EUROFUSION WPDTT2-PR(16) 16233

G Granucci et al.

The DTT device: system for heating

Preprint of Paper to be submitted for publication in
Fusion Engineering and Design

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.

The DTT device: system for heating

G.Granucci[a], S.Ceccuzzi[b], G.Giruzzi[c], P.Sonato[d], P.Agostinetti[d], T.Bolzonella[d], A.Bruschi[a], A.Cardinali[b], L.Figini[a], S.Garavaglia[a], R.Maggiora[e], D.Milanesio[e], F.Mirizzi[f], S.Nowak[a], G.L.Ravera[b], C. Sozzi[a], A. A. Tuccillo[b], P. Vincenzi[d]

[a]*Istituto di Fisica del Plasma- CNR,via R.Cozzi 53- Milano, Italy*
[b]*ENEA, Dipartimento FSN, C.R. Frascati, via E. Fermi 4,5 00044 Frascati (Roma) Italy*
[c]*CEA, IRFM, F-13108 Saint-Paul-lez-Durance, France*
[d]*Consorzio RFX, I-35127 Padova, Italy*
[e]*Politecnico di Torino, Dipartimento di Elettronica, Torino, Italy*
[f]*Consorzio CREATE, via Claudio 21, I-80125 Napoli, Italy*

The proposed Divertor Test Tokamak, DTT, aims at studying power exhaust and divertor load in an integrated plasma scenario. Additional heating systems have the task to provide heating to reach a reactor relevant power flow in the SOL and guarantee the necessary P_{SEP}/R together adequate plasma performances. About 40 MW of heating power are foreseen to have $P_{SEP}/R \geq 15$ MW/m. A mix of the three heating systems has been chosen, assuring the necessary flexibility in scenario development. An ECRH system at 170 GHz will provide 10 MW at plasma for several tasks, such as: bulk electron heating, localized CD, avoidance of impurity accumulation and MHD control. In addition 15MW of ICRH (in the range 60-90MHz) will provide the remaining bulk plasma heating power, on both electrons and ions. ICRH, in minority scheme, will produce fast ions, allowing the study of fast particle driven instabilities like alphas in D-T burning plasmas. The heating schemes foreseen in DTT are 3He and H minority as well as Deuterium 2nd harmonic. The addition of 15 MW of NBI, later in the project, could provide a mainly isotropic parallel fast ion distribution to simulate the alpha heating scheme of a reactor. The NBI primary aim is to support plasma heating during the flat top phase when the need of central power deposition and the minimization of the shine-through risk suggest a beam energy around 300 keV. In the first phase of the DTT project the available power will be at least 25 MW, to be increased during the lifetime of the machine.

Keywords: ECRH, ICRH, NBI

1- Introduction

The Divertor Test Tokamak (DTT) is a device designed to study the power exhaust and the heat load in alternative divertor configurations, in order to contribute to the development of a viable solution for the demonstration fusion reactor (DEMO). In order to perform such studies, DTT requires a powerful heating system, with the main target to simulate the power flow P_{SEP} (the power crossing the separatrix) in the Scrape Off Layer (SOL) of a reactor, where the nuclear reactions provide the heating source. Long pulse operation (~ 100s) is also a basic requirement for a meaningful test of the actively cooled Plasma Facing Component (PFC) behaviour. Moreover, systematic tests require high machine and sub-systems reliability, which in particular applies to the heating system. Since any possible solution of the power exhaust issues must be compatible with high performance bulk plasma and reactor relevant scenarios, additional heating system must guarantee the necessary P_{SEP}/R (R being the tokamak major radius) and reliable plasma heating capable to simulate as much as possible the reactor situation. A good balance between ion and electron energy content is required and the presence of a significant fraction of fast ions is highly desirable.
In its present reference design [1], DTT will be equipped with superconducting magnetic field coils, which would allow steady-state operation. However, steady state (i.e., fully non-inductive plasma regime) is not in the objectives of DTT, and long pulse (~100 s) achievable with inductive CD will be sufficient. Taking into account the main machine parameters ($R = 2.15$ m, $n_e \sim 2\ 10^{20}$, $I_p = 6$ MA), full non-inductive current drive (CD) would require an additional power > 100 MW, by far exceeding that required for heating. This implies that bulk CD will not be one of the main objectives of the system, although useful to assist the transformer and possibly to tailor the current density profile. On the other hand, localized CD could be crucial to control MHD phenomena, i.e., the large sawteeth that are expected in the reference DTT scenarios and classical or neoclassical tearing modes.
As discussed in [1], a heating power in excess of 40 MW is necessary to provide the reactor relevant figure of $P_{SEP}/R \sim 15$ MW/m. In order to guarantee a robust an reliable heating (mainly heating the electrons, as the alpha particles do in a burning plasma), it seems effective to choose a mix of the three heating systems presently proposed also for ITER. This could ensure a wide flexibility in scenario development and the necessary availability of power.
The main requirements for the DTT heating and current drive system have been determined on one side from the machine rationale and objectives, on the other on the basis of the first scenario simulations. Concerning the choice of the heating and CD system, the relative merits of the various systems used on tokamaks are well known: Lower Hybrid (LH) and Neutral Beam Injection (NBI) have higher CD efficiencies; Electron Cyclotron (EC) waves have better localization capabilities, both for

author's email: please visit http://www.ifp.cnr.it/staff.html

heating and CD; Ion Cyclotron (IC) waves are the cheapest system, mainly used for heating, both electrons and ions. In order to make a choice, the following rationale has been employed:

- In order to limit the complexity of the plant, it has been chosen to use not more than two systems, at least in the first phase [1].
- Since bulk CD is not a main objective, the high CD efficiency of LH and NBI is not considered as a decisive advantage.
- The need to heat both ions and electrons favours the use of IC waves.
- The usefulness of MHD control, requiring localized CD, favours the use of EC waves
- EC waves are also highly suitable for the control of W accumulation

As a consequence, it was chosen to use as a reference, for the first DTT phase, a system composed by 15 MW of ICRH and 10 MW of ECRH. For both the systems an upgrade to 20+20 MW is taken in account in the design, while the addition of NBI is also being considered as another possible candidate for a further power upgrade.

A system of installed 12MW of ECRH at 170 GHz should be able to provide 10 MW absorbed in the plasma for several tasks, such as:

- Bulk electron heating to bring the plasma in the high confinement regime.
- Current profile tailoring during plasma ramp-up by moderate but localized CD
- Pulse length extension by assisting current ramp-up reducing the transformer flux consumption
- Control of central impurity accumulation, avoiding unstable hollow temperature profiles
- Sawtooth control by localized CD, in order to avoid NTM)(Neoclassical Tearing Mode) seeding
- NTM control to assure plasma stability in high beta scenarios
- Plasma termination control, avoiding disruption during the current ramp-down phase.

In order to fulfill these requirements, the design of the system should be flexible enough to guarantee the power absorption, its localization and control. Nevertheless, being the scope of DTT mainly devoted to power exhaust studies, only a partial flexibility in EC beam steering during the pulse is foreseen, letting the possibility to change the deposition location and CD shot by shot.

Together with the EC system, since the beginning 15MW of ICRH (at a frequency around 60-90MHz) can provide the remaining bulk plasma heating, on both electrons and ions. ICRH is also a useful tool for heating scheme acting like the alpha particles, but with an isotropic perpendicular fast ion distribution. The most suitable schemes based on ICRH are ^{3}He minority heating, H minority heating and Deuterium 2nd harmonic. ^{3}He minority is preferable for bulk heating, whereas H minority may mostly heat the tails of the hydrogen distribution function. Minority concentrations in the range 2% – 10% are adequate for such operations; higher concentrations would incur significant mode conversion at the hybrid two-ions resonance. The operational schemes do not dictate particular requirements on launched power spectra, which can be symmetrical and broad.

Eventually 15MW of NBI could provide a mainly isotropic parallel fast ion distribution that, together with the ICRH could allow simulating the alpha heating scheme of a reactor. NBI Heating primary aim is to reliably support central plasma heating during the main phase of the plasma confinement. On the other hand, NBI system parameters should also minimize the risk of shine through and for this reason its use during early current ramp-up and late current ramp-down should be carefully evaluated. The necessity of central power deposition and the minimization of the shine through risk suggest beam energy around 300keV. The power will be absorbed both by electrons and by ions. In addition to plasma heating, NBI can support plasma current sustainment; for this purpose a dedicated port will allow a tangential (45° on the plasma axis) injection.

2. ICRH Heating

The Ion-Cyclotron Resonance Heating (ICRH) system of DTT is primarily aimed at coupling a power of 15 MW for central plasma heating, despite other employments, like current drive, assisted start-up, suppression of MHD instabilities, wall conditioning, density peaking control, transport study of fast ions or generation of toroidal plasma rotation [2] are not excluded. Its conceptual design has been tailored to fit the first aim, setting the highest working frequency for operations with the reference H-mode, B_0 = 6T scenario. The H minority cyclotron damping appears the most suitable mechanism to this aim, leading to a maximum resonant frequency of 90 MHz. With acceptable impact on system design and complexity, the lowest frequency has been set at 60MHz, allowing central heating in D(H) plasmas down to 4T as well as in D(^{3}He) plasmas at 6T, which is an alternative useful heating scheme. The main parameters of the ICRH system of DTT are summarized in Table 1.

Table 1: Avoid the use of vertical lines within the table; use horizontal lines to separate headings from table entries

Frequency range [MHz]	60 - 90
Bandwidth at -1 dB [MHz]	±2
Pulse width [s]	100
Number of antennas	4
Peak couple power [MW]	≥ 15

2.1 System description

The design of ICRH system benefits from a mature and relatively cheap technology validated over many multi-megawatts installations. Regarding wave generators, final power amplifiers (FPAs) have achieved important results in past years and now ITER is pushing forward with further progress [3]. According to predictions and extrapolations of power derating at higher frequency, a delivered power per tube of 1.3 MW at 90 MHz with 100 s pulse duration, 12.5% duty cycle, VSWR < 1.2, is achievable, leading to an available power in excess of 20 MW with 16 generators. The tetrodes TH526 by Thales Electron Devices and 4CM2500KG by Communications & Power Industries are among the main candidates. Each transmitter also includes a solid-state amplifier and a tetrode driver, acting as preparatory stages for the FPA. A low-power frequency source provides the RF signals to a group of four transmitters with real-time control of amplitude and phase for both modulation and safety purposes. Optimal working conditions of driver and FPA cavities can be adjusted through motor tuners without changing the internal components.

Power supplies are available from relevant vendors; crowbarless pulse step modulator, equipped with modern IGBT switching devices, can be assumed for the high voltage power supply. Considering the FPA full performance below 80 MHz, i.e., around 2 MW for 30s, summing up the requirements of the other units and taking into account quoted efficiencies; the requested DC power for the overall ICRH system is about 48 MW.

As far as the transmission line (TL) is concerned, a matching and tuning scheme based on one T-junction and two real-time controlled phase shifters (PhS) per generator has been preferred, as depicted in Fig.1, in order to achieve good resilience to plasma load variations at relatively modest cost [4]. Two rigid coaxial cables are considered: the standard 9" with a characteristic impedance Z0 = 50Ω and a proprietary solution with Z0 = 30Ω, respectively giving a maximum peak voltage of 50 and 42 kV with dry air (safety factor = 2). The former is used up to the T-junction, whose input impedance is matched to 50Ω with a PhS and a stub (PSt). An additional service stub (SSt) provides access to the cooling circuit through the inner conductor of the coaxial cable. A double conical alumina feedthrough separates the pressurized and vacuum TL sections.

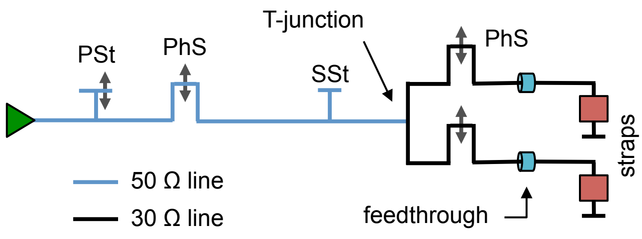


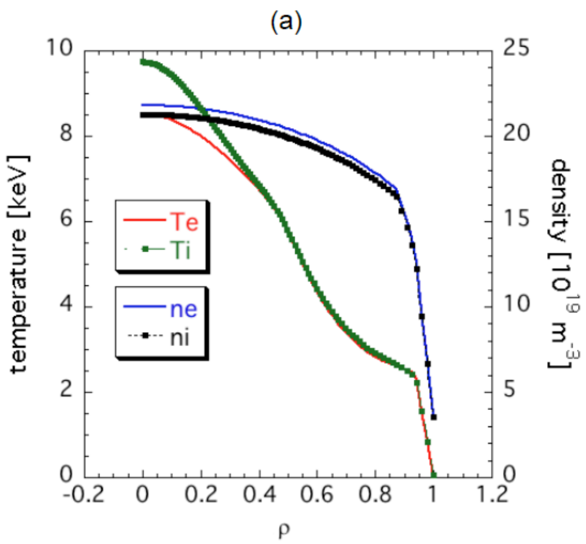
Fig.1 TL schematic layout, based on the external conjugate-T matching scheme

To meet DTT port size (0.65m x 1.098m) and maximize coupled power, an antenna made of 4 poloidal by 2 toroidal straps has been conceived; such arrangement is suitable for plasma heating operations and minimizes the mutual coupling between straps. A total of four port-plugged antennas of this kind are foreseen, with contacts grounded to the port and allowing for shimming of the antenna front face.

The system will be equipped with a comprehensive set of diagnostics such as optical detectors, directional couplers, shunts, etc. so as to monitor operational conditions and acquire experiment-relevant data. Reflectometres will be installed to accurately measure density profiles [5].

2.2 Performance prediction

The launcher has been preliminarily optimized with CST Microwave Studio using a dielectric-loaded flat antenna in the frequency range 60-90 MHz. A curved model of the final geometry has been also simulated with TOPICA [6] at 60 and 90 MHz in presence of a plasma load with the profiles of Fig.2(a). From the last closed magnetic surface, exponential profiles with decay length of 4.5 mm have been assumed. Results are shown in Fig.2(b) for a single antenna using dipole phasing and matched coaxial feeders with a characteristic impedance of 13.4 Ω. With the antenna at 5 cm from the plasma separatrix, the requirement of 15 MW of coupled power at 90 MHz is fulfilled with four launchers; more internal positions of antennas allow a significant enhancement of coupling performance, which looks rather constant with frequency.



§Fig.2 (a) Plasma profiles vs. normalized radius. (b) Coupled power for 0π phasing simulated by CST with a seawater load and by TOPICA with the antenna at 5 (plasma1) and 2 (plasma2) cm from the separatrix of the DTT reference plasma.

By using 6 T reference equilibrium and kinetic profiles, ICRH wave propagation and quasi-linear absorption has been simulated with the 2D full-wave code TORIC [7] for Deuterium plasmas with 3% of H minority at 90 MHz and 3He minority at 60 MHz considering two reasonable power spectra, respectively peaked at $N_{\parallel}$=7 and 10. In the former case the predicted absorption is distributed as follows: 27% to D, 49% to H, 24% to free electrons via electron Landau damping, while in the latter case as 2% to D, 54% to 3He, 44% to free electrons, consisting of 31% via electron Landau damping and 13% via mode conversion into ion Bernstein wave.

3. ECRH Heating

3.1 Physics requirements

The Electron Cyclotron Resonance Heating (ECRH) system of DTT is in charge of three main tasks: plasma heating, MHD control (sawteeth, tearing modes and NTM) and additional current drive generation for transformer assistance or current profile tailoring. The power required for plasma heating is 10-20MW, determined by scenario simulations; 10MW has been assumed as a reference, with the possibility of a successive upgrade to 20MW. Wave injection at various poloidal and toroidal angles is considered in order to assure EC power absorption for a broad range of magnetic field up to 6T. Taking into account the nominal magnetic field value (6T) and to minimize the risk and the cost of wave source developments, the frequency of 170 GHz has been chosen for the system design. Presence of equatorial and upper launcher is also considered.

3.2 Wave frequency and injection parameters

An assessment of the EC wave absorption and CD capabilities has been performed with the beam-tracing code GRAY [8] on scenario obtained by the METIS code. The calculated current I_{cd} driven from the upper port (dedicate to MHD control) is reported in Fig.3. Using the upper launcher, current can be driven in the range $0.3 \leq \rho \leq 0.9$, with an efficiency ranging from 9kA/MW to 2.2kA/MW depending on deposition radius. From the equatorial launcher the maximum current driven in the core at $\rho=0.3$ is I_{cd}=16kA/MW.

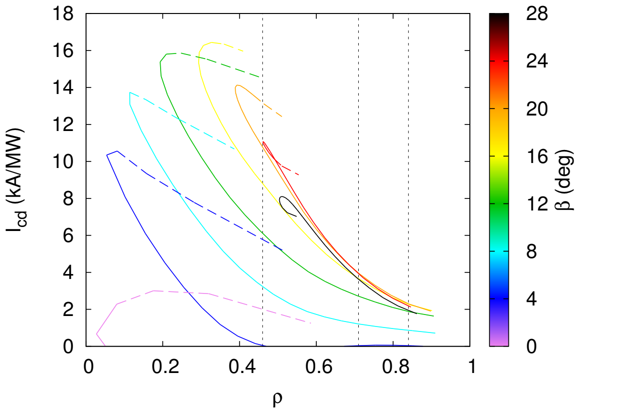


Fig.3 Current I_{cd} driven from the upper port, varying the poloidal angle for some discrete toroidal injection angles $0° \leq \beta \leq 28°$. The radial location ρ of the 3/2 and 2/1 NTM islands (dashed vertical lines) can be reached by aiming the beam both at the upper (solid) and at the lower (dashed) half of the plasma, with different efficiency.

First analysis of NTM stabilization for the 2/1 and 3/2 modes have been carried out injecting wave from the Upper Port. The NTM evolution has been calculated using a Modified Rutherford Equation, at this first stage neglecting the mode rotation effects [9]. As shown in Fig.4, 4 MW of EC power can stabilize the 2/1 mode with a current deposition of 5cm, provided that the current efficiency I_{cd} =2.5 kA/MW (as seems feasible for some launch directions, looking to Fig.3). For the 3/2 mode, stabilization is obtained even more easily.

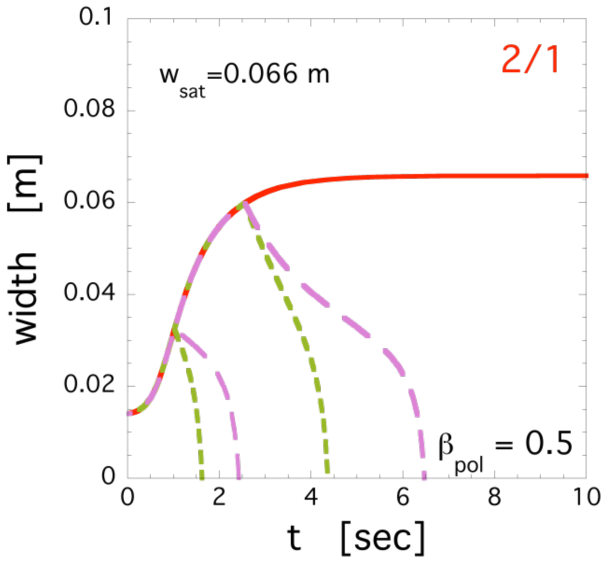


Fig.4 Time evolution of the 2/1 island width from the Generalized Rutherford Equation. The green dashed lines refer to w_{CD}=4cm and the violet to w_{CD}=5cm.

3.3 Wave generator

The power of EC system will be installed in time with a first step of 12 MW up to 20MW at plasma at the final stage of DTT. The conceptual design is based on a total number of 24 gyrotrons (1MW/170GHz/100s) for the complete system (12 tubes for the first step), assuming an average power loss of 10% in the transmission line. The gyrotron will require two power supplies, one for the cathode (55kV/50A) and one for the anode (35KV, 0.1A) and will be based on the depressed collector technology (with an efficiency of 50%). The overall coolant flow rate required is 24 l/s. The distance between adjacent tubes (4.25m) is determined by the maximum allowed magnetic stray field at the cathode region, i.e., 0.5mT.

3.4 Transmission lines and launchers

At the present status of layout and buildings distribution, we consider a TL average length of 120m to transmit the EC power from the sources to the launchers. Since no nuclear safety requirements are to be considered for DTT (no tritium), the design for the TL is open: it could be the ITER like solution, i.e., evacuated waveguides (EWG), or a quasi optical (QO) solution as in the W7-X stellarator. The first one makes use of evacuated aluminum corrugated waveguides (i.d. 63.5mm). The expected EWG losses are of 10%, which means 100kW in 100s to be removed by a coolant flow rate of 1.1 l/s per line. The QO scheme is based on the use of large multi-beam mirrors exploiting the quasi-optical propagation, as on W7-X [10] with up to 7 beams that can be allocated on one multi-beam mirror. Given the diameter of the mirrors (50cm), the distance between mirrors (5m) and their total number (16) has been chosen. An extra option for an Evacuated QO line is also considered. The calculated losses in both cases are around 11%.

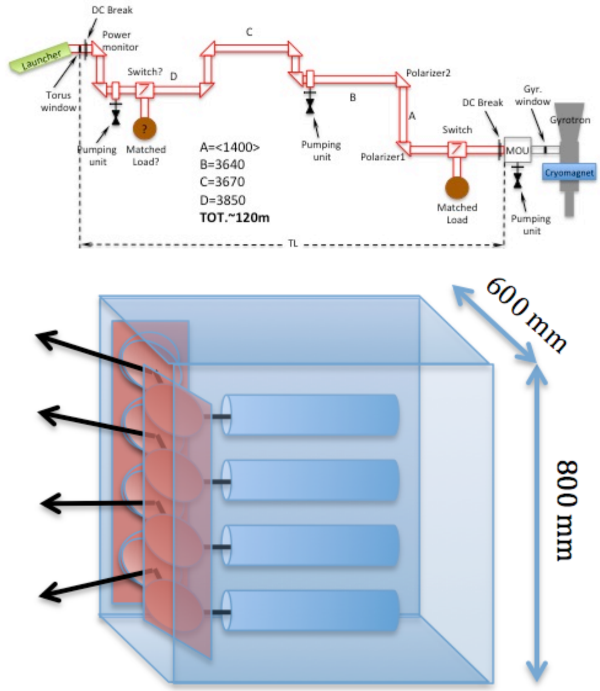


Fig. 5: (Top) Generic EWG TL routing from the Gyrotron Hall to the tokamak. (Bottom) Sketch of the equatorial launcher with the poloidal steering option, accommodating up to 4 beams.

The lines will be connected to the 4 launchers (fed by 4 beams each), three on equatorial ports and one in an upper port, to control MHD activity. At the first stage of installation, for 10MW of power to the plasma, only 2 equatorial launchers will be installed, fed by 4 lines, for a total of 8 launching mirrors (plus the 4 in the upper port).

Two different kinds of launching systems are foreseen: one Real Time (RT) launcher for MHD control and four simplified equatorials launchers, both will be designed to be fed by 4 gyrotrons. The equatorial ones will be steerable only in one direction (toroidal or poloidal), shot by shot. For these launchers, located in plug-in structures to be easily removed and maintained, two different layouts are considered, depending on the preferred steering plane: for poloidal one the beams are in a column of four (see Fig.5 bottom), while for the toroidal steering they are in a square arrangement. The port fraction occupied by the optics is estimated from 50% to 100%, depending on the needed launching angles. The Upper launcher is the antenna for NTM and sawtooth control with several features as a poloidal steering range capable to cover from q=1 to q=2 surface, a composite mirror technology with peak thermal load of ~100 W/cm^2 (on the copper coating) and a water flow per mirror of 60 l/min total. The toroidal steering is included but not in Real Time.

3.5 Power supplies Machine layout and interfaces

The gyrotrons will be fed by Solid State High Voltage Power Supply (SSHVPS) in a configuration as simple as possible. Each gyrotron requires two PSs, one for the main high voltage power supply (MHVPS, -55kV, 50A), connected between ground and the gyrotron cathode with a relatively low stability (1%) and a second stage called Body Power Supply for gyrotron anode (BPS, +35 KV, 0.1A) with a better control of the output voltage (~0.5%). Considering that the main task of the EC system is the bulk heating to sustain the power load on the divertor, a time-detailed control of the main EC power is not required, therefore each MHVPS will feed up to 4 tubes simultaneously, while the BPS will remain singular. The four tubes dedicated to MHD control could be fed in pairs, if necessary.

The EC power will be delivered to the plasma using 2 different types of launcher, as previously described. The port occupation by the EC plug-in launcher is partial for the equatorial one, leaving space for other systems (cooling pipes) or diagnostics insertion, while it is full for the upper launcher. The plug-in concept will be at the basis of the design, allowing also remote maintenance. The diamond windows will be on the port plate and removed with the launcher during the maintenance periods. In order to simplify installation and layout, all the equatorial launchers will be identical and located in a toroidally symmetric position on the machine to duplicate also the interface component between the TL and the launchers. In each TL and after each window a gate valve will be mounted in order to operate the machine in case of failure.

The architecture of EC Control system will take benefit from what has been under development for ITER and for JET. The system will be divided in sub-units and based on well-developed and consolidated technology avoiding custom solutions that are not generally supported in time.

4. NBI heating

4.1 Physics requirements

Neutral Beam Injection (NBI) Heating primary aim is to reliably support central plasma heating during the main phase of the plasma confinement. On the other hand, NBI system parameters should also minimize the risk of shine-through and for this reason its use during early current ramp-up and late current ramp-down should be carefully evaluated.

A NBI system based on the acceleration of positive ions would inject into the plasma particles with energies of the order of 80-120 keV. This energy however, given the density profile estimated for the DTT reference scenario, would lead to a too peripheral power deposition. In order to heat particles in the region of good confinement, i.e. in the core plasma, a negative NBI system working efficiently at higher energies is then proposed.

The power required for this main function has been estimated in the order of 15 MW. The trade-off between

the necessity of central power deposition and the minimization of the shine-through risk, given the DTT device geometry, suggest beam energy of 300 keV. At this energy level the power will be absorbed both by electrons and by ions. In addition to plasma heating, NBI can support plasma current generation, provided that an optimized (tangential) injection geometry is possible.
Two injectors each delivering approximately 7-8 MW are proposed.

4.2 Beam particle energy and injection parameters

A METIS simulation has been performed to model a DTT scenario at full power (40 MW) heated only by two 300 keV perpendicular NNBI systems, 20 MW power each. The power deposition waveform and shine-through fraction are shown in Fig.6.

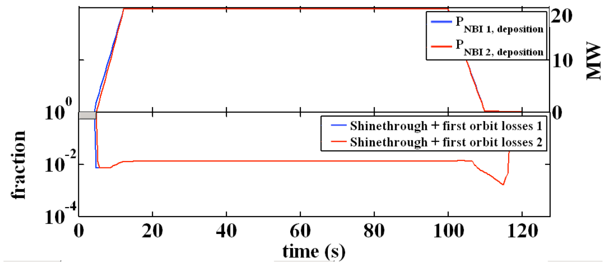


Fig.6 NBI power deposition waveform and shine-through fraction.

For the same system an acceptable shine-through has been confirmed by the METIS code. The choice of 300 keV beam energy would also then leave a suitable operational margin in case lower plasma densities are required during some DTT operations.
As the main aim of this system is plasma heating, perpendicular injection of the beam is also a valid option. This solution minimizes the impact of the NBI system on the overall design. If some current drive will be considered as an optional asset of the system, sufficient tangential injection geometries could be considered.
In DTT one tangential port is provided for the installation of one injector, whereas the other will have a perpendicular access to the plasma.

4.3 Design choices and concepts

Taking into account the previous simulation and assuming a conservative approach by the adoption of the gas neutralizer, in which the negative ion neutralization yield is theoretically limited to 60%, and also taking into account additional losses, it is necessary to accelerate at 300kV approximately 45A of negative ion current. This corresponds to extract approximately 50A from the plasma source in a beamlet configuration of a matrix of 20 (4x5) beamlet groups each group having 70 (5x14) beamlets.
A layout scheme similar to those adopted in JT-60SA [11] [12] and in LHD [13] is proposed, in which four different sections can be identified: beam source (including negative ion source and accelerator), neutralization section, residual ion dump (including beam dump) section and duct to connect the injector to the vacuum vessel.

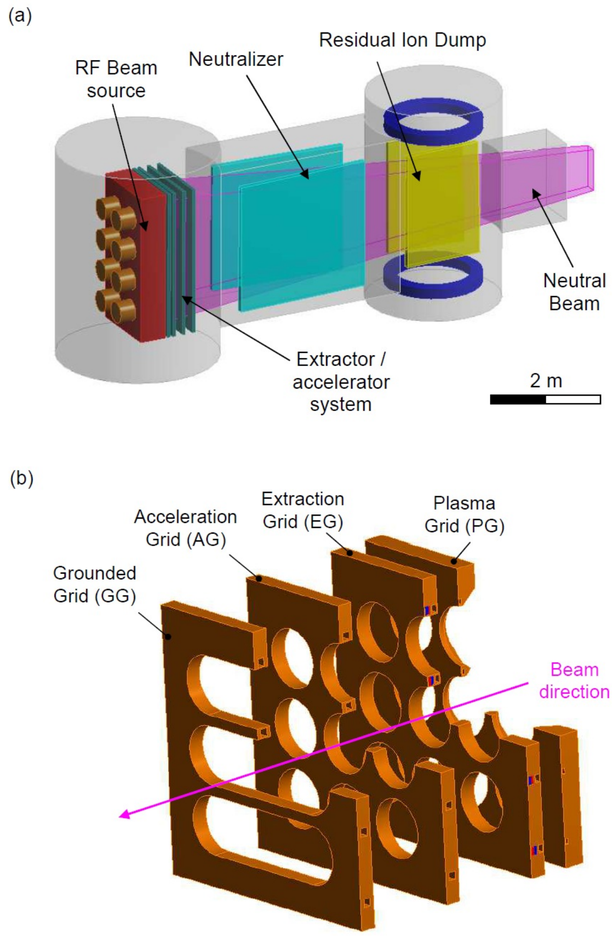


Fig.7. Neutral beam injector for DTT: (a) sketch of the main components; (b) acceleration stages concept.

For the plasma source, it is proposed to use the same RF source concept adopted for ITER, instead of the filament arcs concept adopted in both Japanese NNBI systems. The plasma grid will be coated with Caesium in order to enhance the surface negative ion production that will be released by commercial Caesium dispensers, accordingly to the development under progress for ITER MITICA/HNB [14] and under EUROfusion WPHCD [15]. The accelerator system will foresee a two stage acceleration, with each stage accelerating the extracted negative ion beam of 150 keV, as shown in Fig.7. The geometry of the grids is constituted by a four grid system including: plasma grid (-310 kV), extraction grid (-300 kV), acceleration grid (-150 kV) and grounded grid (0 V). The voltage level of plasma grid and extraction grid are tuned to guarantee the correct divergence of the extracted beam. The first three grids will be manufactured with the beamlet configuration, whereas the grounded grid could be developed according to the slot concept adopted in LHD. This configuration guarantees higher acceleration efficiency.
The accelerated negative ion beam will be neutralized by a gas neutraliser followed by a Residual Ion Dump. The choice of concept for the RID will be defined on the basis of the results that will be obtained in the first few

years of MITICA operation, in which the electrostatic RID will be tested to confirm the choice for ITER. This solution is more compact with respect to the magnetic deflection RID that represents the standard backup well-experienced solution.
In order to optimize the vacuum level in the different sections of the injectors, the neutraliser conductance will be minimized and NEG (Non-Evaporable Getter) pumps will be adopted in all four sections of the injectors.
As soon as a CAD model of the DTT tokamak will be available, the integration of the injector including the duct connecting the injector to the tokamak will be developed in detail.
Since under the WPHCD of EUROfusion the concept of the photoneutraliser is under development to increase the neutralization efficiency, this option could be considered if the concept will demonstrate its validity under the present R&D activities. In this case, a theoretical increase of power up to 10-12 MW released to the plasma by each injector at the same energy could be expected.
The technologies involved in the concept of this NNBI are already available and adopted in the presently operating injectors in Japan; further optimizations could be derived from present R&D activities developed under the EUROfusion programme in view of the application in a reactor relevant device.
Therefore the proposed negative NBI is on one hand sufficiently robust and, on the other, ready to adopt new technologies to further improve efficiency, reliability and availability. An optimized NNBI system will also have an optimized set of auxiliaries and a suitable cooling system. Moreover, with the adoption of NEG pumps the cryogenic system will be excluded with significant reduction of costs and risks and increased reliability, availability and safety.

4.4 Power supplies

With regard to the Power supply system, this could be developed as a downscaled version, with reduced number of components, with respect to the ITER-like 1 MV Power Supplies and source Power Supplies based on a SF_6 insulated High Voltage transmission line. This choice requires a deep assessment and comparison with the alternative solution to adopt a full air insulated system.
The high voltage deck in air that will host the plasma source PS insulated at 300 kV will have the similar size of SPIDER system [16], where the main difference will be the insulation distance that will be longer, in order to cope with the 300 kV insulation requirement.
The layout of the two injectors could be functionally identified as in Fig.8, with a reasonable space allocated to the different PS sub-systems.

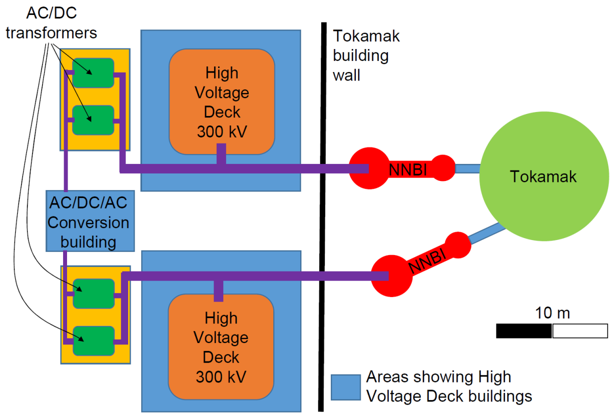


Fig.8 High voltage deck hosting all the power supplies and diagnostics dedicated to the plasma source

5. Conclusions

The additional heating systems in DTT has the main target to guarantee the necessary PSEP/R and a reliable plasma heating capable to simulate as much as possible the reactor situation. A mix of the three heating systems has been chosen for a wide flexibility in scenario development and the availability of power. The full power will not be necessary from the very beginning of the DTT operation in which 15MW of ICRH and 10 MW of ECRH will be installed. After the first phase of operation an upgrade of other 10MW of ECRH and in the introduction of 15 MW of NBI is foreseen, reaching in this way the goal of 40MW of additional power to the plasma.

Acknowledgement 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.

References

[1] F.Crisanti et al - FED_E The DTT device: choice of parameters – *this special issue*
[2] M. J. Mantsinen et al., Plasma Physics and Controlled Fusion, 45 (2003), A445–A456
[3] F. Kazarian et al., Fusion Eng. And Des., 84 (2009), 888–891
[4] G. L. Ravera et al. EuMC (2010), 1198-1201
[5] O. Tudisco et al., AIP Conf. Proc., 1580, 566-569, (2014).
[6] V. Lancellotti et al., Nucl. Fusion 46 (2006) S476-S499
[7] M. Brambilla, Plasma Phys. Control. Fusion, 31 (1989), 723-757
[8] D. Farina, Fusion Sci. Technol., 52 (2007), 154-160
[9] E. Lazzaro, S. Nowak, Plasma Phys. Control. Fusion, 51 (2009), 035005
[10] V. Erckmann et al., Fus. Sci. Technol. 52 (2006), 291-312
[11] Kojima et al. Nucl. Fusion 55 (2015), 063006
[12] A. Kojima et al., Rev. Scient. Instr. 81 (2010), 02B112
[13] Y. Takeiri et al., Fus. Sci. Technol. 58 (2010), 482-488
[14] P. Sonato et al., AIP Conf. Proc. 1515 (2013), 549-558

[15] T. Franke, et al., IEEE Trans. Plasma Sci. 26th Symp. On Fus. Eng. (2015), 1-6
[16] M.Boldrin, et al., Fusion Eng. And Des., 96-97 (2015), 411-415