TRT PLASMA CONTROL COMPLEXES CONCEPTUAL DESIGN ON THE BASE OF THE ITER FUSION TECHNOLOGY DEVELOPMENT

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The main contributions to the ITER technology platform of Russian research centers are: Nb_3Sn, Nb-Ti superconductors and poloidal PF-1 coil (JSC TVEL, Bochvar Institute, JSC NIIKP, NRC "Kurchatov Institute", Efremov Institute, JSC SNSZ), electrotechnical equipment, upper ports, divertor dome, first wall and blanket module connectors (Efremov Institute and Dollezhal Institute), gyrotrons (IAP RAS, JSC GYCOM), Port Plug Test Facilities (JSC GKMP), diagnostics (Project Center ITER, Fusion Center, Ioffe Institute RAS, Budker INP SB RAS, JSC NIITFA, NRC "Kurchatov Institute", IAP RAS, etc.) [1]. Creation and supply of the several systems (superconductors, PF-1 coil, upper ports, gyrotrons, part of electrotechnical equipment) are already successfully made and others are at the stage of prototyping and tests. Change of the ITER first wall from Beryllium to Tungsten causes the start of development and studies of Tungsten covered by B_4C first wall elements in the number of Rosatom and RAS scientific centers under RF DA coordination [2]. Involved in the ITER Project Russian research centers operate today with required highest technical levels and quality standards. Accumulated knowledges, technologies and skills form the basis for creation of the next step fusion project in Russia – Tokamak with Reactor Technologies (TRT).

The missions of the high magnetic field (B=6-8T), high temperature superconducting (HTS, REBCO), compact (R=2.15m), classical (R/a=3.8), quasi-stationary (>100sec) TRT [3] are to provide: demonstration of the possibility of quasi-stationary discharge scenarios in a plasma with reactor relevant characteristics (creation of the stationary current, temperature, density profiles of the main plasma, creation of the optimized plasma parameter profiles on its periphery for diminishing the plasma-wall interaction, including mitigation of ELM instabilities, optimization of the divertor operation, disruption mitigation technologies development); development of the quasi-stationary noninductive current drive technologies; plasma technologies and materials development for effective first wall and divertor operation; development of the integrated methods of plasma control.

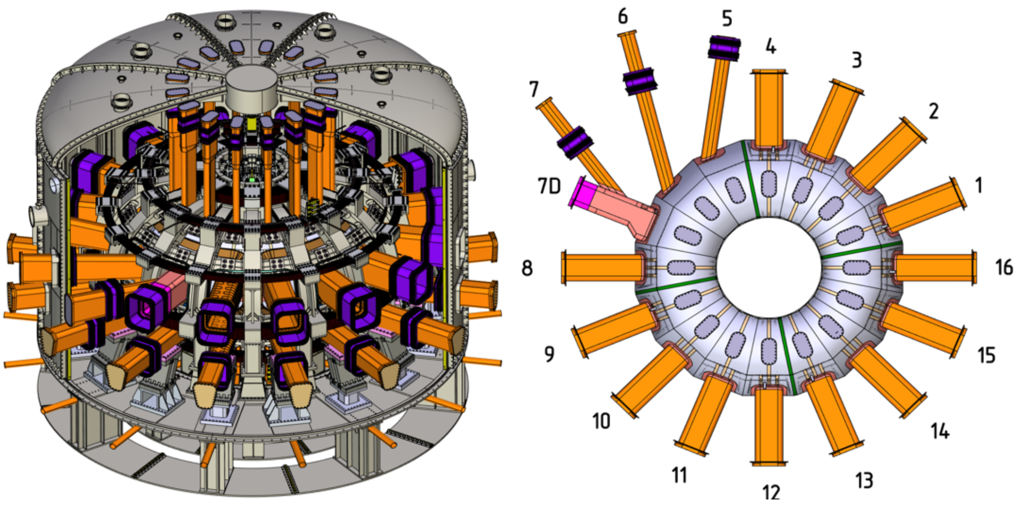

Fig. 1. TRT conceptual design with port numeration for diagnostic systems distribution

Essential fusion reactor components and equipment to be developed are: HTS electromagnetic system; vacuum vessel (double layer and cooling); advanced first wall and divertor (W, B_4C, Li …); auxiliary heating and current drive systems (0.5MeV 20 MW D NBI [4], 230GHz 10 MW ECRH [5], 60-80 MHz 5 MW ICRH, and possibly 1GHz few MW helicons [3] and 4.6 GHz 5 MW low hybrid current drive); elements of tritium breading and hybrid blanket modules; fusion reactor relevant diagnostics and plasma control systems.

Conceptual designs of the TRT diagnostic and plasma control complexes were developed in 2021-2024 (see special issues of Plasma Physics Report Vol.48, No.8 and No12, 2022 and Vol.50, No.4, 2024 and [3-5], respectively). At this stage of development, the diagnostic complex includes 17 diagnostic systems that have spatial and temporal resolution [6] providing possibility for real time plasma control to meet TRT mission. Plasma control complex provides kinetic (NBI (tangential equatorial port (EP) 5-7 at Fig.1) [3,4], ECRH (EP 16) [5] and ICRH (EP 3, 11) [3] systems, fuel and impurity pellet injection, etc.) and magnetic controls (control coils, divertor coils). Values and profiles of the main plasma characteristics and current drive will be in particular controlled by combined application of NBI with beam energy optimization [4,7], frequency, k_{II} and scheme of ICRF waves [3], by real time control of the ECRF waves injection direction (see Fig.2) [5]. Fuel and impurity gas puffing and pellet injection will be applied for plasma periphery and divertor optimization.
The neutronics modelling is one of the drivers for ports and diagnostics design. Allocation of the diagnostic and plasma control systems and their integration in TRT ports will be shown in presentation.

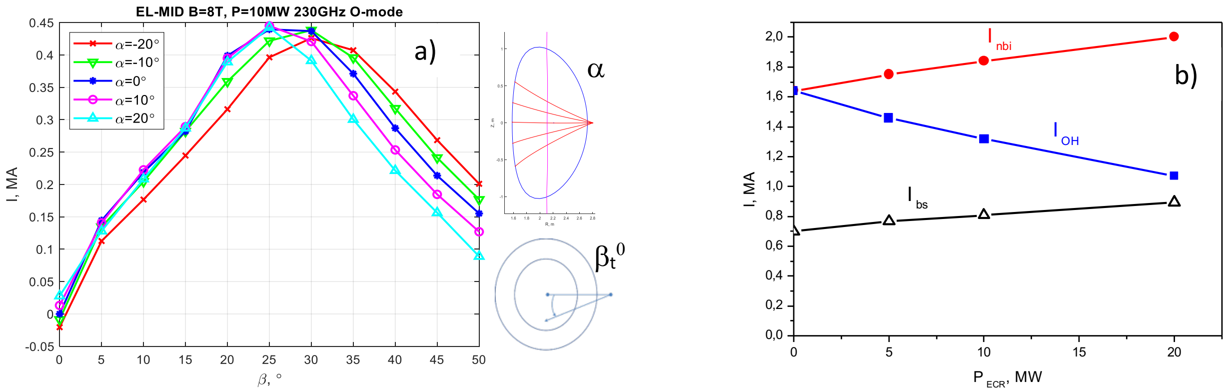

Fig. 2. a) Value of ECRF wave driven current at P_{ECRH}=10 MW, B_t = 8T, 230 GHz, depending upon injection angles (α – with respect to equatorial plane, β_t^0 – *with respect to perpendicular direction in equatorial plane, b) plasma currents upon P_{ECRH} (fixed wide ECRF wave absorption profile at ρ~0.4, P_{NBI}= 20 MW)*

During the first 6-10 years of TRT operation, the research program will be arranged according to staged approach: from integrating commissioning (stage 1) to short discharges for HTS magnetic system testing and DMS adjustments (stage 2), then to quasi-stationary long plasma with low Z (stage 3) and liquid lithium (stage 4) first wall, and at last fusion technological experiments for pure fusion and hybrid reactor development (stage 5-6). Strong cooperation is expected with ITER partners domestic experiments and especially with BEST.

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