Annex 2: Overview of WPTE devices

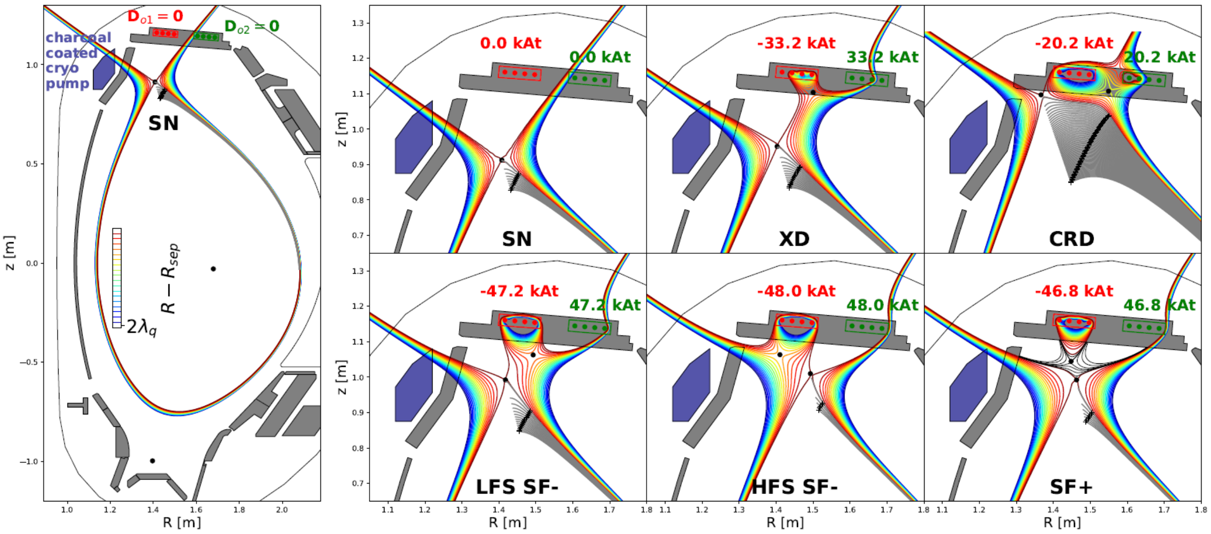
Overview of WP TE devices

In 2025, WPTE will execute experiments on the following devices: ASDEX-Upgrade, MAST Upgrade (MAST-U), TCV and WEST. A short overview of the various devices, including new capabilities for 2025, is found below, and more information is available on the WPTE wiki pages and on the each device's intranet site (credentials required). For preparatory work on current EU devices in view of JT-60SA, the applicant can consult the information available on the JT-60SA section on WPTE wiki-page (https://wiki.euro-fusion.org/wiki/WPTE_JT60SA) as well as the latest version of JT-60SA research plan (https://www.jt60sa.org/wp/wp-content/uploads/2021/02/JT-60SA_Res_Plan-5.pdf)

AUG

ASDEX (Axially Symmetric Divertor Experiment) Upgrade is a midsize tokamak operating since 1991. From 2007 onwards, the machine has been almost completely equipped with tungsten (W) plasma-facing components (PFCs). The lower divertor configuration, presently DIV III from 2014 onwards, has been optimized to enable ITER and DEMO-relevant physics investigations while upgrading of the upper divertor will be completed for the 2024 experimental campaign. For the 2014 campaign, a new divertor manipulator was installed on AUG, allowing the exposure of two full-size divertor PFCs in the low-field side (outer) strike point region during a pre-determined number of discharges. AUG also has manipulators in the outer midplane and in the typical X-point region at the outer divertor. In addition, the machine has an extensive set of in-vessel coils for inducing resonant magnetic perturbations, e.g., for suppressing ELMs.

AUG makes it possible to execute various plasma configurations from lower single null (SN) to upper SN, double null (DN), limiter plasmas, and a number of alternative divertor configurations (ADC) at the upper divertor. The relevant ADC scenarios will become gradually available during the period 2024-2025 and will range from SN to X-point divertor (XD) and from compact radiative divertor (CRD) to various snowflake (SF) varieties. Examples of the possible upper-divertor configurations can be seen in the figure below.



The lower divertor has a fixed helicity and here the directions of the plasma current and toroidal field need to be changed together. Reversing the current requires special efforts, thus typically only

1-2 weeks with reversed current and toroidal field is executed per campaign. Presently, no experimental time is foreseen for the reversed current configuration in 2025. In contrast, reversing the toroidal field can be done flexibly for upper SN discharges.

AUG has a variety of non-standard control schemes. If interested in applying such a scheme, direct contact with a member of the AUG control group is recommended. AUG has an operational shattered pellet injection (SPI) system, which will be applied under a special scheme with the ITER organization in 2025. For sophisticated scenario development, a new discharge program editor *ndpe*, visualisation tools and a flight simulator *fenix* will be deployed for all users with an IPP userid.

Technical data of AUG:

Major/minor radius (m)	1.65/0.5
Max height/width of the plasma (m)	0.8/0.5
Plasma volume (m^3)	13
Triangularity range	-0.35 → 0.6
Max plasma current (MA)	1.4
Max toroidal field (T)	3.2
Discharge duration (s)	<10
Discharge frequency (min)	20

Applied gases in AUG:

- Main plasma: D_2, H_2, He
- Other injected gases: N_2, Ne, Ar, Kr, Xe, Kr+D_2, Xe+D_2, CD_4 (on request)

Standard boronization is routinely available for conditioning purposes.

The plasma of AUG can be heated as follows:

- Ohmic heating: up to 1 MW
- Neutral beam injection (NBI) - 2 boxes, 8 sources. NBI injector II is now fully equipped with variable gap sources, allowing separate changes of power and acceleration voltage
    - ✓ 10 MW @ 60 kV
    - ✓ 10 MW @ 93 kV
- Electron cyclotron resonance heating (ECRH) - 8 gyrotrons, ECRH II + III
    - ✓ 6 MW @ 140 GHz
    - ✓ 4 MW @ 105 GHz
- Ion cyclotron resonance heating (ICRH)
    - ✓ 6 MW @ 30 or 36 MHz, additional frequencies with lower power on request

For the 2025 campaign, the main upgrades will become available in late 2024 and include besides the new upper divertor in-vessel coils for the upper bidirectional divertor, a cryopump with active charcoal coating for He pumping, a new set of upper-divertor Langmuir probes as well as divertor spectroscopy and divertor Thomson scattering for the upper divertor.

More information of the machine and the list of available diagnostics can be found on https://www.aug.ipp.mpg.de/wwwaug/

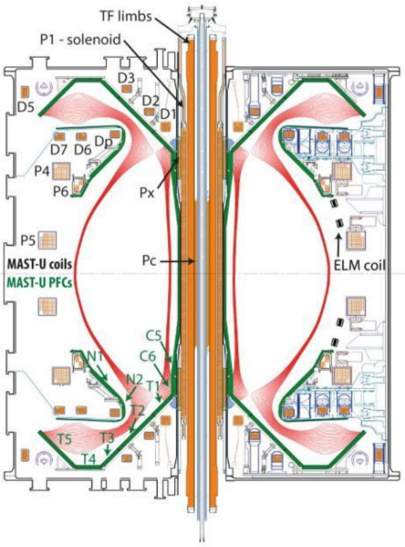
MAST-U

MAST-U is an upgrade to the MAST (Mega Ampere Spherical Tokamak) device in operation since late 2020. As the name says, MAST-U is a spherical tokamak designed for investigating a wide range of alternative divertor configurations, especially the Super-X and comparison with conventional configurations, X-divertor and snowflake configurations are also supported. The machine is equipped with carbon PFCs and features two rows of resonant magnetic perturbation coils for ELM control and error field correction. The produced perturbations have a maximum toroidal mode number up to n=2 with the upper coils and n=4 with the lower coils. Two external pairs of coils are used to provide n=1 and n=2 error field correction.

The MAST-U divertors are up-down symmetric and closed, and presently lower cryopump is expected to be in operation for the MU04 campaign in 2024-2025; the upper divertor cryopump is expected in the MU05 campaign. Inter-shot glow discharge cleaning is used between pulses to ensure strong wall pumping and regular boronizations are used to moderate the oxygen content of the plasma. There are separate toroidal chambers for realizing long-legged divertor configurations such as the Super-X. At present only operations in the forward-field configuration (the ion $Bx\nabla B$ drift points downward) and nominal plasma current direction are supported.

Technical data of MAST-U:

Major/minor radius (m)	0.7/0.5
Max elongation	2.5
Triangularity range	up to 0.6
Max plasma current (MA)	1.0
Max toroidal field (T) on axis	0.65 - 0.72
Discharge duration (s)	<2
Discharge frequency (min)	20-25



Applied gases in MAST-U:

- Main plasma: D_2
- Extrinsic impurities: He, CD_4

Standard boronization is routinely available for conditioning purposes.

During MU04 the plasma of MAST-U can be heated as follows:

- Ohmic heating: 0.2-1 MW
- NBI – 1 on-axis beam, 1 off axis beam
  - ✓ 3.5-4.2 MW @75 keV
  - ✓ Modulation frequency: up to 4 notches, beam off for minimum 20 ms; increasing the number of notches being tested during MU04, may be available on request

Control capabilities:

- Plasma current control
- Vertical position control

- Control of outer radius with data from either magnetics or linear Dα camera
- Detachment control
- Density control
- Improved equilibrium shape control of the confined plasma and divertors with real-time magnetic measurements

More information on the machine and the list of available diagnostics can be found on https://wiki.euro-fusion.org/wiki/WPTE_MAST-U

TCV

TCV (Tokamak à Configuration Variable) is a midsize tokamak featuring flexible shaping and positioning of the plasma as well as high power-density heating and current drive by ECRH (ECCD), in addition to NBI. The machine has carbon PFCs and open lower and upper divertors, but since 2019 lower divertor baffles of different sizes are available to be installed for dedicated operation periods.

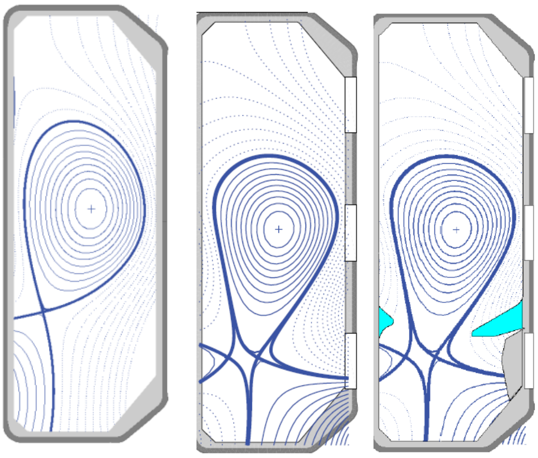
Owing to the flexibility of the machine, the triangularity (positive and negative), squareness, and elongation of the plasma can be varied over a large range and various alternative divertor configurations can be realized, including snowflake +/-, super-X, X-divertor, and doublet configurations. The directions of both the toroidal field and the plasma current can be switched in between discharges.

TCV has a variety of non-standard control schemes. The most notable of these are:

- Multi-parameter integrated control with supervisory actuator management: this can include a combination of beta, current-profile, density-profile, NTM, and sawtooth period control (possibly including realtime modeling, i.e., RAPTOR), and can count on sub-ms full Grad-Shafranov equilibrium reconstruction and on both NBI and ECRH actuators.
- Detachment control through radiation front observation.
- ECRH-assisted breakdown control.

Technical data of TCV:

Major/minor radius (m)	0.88/0.25
Max elongation	2.8
Triangularity range	-0.8 → 0.9
Max plasma current (MA)	1.0
Max toroidal field (T)	1.54
Discharge duration (s)	<4
Discharge frequency (min)	12



Applied gases in TCV:

- Main plasma: D_2, H_2, He
- Other injected gases: N_2, Ne, Ar, Kr
- 10 toroidally and poloidally distributed fueling and seeding valves, 3 GPI valves

Standard boronization is routinely available for conditioning purposes.

The plasma of TCV can be heated as follows:

- Ohmic heating: 0.2-1 MW
- NBI - 2 sources (in opposite directions i.e. 1 source is always counter-I_p)
  - ✓ NBH-1: 1.32 MW @ 25 kV
  - ✓ NBH-2: 1.12 MW @ 45 kV
- ECRH - X2 and X3 configurations (NB! not all powers are available simultaneously)
  - ✓ 2.4 MW + 0.9 MW (new in 2025) @ X2 lateral (82.7-84 GHz, 3 launchers (4 in 2025))
  - ✓ 0.9 MW @ X3 lateral (118 GHz, 2 launchers)
  - ✓ 1.8 MW + 0.9 MW (new in 2025) @ X3 top-launched (126 GHz, 2 launchers (3 in 2025))

More information on the machine and the list of available diagnostics can be found on https://wiki.euro-fusion.org/wiki/WPTE_TCV

WEST

WEST (tungsten-W Environment in Steady-state Tokamak) is a superconducting, full-W tokamak, targeted at long-pulse operation. To this end, it is equipped with superconducting toroidal field coils, active cooling loop for the PFCs, and long-pulse heating/fuelling systems. WEST allows flexible magnetic configurations to be performed, from lower or upper SN to DN, and features a large aspect ratio of 5-6.

For the phase 2 of WEST (which started in 2022), WEST is equipped with a fully actively cooled tungsten ITER-grade lower divertor, designed for 10 MW/m^2 steady-state heat exhaust and up to 20 MW/m^2 for slow transients (limited number of cycles), enabling to reach the full pulse length capability (1000 s). Long pulse operation is also possible on the actively cooled upper divertor (tungsten coating on copper components), but at a lower heat flux (typically < 8 MW/m^2).

The divertor Plasma Facing Units (PFUs) consist of tungsten mono-blocks (MBs) bonded on a CuCrZr cooling tube, with ≈ 0.5 mm toroidal gaps between blocks. In WEST phase 2, the shaping consists of a 0.5 mm height toroidal bevel as foreseen in ITER. The inner bumpers and the start-up limiter (APL) are equipped with bulk W tiles starting from the C10 campaign in October 2024. All BN-tiles used between 2020 and 2023 have been removed, thus providing a machine with a full-W environment.

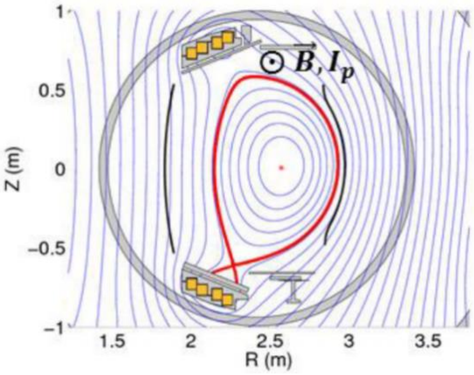
Negative triangularity configuration can be achieved by inverting the coil current direction in the upper divertor. This inversion requires one day before the experiment, and one day to reverse back again. Fixed helicity is required for the lower and upper divertor, so experiments with I_p or B_t reversal are not considered for 2025.

WEST has a versatile fuelling system (gas injection, pellet injection and supersonic molecular beam injection) and can be operated in D and He. A massive gas injection system has been implemented to study the disruption and runaway mitigation. An impurity powder dropper (IPD) with four reservoirs for real-time conditioning is also available.

The plasma control system (PCS) is targeted at handling events for long pulse operation (in particular, a Wall Monitoring System is operational for PFC safety). If interested in applying a non-standard PCS scheme, it is recommended to contact a member of the WEST team beforehand.

Technical data of WEST:

Major/minor radius (m)	2.5/0.5
Plasma volume (m^3)	15
Max elongation	1.35
Triangularity range	up to 0.5
Max plasma current (MA)	1.0
Max toroidal field (T)	3.7
Discharge duration (s)	From 10 up to 1000 s (364 s achieved)
Discharge frequency (min)	~20*



Applied gases in WEST:

- Main plasma: D_2, He
- Other injected gases: H_2, N_2, Ne, Ar, three impurity lines available simultaneously (^{3}He, Kr, Xe, possible on request)

Standard boronization is routinely available for conditioning purposes.

The plasma of WEST can be heated as follows:

- Ohmic heating: up to 1 MW
- Lower Hybrid (LH) current drive - 2 launchers
  - ✓ 6 MW (for 1000 s)
- ICRH - 2 antennas available in 2025
  - ✓ 6 MW for 30 s; 4 MW for 60 s; 2 MW for 1000 s
- ECRH and ECCD: one gyrotron in early 2025, two gyrotrons in late 2025
  - ✓ < 1 MW / 3 s in early 2025; < 2 MW / 10 s in late 2025

The LHCD and ICRH antennas can be moved radially between pulses and are equipped with local gas injection systems. The EC-wave injection angles can be varied between +25° and -25° in both the toroidal and poloidal directions.

The core Thomson scattering system will provide n_e and T_e measurements in the central plasma with up to 20 channels in 2025. The edge Thomson scattering system for pedestal n_e and T_e will become available in early 2025. The fast infrared camera (FIRCA) will become available in the second half of 2025, installed in a vertical port viewing the lower divertor.

More information on the machine and the list of available diagnostics can be found on https://wiki.euro-fusion.org/wiki/WPTE_WEST

Divertor sector for PFU testing:

The test divertor sector has 30° toroidal extension and is dedicated to technological tests or physics studies on the components. A very high spatial resolution IR system (called VHR IR) enables to monitor the surface temperature of the W MBs with a field of view of 65 mm × 50 mm (0.1 mm / pixel) moveable on an observable area covering a grid of 17 PFUs by 20 MBs, with 20 Hz sampling rate. The VHR IR camera position can be changed between pulses to investigate different areas of the test divertor sector and therefore different heat loading conditions with respect to the toroidal ripple field effect.

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* Longer if long pulses (>60 s) are performed.

Three PFUs have been taken out from the production batch and exposed in a HHF test facility (e-beam gun) to generate well-controlled and different type of damages:

- Pre-damaged PFU#1 was exposed to C3-C4 experimental campaigns. This PFU has been removed from WEST and is now used for post mortem analysis.
- Pre-damaged PFU#2 (featuring crack network on MB25) was exposed to C6-C7 (PFU location #7) and C8-C9 (PFU location #8) WEST experimental campaigns. In C10-C11, pre-damaged PFU#2 will be placed in PFU location #7 in the maximum heat flux area on the low field side.
- Pre-damaged PFU#3 (self-castellation on MB27 and MB31) was exposed to the C8-C9 campaigns (PFU location #9). In C10-C11, pre-damaged PFU#3 will stay at the same position (PFU location #9) with the maximum heat flux on the low field side.

Two PFUs are dedicated to technological tests, one in PFU location #6 and one in PFU location #19 to reach maximum heat load on the low-field and high-field sides, respectively. Four PFUs have been chosen to assess net erosion and deposition over the campaigns (PFU locations #20, #21, #22 and #32). No surface cleaning or change will be applied during shutdowns, in order to preserve the erosion and deposit history along WEST phase 2. In contrast, a test of cleaning the divertor using a laser was performed during the 2024 shutdown on PFU locations #23 - #31 and #33 - #38 (see figure below).

Vertical and poloidal misalignments are deliberately introduced to investigate heat load in different configurations. A first series of six consecutive PFUs were installed for the C6-C7 campaigns with misaligned MBs in the poloidal direction (1.5 and 3 mm shifts) to generate amplified thermal signatures of the Optical Hot Spot (OHS) on the 45° inclination chamfer of the adjacent downstream MB. They are situated in PFU locations #10 - #15. This will be kept for C10-C11.