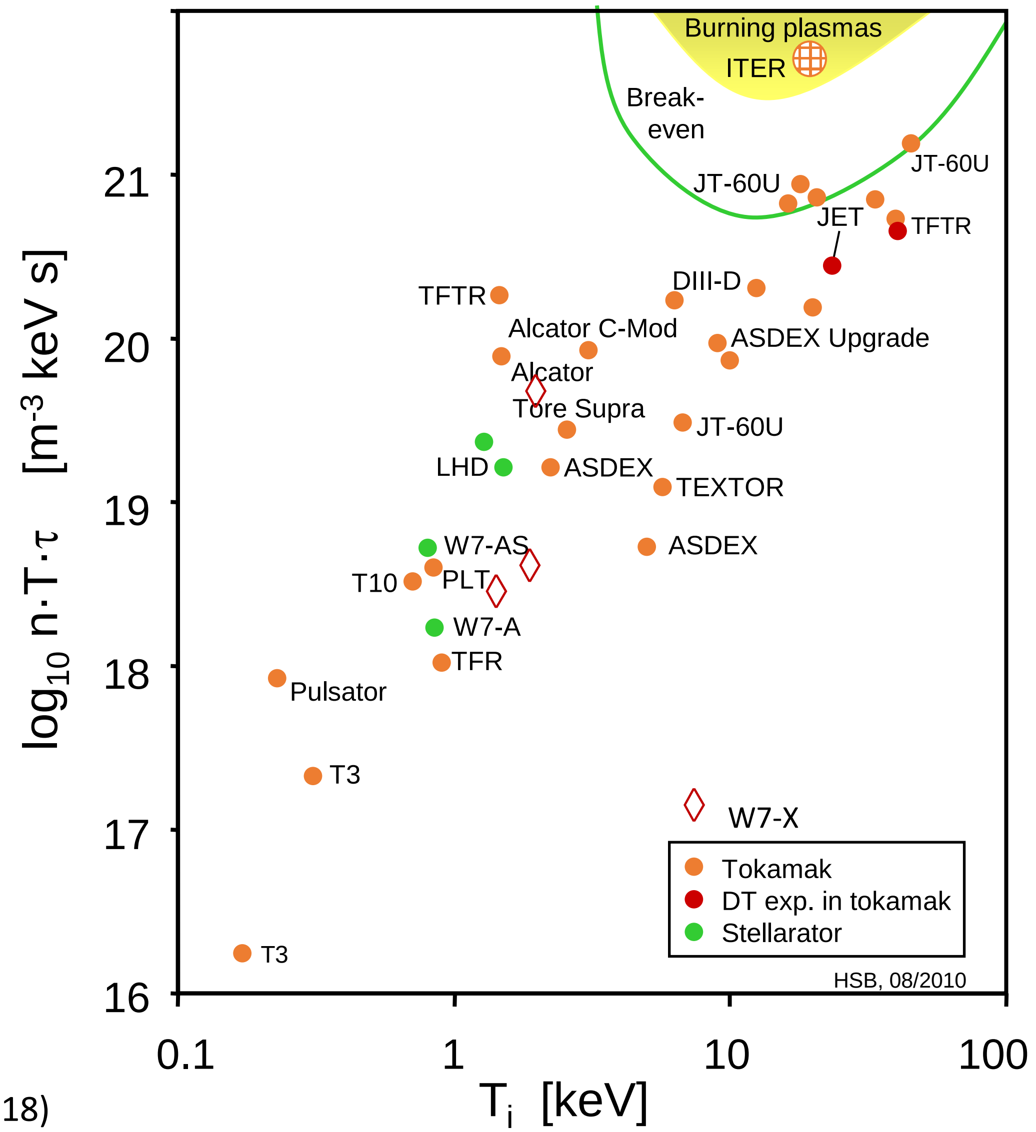
1. Why stellarators?
2. Technologies of W7-X
3. Performance
4. **conclusions and the future**

nTτ-diagram

- J. D. Lawson, 1957
- power balance for a fusion plasma
- 2 curves derived
- logarithmic scale

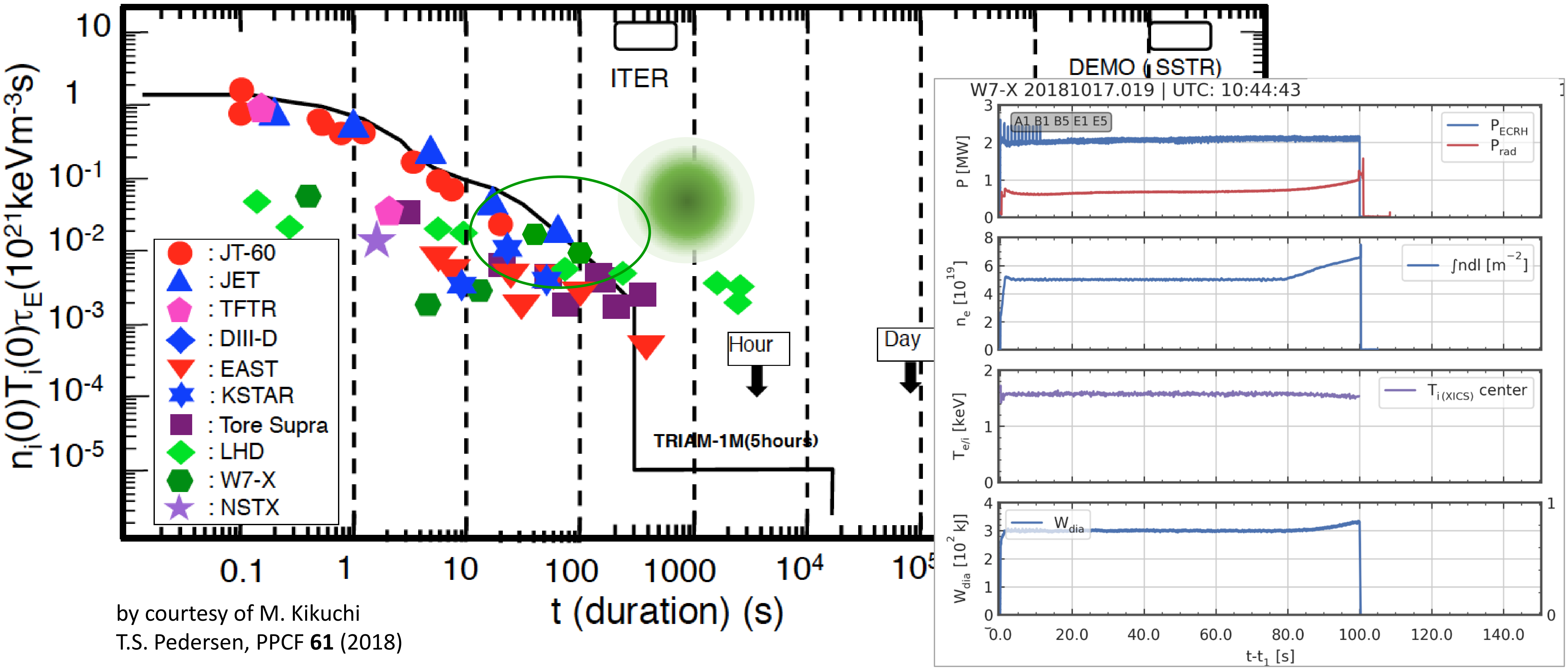
- a measure for the success of fusion
- factor 10^5 in 50 years

- However, most of these values are not stationary!



T.S. Pedersen, PPCF **61** (2018)

Fusion triple product and pulse length



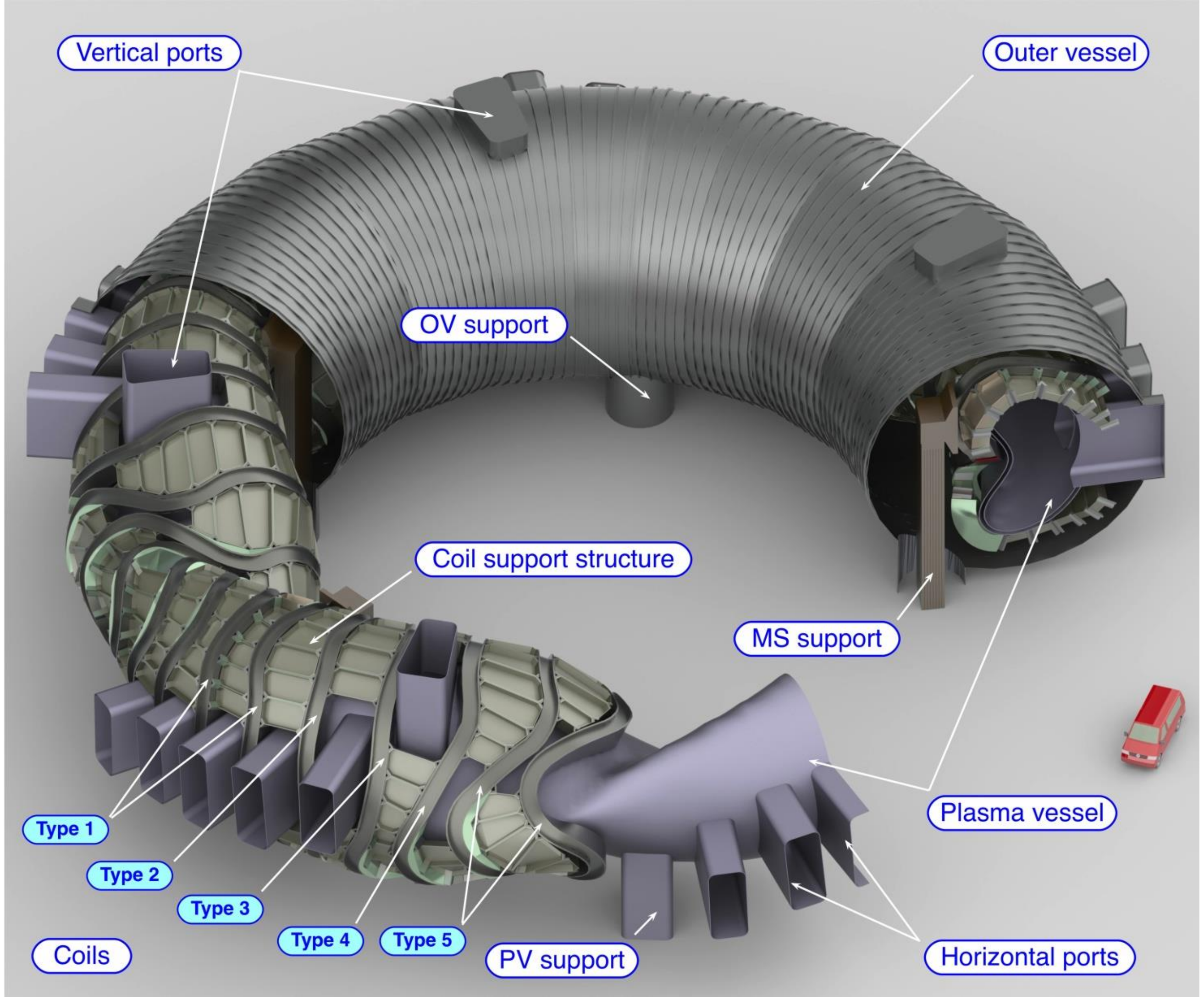
by courtesy of M. Kikuchi
T.S. Pedersen, PPCF **61** (2018)

From W7-X to a HELIAS fusion power plant

Requirements / parameters

- Average magnetic field on axis 5 – 6 T (max. field at coils 10 – 12 T)
- Size of coils and magnetic field similar to ITER (ITER coils technology can be applied)
- Sufficient space for blanket (~1.3 m between plasma and coils)
- <β> = 4 – 5 % (W7-X value!)
- Fusion power ~ 3GW
- Advantage of larger aspect ratio: Reduced neutron flux through wall (average 1 MW/m^2, maximum 1.6 MW/m^2)

R = 22 m, A = 12



Summary and conclusions

1. **Fabrication and assembly of W7-X have been tedious, but the device turned out to be highly reliable and stable – it is easy to operate and mechanically stable.**
2. **The island divertor shows even heat load distribution, power detachment and discharges with controlled plasma radiation and high neutral compression rates.**
3. **Even before its completion, W7X has reached record nTt-vaues for helical devices.**

- **Water cooling of all in-vessel components and active divertor pumping are now needed to further extend the discharge duration at high heating power.**
- **More ECRH heating power is needed and aimed for (perspective 12 × 1.5 MW).**