# THE STATUS AND DESIGN CHALLENGES OF THE HEATING AND CURRENT DRIVE SYSTEMS FOR DTT

G.GRANUCCI[1,2], S.CECCUZZI[3,2], A.MURARI[4,2], A.ROMANO[3,2], P.AGOSTINETTI[4,5], A.BRUSCHI[1], A.CIOFFI[2], F.FANALE[1], A.FERRO[4], S.GARAVAGLIA[1], A.MORO[1], A.PEPATO[6], G.L. RAVERA[3], M.RECCHIA[4,5], N.RISPOLI[1], C.SALVIA[7], A.A. TUCCILLO[8], AND THE DTT CONTRIBUTORS

[1] Institute for Plasma Science and Technology, National Research Council (ISTP-CNR), Milano, Italy
[2] DTT S.C.a r.l., Frascati, Italy
[3] ENEA, Fusion and Nuclear Safety Department, C.R. Frascati, Frascati, Italy
[4] Consortium RFX, Padova, Italy
[5]Institute for Plasma Science and Technology, National Research Council (ISTP-CNR), Padova, Italy
[6]National Institute for Nuclear Physics (INFN), Padova, Italy
[7]University of Padua, Padova, Italy
[8] CREATE, Napoli, Italy

Email: Gustavo.Granucci@DTT-Project.it

To achieve the goal of the Divertor Tokamak Test facility (DTT) [1] powerful Heating and Current Drive (HCD) systems have been designed in order to load the divertor at the necessary relevant level (45 MW) to perform power exhaust studies in the reactor relevant conditions typical of ITER and DEMO. Based on a well consolidated knowledge of physical interaction with plasma, the heating systems chosen for DTT are Electron Cyclotron (ECH - 32 MW at 170GHz), Ion Cyclotron (ICH - 8 MW at 60-90 MHz) and Negative Neutral Beam Injection (NBI - 10 MW at 510 keV). The DTT design is characterized by a large flexibility in the magnetic configurations, with the capability of investigating Single Null, X-Divertor, Double Null and Negative Triangularity, to allow the assessment of different divertor solutions. A large flexibility it is therefore also one of the requirements of the design of the DTT plasma heating systems, to properly sustain the needed performances in all the operational conditions foreseen in DTT. The main DTT parameters are: major radius R=2.19 m, minor a=0.70 m, toroidal magnetic field $B_T$=5.85 T and plasma current up to $I_p$=5.5 MA for a whole pulse length of 100 s. DTT is a complex experiment and the integration challenges are of great importance in the design and development of the heating systems, providing, in turn, specific constraints to other plant auxiliaries like the water cooling system, the power grid and the buildings for the HCD systems.

The HCD power will be installed in 3 consecutive steps, in accordance with the DTT exploitation program: a first step with the installation of 16 MW of ECH and 4 MW of ICH, a second step with the addition of the 10 MW of NBI and the last one with the completion of the radio-frequency RF systems with further 16 MW for ECH and 4 MW for ICH. The duration of each step is presently foreseen to be approximately 5 years.

The relatively small dimension and the high energy stored in the DTT plasma are a severe challenge for the heating systems facing plasma with antennas for the mechanical constrains due to the thermal loads and to the forces induced by disruptions. For these subsystems a careful design verification, construction specification and commissioning test have been provided.

The ECH system [2] is foreseen to play the major role in DTT performing several functional tasks for plasma heating and control. The system is organized in clusters, each of them composed of 8 gyrotrons (1 MW, 170 GHz) connected to a single quasi optical multibeam transmission line, at the end of which the 8 beams are split and injected separately in the plasma through a launcher, steerable in the two directions. In the present first DTT phase, two clusters will be realized and installed for a total power of 16 MW. The procurement of the first 16 sources is started and the gyrotrons are under manufacturing by THALES. A first pre-series unit has been tested up to the required performances. Two gyrotrons are fed in parallel by a single High Voltage Power Supply set, for which the design has been completed (similar to those done for ITER) and the technical specifications have been defined. The conceptual design of the quasi-optical transmission line has been completed and a prototyping activity of mirrors has been started, exploiting the additive manufacturing technique to realize optimized cooling circuits capable to sustain and manage the highly localized thermal load minimizing also the surface deformations. An *evacuated connection line* to direct the gyrotron output to the RF load, designed to host different mirrors, has been realized to be used in the FALCON test facility at EPFL, with the DTT pre-series gyrotron, to validate the mirrors design and the manufacturing techniques. The antenna has been designed assuring the maximum flexibility in term of poloidal and toroidal steering angle of the launching mirrors. The challenge in designing the launcher system is to face the paramount forces induced by major disruptions of DTT that are larger than in all the existing tokamak equipped with movable mirrors close to the plasma.

The ICH system [3] is based on 4 RF generators (transmitters) for each module, realized with Solid-State Power Amplifiers connected, by a transmission line and matching units, to two movable 3-straps antennas designed in order to guarantee efficient coupling in all the foreseen magnetic configuration of DTT. The procurement of two of the four transmitters of the first module of ICH system is ongoing and is expected by 2026. Together with the transmitter a batch of RF components, needed to realize a test-bed for the transmitter test, has been procured with components, as the 2.5 MW RF load, to be used also in the final installation. The ICH system is designed to operate in the range 60-90 MHz as requested by the research program of DTT, allowing minority heating schemes in Hydrogen or in $^{3}$He.  Each ICH module is expected to couple 3 MW to the DTT reference plasma scenario for 50 s every hour while during wall conditioning operations, a larger duty cycle of 1 hour every 2 hours with a maximum coupled power of 200 kW was instead assumed. Each transmitter will consist of 128 RF modules that have been conceived with the robust design reliability used in broadcasting transmitters to sustain the reflection issue typical of plasma ICH coupling. A gradual, multi-stage, combination strategy has been designed, optimizing the performance at the edges of the frequency range, where the transmitters are mostly expected to operate. The transmitter outputs are combined in pairs and, after around 60 m of coaxial line, again split by 3 dB hybrid couplers and routed toward different antennas to feed equivalent straps. Each 3 straps antenna has 4 feeds, two for the central strap and the other two are for lateral ones. Taking into account all geometrical constraints, the antenna design was optimised in terms of RF performance reaching a coupling capability in single null scenario well above the target value of 1.5 MW. To keep at a reliable level the power density the antenna requires a space larger than the port area, forcing to install it from inside by remote handling. Therefore, the current design is based on a “semi plug-in” concept, where straps and most part of antenna box are installed through the port whereas the remaining components like Faraday screen and limiters will be assembled by the Remote Handling System.

The DTT NBI system [4] will be installed for the second phase of the scientific program. The conceptual design has been completed. Starting from the prototypes for ITER, SPIDER and MITICA, some specific optimization has been done in order to adapt the design to the DTT requirements. The ion source has been based on the MITICA one, while the DTT accelerator exploits the design flexibility allowed by the additive manufacturing. This choice will assure a higher efficiency thank to an innovative accelerating grid design, verified with the realization of a full-scale prototype, that allows a better beam optics focalization, reducing the spread of the beamlets and therefore the total losses. The DTT NBI is even more demanding than the ITER system, due the high magnetic field near the tokamak and its quite high power density. Magnetic shielding has to be provided to avoid stray magnetic field along the beam-line which would cause a dramatic loss of beam efficiency. In turn, the NBI shielding must not introduce error fields in the plasma region. For the High Voltage power supply an alternative solution based on Multilevel Modular Converter has been explored, in order to enhance the efficiency and to exploit a more industrialised solution for a longer term sustainability.

## REFERENCES

[1] F. ROMANELLI, Nucl. Fusion **64** (2024) 112015

[2] G. GARAVAGLIA et al., **Development of the electron cyclotron resonance heating system for Divertor Tokamak Test** J. of Vacuum Science & Technology B (2023) https://doi.org/10.1116/6.0002396

[3] S. CECCUZZI - et al, **Progress in the development of the ICRF system of DTT**, Fusion Engineering and Design 213 (2025), 114849 doi:10.1016/j.fusengdes.2025.114849.

[4] P. AGOSTINETTI et al. **Improved Conceptual Design of the Beamline for the DTT Neutral Beam Injector** - IEEE Transactions on Plasma Science, vol. 50, no. 11, pp. 4027-4032, Nov. 2022, doi: 10.1109/TPS.2022.3162902.