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
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

- currents in coils and plasma
- + good heat isolation
- + highly symmetric
  - steady state operation with current drive
  - active control of instabilities

optimized Stellarator

- 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 @ $<\beta> = 5\%$
3. good MHD stability properties @ $<\beta> = 5\%$
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

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 confinement like a equivalent tokamak

J. Nührenberg et al.
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Major elements and parameters of Wendelstein 7-X

- 50 non-planar NbTi coils
- 20 planar NbTi coils
- 254 ports 120 shapes
- Plasma vessel 80 m³
- 113 NbTi bus bars
- 2 types DC <18 kA
- 14 HTSC current leads
- 2 types DC <16 kA
- about 1000 helium pipes
- 10 central support ring elements
- 265 m² in-vessel components
- Machine height 4.5 m
- 10 central support ring elements
- Machine diameter 16 m
- About 1000 helium pipes
- Device mass 735 t
- Cryostat vessel 420 m³
- Cold mass 435 t
- Thermal insulation 3.4 K
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 m²
heat flux ≤ 10 MW/m²
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

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<tbody>
<tr>
<td>Commissioning</td>
<td>OP 1.1</td>
<td>CP 1.2a</td>
<td>OP 1.2a</td>
<td>CP 1.2b</td>
<td>OP 1.2b</td>
<td>CP 2</td>
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**Device configuration**

**OP 1.1**
- 5 limiters

**OP 1.2a/b**
- Test Divertor elements (TDU)
- Baffle elements
- Divertor closures
- Graphite wall tiles
- 2 TDU scraper elements (OP 1.2b)

**Operation parameters**

- \( P < 5 \text{ MW} \)
- \( \int P \, dt \leq 2 \text{ MJ} \)
- \( \tau_{\text{pulse}} \sim 1 \text{ s} \)

- \( P \leq 10 \text{ MW} \)
- \( \int P \, dt \leq 80 \text{ MJ} \Rightarrow 200 \text{ MJ} \)
- \( \int \tau_{\text{pulse}} \sim 10 \text{ s} \)
- (...) 60 s @ reduced power

**OP 2**
- High-Heat-Flux divertor (steady-state Water cooling)
- Port protection liners
- Cryo pumps (10 units)

- \( P_{\text{cw}} \sim 10 \text{ MW} \)
- \( P_{\text{pulse}} \sim 20 \text{ MW} (10 \text{ s}) \)
- \( P/A \leq 10 \text{ MW/m}^2 \)
- Technical limit: **30 minutes** @ 10 MW
Time line of the project Wendelstein 7-X

<table>
<thead>
<tr>
<th>Year</th>
<th>Assembly</th>
<th>Actions</th>
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<tbody>
<tr>
<td>2004-2014</td>
<td>10 years</td>
<td>Pump-down cool down magnet ramp-up flux surfaces</td>
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<tr>
<td>2014</td>
<td>1st plasmas</td>
<td>5 limiters $E_h \leq 4$ MJ $T_p \sim 1$ s</td>
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<tr>
<td>2015</td>
<td>18 months</td>
<td>Inertial cooling 10 divertors $E_h \leq 200$ MJ $T_p \sim 10-100$ s</td>
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<tr>
<td>2016</td>
<td>24 months</td>
<td>Water cooling 10 HHF divertors $E_h \leq 18000$ MJ $T_p \sim 100-1800$ s</td>
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<tr>
<td>2017-2018</td>
<td>assembly</td>
<td></td>
</tr>
<tr>
<td>2018-2020</td>
<td>assembly</td>
<td></td>
</tr>
<tr>
<td>2021</td>
<td>18 months</td>
<td></td>
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FUSION POWER ASSOCIATES 39TH ANNUAL MEETING, Washington, December 4-5, 2018
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 lengths up to 30 min possible
- full heating power
  - 10 MW ECRH, steady state
  - 10 MW NBI, for 10 s
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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

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$^{-3}$

Stray radiation < 20 kW/m$^2$

OXB mode conversion schemes under development

By courtesy of T. Stange
NBI heated discharges with high density

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

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

by courtesy of D. Hartmann
10 high resolution IR cameras

asymmetry toroidal $\leq 40 / 25\%$ (trim coils)
up/down $\sim 10-20\%$

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

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

- Detachment by pellet injection
- Before detachment
- H pellet injection into He density ramp-up to $4 \times 10^{19} \text{m}^{-2}$
- After detachment
- Time evolution of heat flux
- Power load drops to $\sim$ zero

FUSION POWER ASSOCIATES 39TH ANNUAL MEETING, Washington, December 4-5, 2018
**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

$Z_{\text{eff}}$ constant over the full discharge
A record high performance plasma

by courtesy of S. Bozhenkov

hydrogen pellet injection into helium target plasma

thermalization $T_e = T_i$

diamagnetic energy $\geq 1$ MJ

MHD event

X2 ECRH power doubled

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

by courtesy of S. Bozhenkov
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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

\[ n_i(0)T_i(0)R_E(10^{24}\text{keV}\cdot\text{m}^{-3}\cdot\text{s}) \]

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)
- $\langle \beta \rangle = 4 – 5 \%$ (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-values 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 $12 \times 1.5$ MW).