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TRT DIAGNOSTIC AND PLASMA CONTROL COMPLEXES CONCEPTUAL DESIGN ON THE BASE AND WITH DRIVE OF THE ITER FUSION TECHNOLOGY DEVELOPMENT IN RUSSIA

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Abstract

Tokamak with Reactor Technologies (TRT) with main features: high temperature superconducting electromagnetic system and B_t= 6-8 T preliminary design and main components of it Research program with quasi-stational fusion reactor like plasma are presented. Results of preliminary design and port integration of ~30 TRT diagnostic systems required for key parameters measurements and plasma control are described. Developed during conceptual and preliminary design the TRT plasma control complex contains: NBI, ECRH, ICRH heating and current drive complexes, ELM pace making and mitigation by pellet injection, disruption mitigation system and advanced divertor with neon gas injection and separatrix swiping.

1. TRT DESIGN ON THE BASE OF ITER ACTIVITY

Magnetic Toroidal Reactor - the concept of future tokamak was proposed by A.D. Sakharov and I.E. Tamm in 1950 [1]. Name Tokamak was invented by I.N. Golovin in 1957. Studies at 22 first tokamaks, starting from TMP (Torus with Magnetic Field, 1955) and up to T-10, T-11 (1975) and TUMAN-3 (1976) were performed in Kurchatov and Ioffe Institutes at the beginning of tokamak era under L.A.Artchimovich and V.E.Golant leaderships. L.A. Artsimovich and V.D. Shafranov have proposed and constructed tokamaks with vertical elongation of poloidal divertor in 1972-73. First superconducting tokamaks T-7 (NbTi) and T-15 (Nb_3Sn) were created in Kurchatov institute in 1979 and 1988, respectively. T-11M (with ICRH and Li first wall) and T-14 (with strong magnetic field and adiabatic compression) were created in Troitsk branch of Kurchatov institute (now TRINITI) in 1983 and 1987, respectively, under E.A. Azizov, V.A. Chuyanov, I.A. Kovan and S.V. Mirnov leadership. Low aspect ratio torus Globus-M and Globus-M2 were created in Ioffe institute in 1999 and 2018. All this tokamak activity in Soviet Union and Russia provides basis for the development of the projects of: hybrid, fusion-fission, reactor (1978), OTR (1988), INTOR (1980) and T-15MD (2021) and proposal of the ITER (1985) under E.P. Velikhov initiative and leadership.

Russian Federation is responsible for development and manufacture of 23 systems (with 6 additional their subsystems) of the ITER Project. These systems include: NbTi and Nb_3Sn superconductors (manufactured by JSC TVEL, JSC ChMZ, JSC VNIINM, JSC VNIIKP, NRC "Kurchatov institute"), upper ports, electrotechnical equipment, first wall modules, divertor center assembling and thermal tests of divertor (JSC "Efremov institute"), Port plug test facilities (JSC GKMP), PF-1 coil (JSC "Efremov institute" and JSC SNSZ), gyrotrons (IAP RAS and JSC GYCOM, diagnostic systems (Institution "Project Center ITER" "Fusion center", "Ioffe institute" RAS, "Budker Institute" SB RAS, JSC NIITFA, NRC "Kurchatov institute", IAP RAS), port plugs ("Budker institute SB RAS), blanket module connectors

(JSC "Dollezhal institute") [2]. Creation and delivery of the several systems (superconductors, PF-1 coil, upper ports, gyrotrons, part of electrotechnical equipment) have already been successfully completed, while others are at the prototyping and testing stage. In general, more than 30 Russian main suppliers are involved in the ITER Project. Due to change of the first wall material from Beryllium to Tungsten in accordance with the order of the ITER Organization, RF DA organized cooperation of Russian institutions [3] (Lavrentyev Institute of hydrodynamics SB RAS, "Budker institute SB RAS, JSC "Efremov institute" JSC "TRINITI", MEPhI) for development and study of the B_4C coating (50-1000 μm) on W first wall prototypes (water and passively cooled) for use in the ITER first wall conditions. Research into properties important for use in ITER (adhesion, sputtering, redeposition, etc.) are started in the number of irradiation installations, including Tsefei-M up to 60kV, 2.5GW/m^2 electron beam facility in Efremov institute, QSPG plasma gun (0.2-3.0 μs, v = (1-3)*10^5 m/s, density= 10^{22}-10^{23} m^{-3}) in JSC TRINITI and also in MEPhI.

Involved in the ITER Project Russian research centers operate today with required highest technical levels and quality standards. All members of Russian ITER cooperation will actively participate in development and construction 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 (T > 10 keV, n ~ 10^{14} cm^{-3}) [4] are to provide: demonstration of the possibility of quasi-stationary discharge scenarios in a plasma with reactor relevant characteristics (maintaining the stationary current, temperature, density profiles of the main plasma, optimization of the plasma parameter profiles on its periphery for diminishing the plasma-wall interaction, including mitigation of the ELM instabilities, optimization of the divertor operation, disruption mitigation technologies development, etc.); 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.

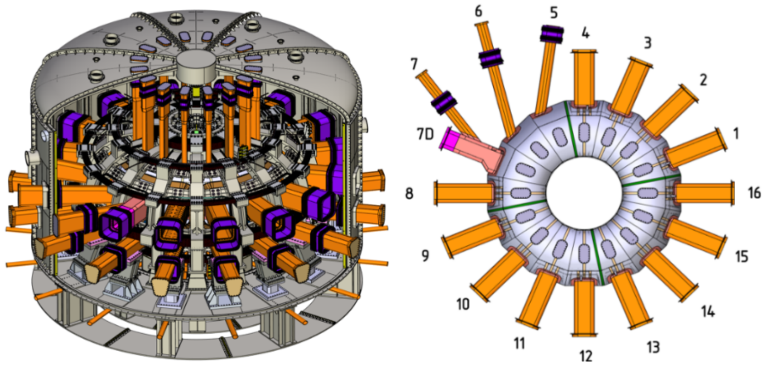

FIG. 1. TRT conceptual design with port numeration for diagnostic systems distribution.

Essential fusion reactor components and equipment to be developed are: HTS electromagnetic system; vacuum vessel (double layer and water cooled); advanced first wall and divertor (W, B_4C, Li …); auxiliary heating and current drive systems (0.5MeV 20 MW D NBI [5], 230GHz 10 MW ECRH [6], 60-80 MHz 5 MW ICRH, and possibly 1GHz few MW helicons [4] 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 [4-6], respectively). At the present stage of development, the diagnostic complex includes 17 diagnostic systems that have spatial and temporal resolution [7] providing possibility for real time plasma control to meet TRT missions. Plasma control complex provides kinetic (NBI (tangential equatorial port (EP) 5-7 at Fig.1) [4,5], ECRH (EP 16) [6] and ICRH (EP 3, 11) [4] systems, fuel and impurity pellet injection, etc.) and magnetic controls (control coils, divertor coils). Values and profiles of the main plasma characteristics will be controlled by combined application of heating/CD systems including NBI with beam energy optimization [5,8], ICRF wave heating and CD scheme allowing frequency and wave number spectrum variation [4], by real time control of the ECRF waves injection direction and changing of EC power absorbtion profile by changing of power ratio of different wavelengths in multifrequency ECRH (see Fig.2) [6]. Fuel and impurity gas puffing and pellet injection will be applied for plasma periphery and divertor optimization.

2. TRT RESEARCH PROGRAM

There are still a number of fundamental problems to be solved on the way to a thermonuclear reactor: experimental demonstration of quasi-stationary plasma burn, generation of quasi-stationary noninductive plasma current, plasma technologies and material development for efficient first wall and divertor operation, development of complex methods for plasma and machine control, disruption and runaway electron mitigation methods development, tritium breading and hybrid fusion-fission blanket design, etc. Main components of TRT research program are devoted to resolve these principle problems of the tokamak-reactor. TRT research plan is aimed at demonstration of the (quasi) stationary discharge scenarios in a reactor relevant plasma simultaneously providing safe stationary plasma – wall (divertor) interaction and non-inductive CD with robust plasma control. Synergetic usage of powerful heating / CD (NBI, ECRH, ICRH, helicons, LHCD), plasma fueling and magnetic control systems provide a wide range of the algorithms for effective and robust plasma control. Disruption mitigation and ELM control technologies will be developed and studied. Usage of the Be or covered by B_4C W or liquid Li first wall components are under experimental analysis now. Several divertor designs are under consideration. Fast ion behavior in plasma with fusion characteristics will be important part of the program. High magnetic field, ~20 MW, 500 keV deuterium and hydrogen injection in combination with ICRH (minority and three ion schemes) and ECRH are providing relatively unique possibility to study at TRT the not only DD and trace tritium DT fusion but also the efficiency of fusion in the number others fuel mixtures such as: p+^{9}Be (2% in plasma), p+^{11}B (2%), D+^{6}Li (0.2%), D+^{9}Be (2%), D+^{11}B (2%), and especially D+^{3}He (0.1-10%) and D+^{7}Li (1.8%) = n+^{4}He+^{4}He (σ=2*10^{-1} b at 200 keV, Q=15.1 MeV). Essential part of the program will be technological studies of breading and hybrid blanket components behavior.

According to current plans the initial phase of research program (6-10 years) consists of six stages: 1) integrated commissioning with plasma, 2) short discharges with maximum magnetic energy, with first wall inertial cooling, limiters and disruption mitigation system, 3) long pulses of plasma with reactor parameters with low-Z covered plasma facing components, 4) Li technologies for first wall and advanced divertor, 5) experiments with trace tritium, 6) full size technological experimental program for pure and hybrid reactor. Strong cooperation in domestic experiments is expected with ITER partners and especially with BEST project.

3. CURRENT STATUS OF TRT DIAGNOSTIC COMPLEX

The TRT diagnostic complex of TRT is being developed [7] to provide plasma characteristics measurements and generating the required signals for plasma and machine controls to meet TRT missions [4] and research program [8] during long quasi-stationary discharges. The neutronics modelling is one of the drivers for ports and diagnostics design. Based on the TRT 3D model (shown in Fig.1), the Open-MC neutron transport calculations were performed [9] for model with TRT related material containment. The results of DT neutron flux distribution in TRT horizontal and vertical cross sections calculated for nominal (n=10^{14} cm^{-3}, T=10 keV) TRT 50%D+50%T plasma parameters are shown in Fig.2. The DD neutron flux is of the 100 times lower for deuterium plasma and total flux is approximately doubleted in the case of 1% trace tritium plasma. These results of neutron flux calculations provide radiation conditions of various diagnostic systems operation and demonstrate possibility of TRT quasi-stationary operation for pure deuterium and 1% trace tritium plasmas and a possibility to have short ~5s pulse in 50%D+50%T plasma. This duration is restricted by the toroidal coils heating for the current design of nuclear shielding.

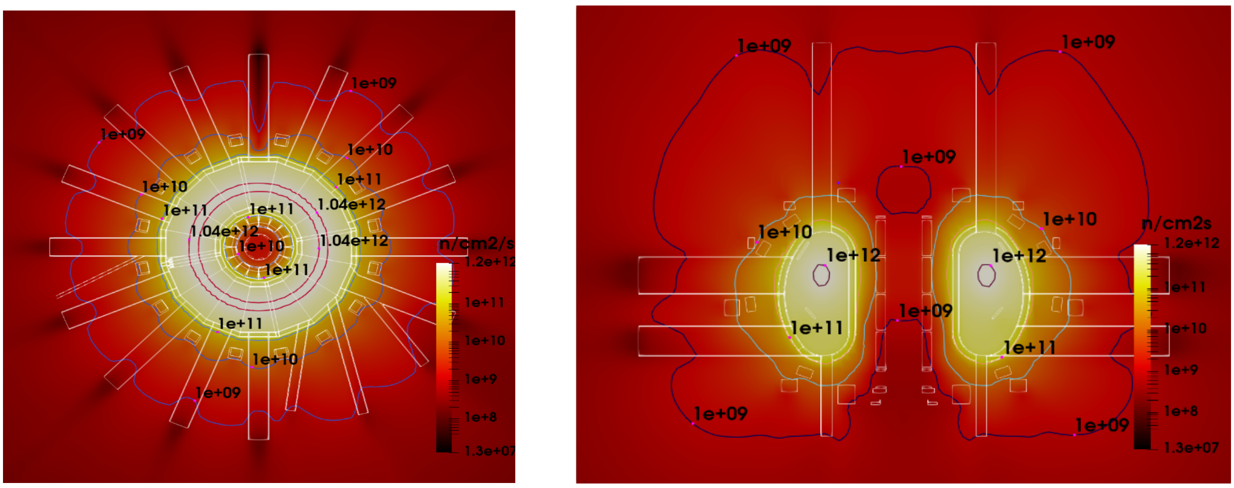

FIG.2. DT neutron flux distribution in TRT horizontal and vertical cross sections calculated for nominal (n=10^{14} cm^{-3}, T=10 keV) TRT 50%D+50%T plasma parameters.

Russian scientific centers are responsible for development and manufacture of 7 diagnostic complexes, 5 port plugs and 4 Port plug test facilities for ITER. These diagnostic complexes are: Neutral particle analyzers (Low energy NPA, High energy NPA, Diamond fast charge exchange atom spectrometer, diamond and scintillator neutron spectrometers and γ-ray spectrometer); Thompson scattering system with interfaces for Laser induced fluorescence; high field side reflectometry and refractometry; H_α-spectroscopy, Charge exchange resonance spectroscopy, vertical neutron camera with U-238 fission chamber counters and diamond compact spectrometers, Divertor neutron flux monitors with the numbers of U-235 and U-238 fission chambers. Designs of DNFM with two units containing three U-235 and three U-238 of various sensitivities are shown in Fig.3. Fission chambers were specifically designed for the harsh conditions and wide range of ITER fusion power measurements from 100 kW up to 700 MW with time resolution of up to 1 ms and with high (~99.9%) availability target for ITER scenarios control with maximum fusion power. The prototypes of developed for use in Vertical neutron camera and NPA collimator in equatorial port 11 compact CVD diamond neutron and fast atom spectrometers are shown in Fig.4 and 5. The results of energy calibrations of the fast atom CVD diamond spectrometer before its successful application in the EAST experiments are also shown in 5.

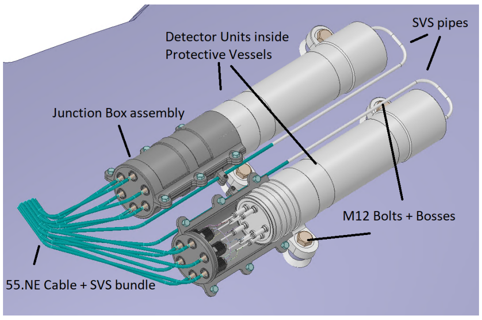

FIG.3. ITER Divertor neutron flux monitor. design.

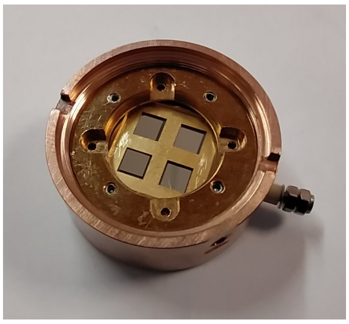

FIG.4. Prototype of ITER fast atom and neutron spectrometer with 4 CVD diamonds

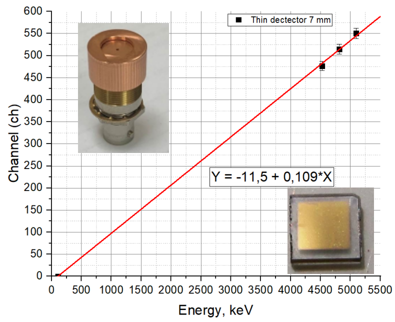

FIG.5. Results of CVD diamond detector (in inserts) fast atom energy calibration

CVD diamond detectors have being grown on Boron doped high pressure high temperature diamond in microwave plasma reactor ARDIS-300. Back contact is substrate of B-doped HPHT diamond and front contact was made of Ti as shown in Fig.5 insert. Highest purity and crystal quality of the growing CVD diamond provide detector charge collection efficiency 97% and 97%, and energy resolution 0.8% and <3% under 14.7 MeV neutron and 5.5 MeV alpha-particle irradiation, correspondingly, as shown in Fig.6.

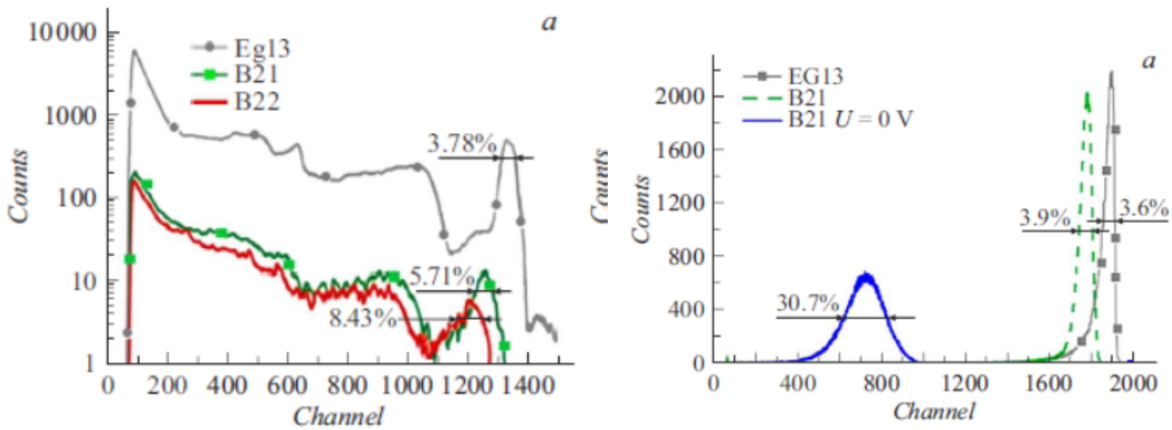

FIG.6. Pulse height spectra of two CVD diamond detectors (B21 and B22) in flux of 14.7 neutron (left) and 5.5 MeV ^{241}Am alpha-source in atmosphere (right) shown in comparison with "Element-6" etalon CVD detector (Eg13).

The experience gained in creating of the ITER diagnostic systems is used in development of TRT diagnostics. Currently, the preliminary design and port integration of the TRT diagnostic complex [10] contain of the order of 30 diagnostic systems. Their allocation in machine ports are presented in Table 1. Some diagnostics, such as the lost ion probes, divertor and technological diagnostic, et.al. are still under consideration. The number of crucial for plasma control profiles of TRT characteristics ($T_i(r)$, v_i (r), J(r), $Z_{eff}(r)$, etc.) active measurements will be done by CXRS, MSE, DES and NPA diagnostics using Diagnostic neutral beam injector which will be allocated in equatorial perpendicular port 7D. Integration of these systems on TRT is presented in Fig.7. Diagnostic NBI will operate in the energy range 60-80 keV on a positive ion source. The integration of the Multichannel neutron collimator inside the TRT equatorial port 1 with results of the neutron flux density calculation inside collimator and in its detectors positions are presented in Fig.8. The most challenging requirement to all diagnostics is operation during long quasi-stationary TRT discharges with corresponding data acquisition systems, time and spatial resolutions required for plasma control.

TABLE 1. STATUS OF DIAGNOSTIC [10] AND PLASMA CONTROL SYSTEMS ALLOCATION IN TRT PORTS*.

Port #	Equatorial	Upper	Divertor
1	Multichannel neutron collimator [11], Neutron flux monitors [12], Neutron activation system, Hard x-ray monitor, γ-ray spectrometers and collimators [13]		
2	Test blanket modules		
3	ICRH system [4, 14]		
4	Test blanket modules	Reflectometer, Refractometer	
5	NBI heating and current drive [5]		
6	NBI heating and current drive, Diamond fast atom spectrometers [15]		
7	NBI heating and current drive Diagnostic NBI (port 7D) [16]	CXRS, NPA, Diamond fast atom spectrometers	
8	CXRS [17], MSE [18], BES, Z_{eff} , Laser for Thompson scattering [19], Tangential NPA [20]		
9	Thompson scattering, BES, H_α-monitor, Impurity lines spectrometer, IR-thermography, first wall diagnostic		
10			
11	ICRH system		
12	Reflectometer [21], MSE, EC-emission measurements	IR-divertor thermography, Reflectometer	
13	TS laser dump, TS in divertor, H_α-monitor, Fast video-range camera, Neutron flux monitor, Neutron activation system		
14	Radiation (bolometry, SXR, XUV) tomography, NPA, Hard X-ray monitor	Radiation tomography, refractometer	Radiation tomography
15		Reflectometer	
16	ECRH system [6]		TS in divertor

- Equipment of the Electromagnetic diagnostics is placed in many ports. Disruption mitigation system, Pellet injectors, Helicons, lost ion probe, divertor and technological diagnostics are still under consideration.

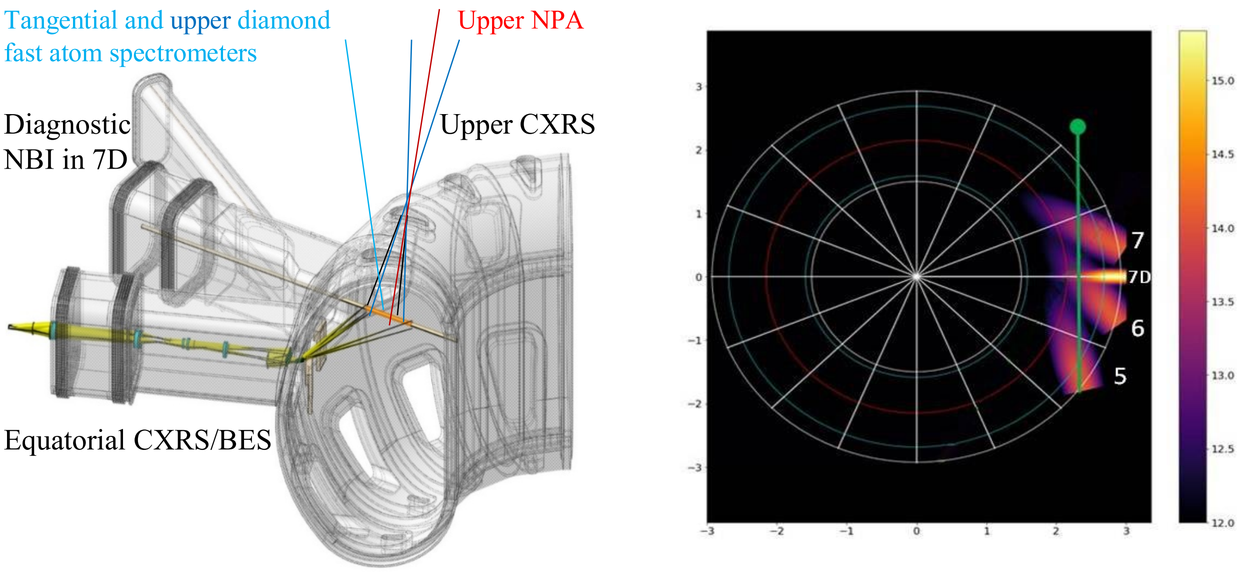

FIG.7. Left: Allocation of Diagnostic NBI in equatorial port 7D, CXRS in equatorial and upper ports, tangential in port 7 and vertical (upper port 8) Diamond fast ion spectrometers and vertical NPA in upper port 8. Right: Results of calculation of NBI and DNBI atom density in TRT plasma equatorial cross section and CXRS/BES/MSE line of sight from equatorial port 8.

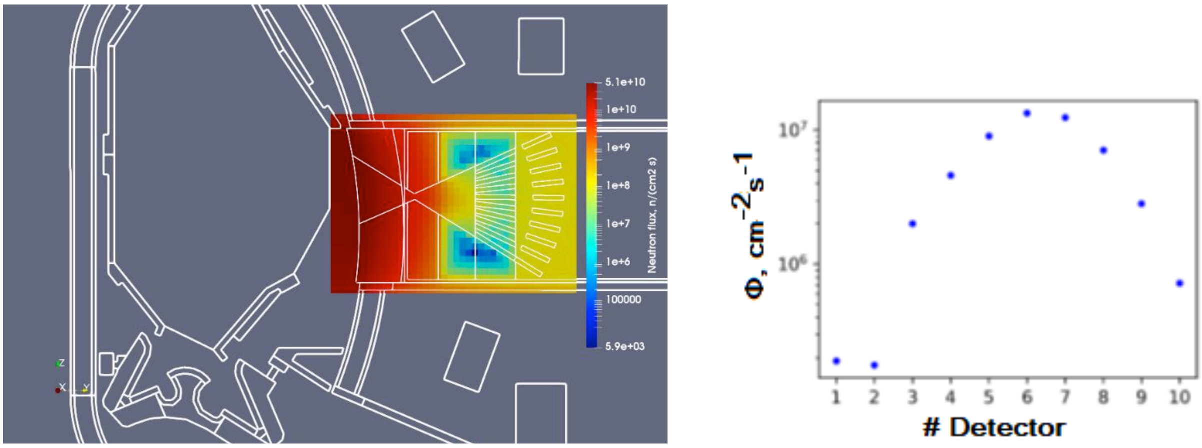

FIG.8. Results of neutron flux density calculation inside (left) of integrated in TRT equatorial port 1 multichannel neutron collimator and in its detector position (right).

4. TRT PLASMA CONTROL COMPLEXES CONCEPTUAL DESIGN

The main TRT parameters: axial magnetic field 6-8 T, electron density ~10^{14} cm^{-3} and discharge duration – more 100 seconds determine the plasma control methods that will be used. The most powerful and efficient control tool looks to be the six up to 500 keV deuterium Neutral beam injectors based on negative deuterium/hydrogen ion sources and having total power of about 20 MW [5]. To increase atom beam power several innovations will be applied in each beam, namely: shift of accelerator axis from ion source axis by magnetic field, plasma stripping cell and the energy recuperation technology [5]. These six NBIs will be installed in specially designed tangential equatorial ports #5, 6 and 7 (two in each port) to provide not only heating but also co-current current drive. Injection angle will not be changeable ($R_{inj}=R_0$-a/2) and plasma control will be provided by changing of the beam energy and power and synergism with ICRF waves, ECRH, pellet injection, etc. Optimal current drive will be reached in density range 0.5-1*10^{14} cm^{-3}.
ECRH system containing ten-twelve 1 MW long pulse gyrotrons with frequency 230 GHz (specially being developed for TRT 8 Tesla magnetic field [6]) and/or 170 GHz (created for ITER 6 Tesla field). Currently ECRH antenna and mirrors for changing wave injection angle are allocated in equatorial port 16. Efficiency of ECRF wave system for current drive in TRT upon the injection angel is shown in Fig.9. Analysis (Fig.9.a)) demonstrates that the ECRH driven current profile could be optimized by adjusting irradiation angles in the ranges α = +/- 10^0 with respect to equatorial plane and β_t = 25^0-30^0 in toroidal direction. The profiles of wave absorption and current generation for f = 230 GHz, $\alpha = 0^0$ and $\beta_t = 25^0$ are in the range 0.2 – 0.38 r/a with HWHM = 0.9 r/a. For better profile control due to wave absorption two independently rotating mirrors (for 5 and 5 MW) are considered for application in TRT. ECRH, in addition to the direct current drive, could increase NBI-induced current drive due to electron heating. Results of calculations of this synergetic NBI and ECRH current drive for TRT conditions with PNBI = 20 MW are shown in Fig.9. b). Important that two major TRT heating methods provide strong synergetic tools not only for heating and current drive total efficiency and profile control but also for the bootstrap current and its profile control altogether with plasma fueling methods and in particular to fueling pellet injection.

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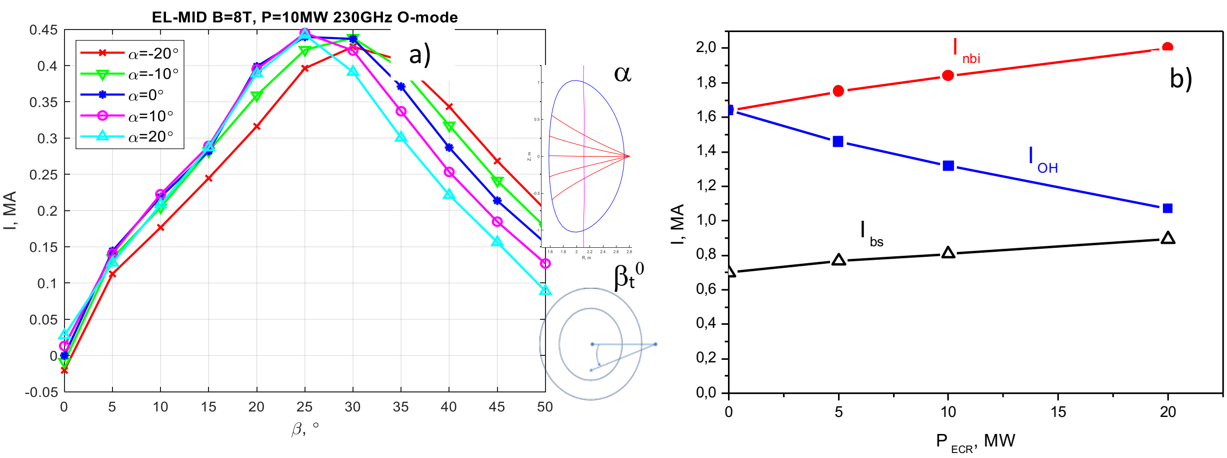

FIG.9. a) Value of ECRF wave driven current at P_{ECRH}=10 MW, B_t= 8T, 230 GHz, depending upon injection angles (α – with respect to equatorial plane, β_t^0 – *with respect to perpendicular direction in equatorial plane, b) plasma currents upon P_{ECRH} (fixed wide ECRF wave absorption profile at ρ~0.4, P_{NBI}= 20 MW)*

Application of ICRF wave heating and current drive in frequency range 45-80 MHz with power up to 5 MW is planned at TRT [4, 14]. Two multi strip ICRH antennas currently allocated in equatorial ports 3 and 11. Several schemes of heating will be applied: $^3He^{2+}$ minority, second harmonic of plasma T^+ at frequency 80/60 MHZ at B_t = 8/6 T and plasma's D^+, $^9Be^{4+}$, $^7Li^{3+}$ and beam's D^+ at frequency 60/45 MHz at B_t = 8/6 T using majority, minority and three ion scheme mechanisms. The wave number (k_{II}) spectrum will be selected to optimize ICRF wave heating and current drive especially in cases of its synergetic application with beam injection. Special interest will be devoted to experimental studies of the possibility to increase fusion rates in ICRH + D^+ NBI into deuterium plasma discharges with $^3He^{2+}$ and wall impurity minorities ($^9Be^{4+}$, $^7Li^{3+}$, $^{11}B^{5+}$) exploring minority and three ion scheme mechanisms. High wave absorption in plasma core and good antenna coupling during H-mode operation are expected. Well proven methods for impurity flux control by respective multi strip antenna phasing and keeping good antenna coupling during ELM oscillations will be applied.

Off-axial generation of noninductive plasma current by few MW helicon waves with frequency 1000 - 1200 MHz and 4.6 GHz low hybrid waves are also under consideration. In the latter case up to 1 MA LHCD at RF power of 6 MW is predicted.

Three options of the first wall: Beryllium [22], covered by low Z material (B_4C) Tungsten [23] and liquid Lithium [24] are under analysis for TRT. Beryllium first wall design is very well developed in frame of the ITER project. Low Z material covered Tungsten and possibility of foil restoration during exploration [25] is under experimental testing now. Especial promise is related with experimentally proved at small and short discharge duration tokamaks technology of B_4C coverage by evaporation of carborane ($C_2B_{10}H_{12}$) directly into tokamak plasma discharge [25]. Technologies of liquid Lithium first wall on the basis of capillary porous structure emiters and collectors are developed in experiments at T-11M, T-10 and will be further studied in T-15MD experiments. Development and experimental studies of the efficiency of all three first wall options from the point of view manufacture, performance and restoration during long discharges will be essential part of TRT research program. Another crucial part of TRT research program will be further development of the plasma discharge control technologies (ELM mitigation, periphery magnetic field perturbation, optimization of plasma profiles, etc.) to provide maximum diminishing of plasma wall interaction during long program of TRT quasi-stationary operation.

One of the major problems of future fusion reactor is the divertor operation. Taking this into account the development of the control of various divertors operations is the main part of TRT research program. Several advanced divertor options are under planning for studies [22]. It is suggested that the machine will start operation with ITER-like Tungsten divertor with neon gas puffing for reirradiation of the energy flux coming into divertor along the separatrix. It was shown in [26] that the broad enough parameter space of ITER-like TRT divertor operation exists for plasma current of the order of 4 MA. In addition to neon gas puffing the possibility of the separatrix swiping [22] is also included into the TRT preliminary project. Other option considered in the preliminary design is the divertor with mechanically swiping of the energy flux accepting targets [22]. The corner divertor with neon gas puffing is the another divertor option [15]. SOLPS-ITER calculations demonstrate possibility to diminish ion and electron temperature at separatrix down to 1 eV having $Z_{eff} \sim 2.4$ by 10^{19} s^{-1} neon gas puffing that simplify transfer to detachment regime. This analysis [27] indicates that plasma temperature close to divertor targets could be further diminished by additional deuterium puff with respective lowing neon puff and such a way in parallel diminishing Z_{eff} down to 2. Development of the TRT advanced divertor control technology will be well related to used design of first wall, its plasma facing material and applied technologies for optimization of plasma – first wall interaction.

Essential attention in the TRT preliminary project is devoted to disruption mitigation. For that it is necessary to 1) diminish (equally redistribute) thermal loads on plasma facing elements at stage of the thermal quench, 2) keep mechanical loads on electroconducting elements within tolerable range at the stage of current quench (not so fast, not so slow), 3) prevent or mitigate formation of relativistic electron beams. For disruption mitigation at TRT it is planned to apply the low-Z impurity injection, preferably from first wall material (Be, B). Preliminary analysis shown that plasma thermal energy can be successfully redistributed over first wall by means of impurity radiation. Low-Z impurity are fully ionized and after this stop radiating. Plasma temperature increases causing corresponding electroconductivity increase and increasing of current quench times up to the values providing plasma vertical control and in addition due to the respective decrease in toroidal electric field, the possibility for relativistic electron generation is disappearing.

5. SUMMARY

TRT Research program focused on achieving and optimizing of the quasi-stationary (>100 s) fusion reactor relevant plasma regimes with the development of quasi-stationary noninductive current drive and bootstrap current formation, development of the technologies for optimization of quasi-stationary plasma - wall interaction and advanced divertor operation. In addition to experiments in DD and, at further stage of TRT operation, in trace tritium DT plasmas the fusion efficiency of the number fuel mixtures will be studied in experiments with combination of NBI, ICRF and

ECRF waves heating. Operation of the components of pure fusion tritium breading and hybrid, fusion-fission, experimental blanket modules will be studied. Beginning phase of the TRT Research program is based on a stage approach.

Preliminary design and first option of port integration of ~30 TRT diagnostic systems required for key plasma parameters measurements for machine operation, plasma control and advanced plasma phenomena studies were performed.

Developed during conceptual and preliminary design the TRT plasma control complex contains:

- Six heating and current drive 500keV deuterium / hydrogen neutral beam injectors of total power ~20 MW,
- ECRF wave heating and current drive complex on the base of twelve 230/170 GHz gyrotrons,
- ICRF wave heating and current drive complex operating with two antennas in frequency range 45-80 MHz,
- Helicon or low hybrid wave off-axis current drive system,
- ELM pace making and mitigation pellet injection system,
- Disruption mitigation system, based on low-Z (Be, B) impurity injection,
- Options of advanced divertor with neon gas puffing and separatrix swiping.

On the basis of ITER project activity and results the TRT team is developing in participating scientific institutions and industrial plants the information technology platform arrangement [28] to provide the remote design and remote participation in joint researches.

TRT Research program is open for close cooperation with the ITER Project, ITER partners domestic experiments and especially with the BEST project.

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