TOKAMAK WITH REACTOR TECHNOLOGIES CONCEPT

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Abstract

Concept of a tokamak with reactor technologies (TRT) is developed to facilitate fast and economically sound transition to the pure fusion reactor as well as to the fusion neutron source (FNS) for the hybrid fusion-fission system. Well controllable steady - state operation and reliable power and particle control in a reactor relevant conditions are principal plasma physics problems to be resolved on the way to both fusion reactor and FNS. Finding optimal solutions to them determines the mission of TRT. To explore wide variety of technically feasible proposals to achieve these goals the experiments should be performed in low activation conditions, i.e. mostly with H and D plasmas. However tritium trace experiments are also foreseen for the TRT research program. TRT electromagnetic system is designed on the base of REBCO high-temperature superconductor providing B_{t0} = 8 T at the machine center. The increased magnetic field will allow achieving the tokamak fusion plasma regimes with Q ~ 1 at moderate machine size (R = 2.15 m, a = 0.57 m) and therefore lower cost. TRT will be able to operate in quasi-stationary (>100 s) regimes with hydrogen, helium, and deuterium plasma and with short (t < 10 s) deuterium-tritium discharges (Q ~ 1) limited by radiation heating of the toroidal field coils. Missions of TRT are: (i) development and integration of the key fusion reactor technologies in one machine; (ii) development and investigation of the quasi-stationary plasma discharges; and (iii) development and investigation of regimes with burning fusion plasma with the domination of alpha-particle core plasma heating during limited by radiation heating deuterium-tritium experiments. The reactor technologies to be tested here include the HTS EMS operating at the extremely high magnetic field; the solid metal and liquid lithium first wall and advanced divertor; the several ten MW, 0.5 MeV-range NBI; 230 GHz, MW-range gyrotrons; 60-80 MHz, MW-range ICRH system; the tritium complex; the remote control technologies; and the reactor-relevant diagnostics. The conceptual design of the main components of the TRT and its expected performance characteristics are presented in the paper

1. INTRODUCTION

Development of the fusion – fission hybrid system based on the tokamak as the FNS is considered to be one of the principal goals of Russian fusion research program. Basic requirements for the FNS to be attractive for industrial fission fuel breeding or suitable for post-combusting of long-lived transuranic radionuclides in a hybrid system are to provide several tenth of MW of fusion power at relatively low gain factor Q~1 with neutron flux to the wall of ≥0.2 MW/m^2. FNS should operate continuously (steady – state) or in pulsed regime with neutron production pulse duration, Δt, to pause ratio exceeding 80%. Thus requirements on power loads on the plasma facing elements in FNS are significantly less stringent compared to the pure fusion reactor.

Fast progress in HTS technology opens new high field operation domain for tokamaks. Along with technical advantages of HTS for implementation in tokamak magnets, higher field allows more compact design of FNS at lower capital cost. At the present TRT design is focused on the relatively compact R/a = 2.15/0.57 tokamak with B_t=8T provided with HTS (REBCO) toroidal field coils. Poloidal field coils (PFC) were originaly designed with use of LTS Nb3Sn (for central solenoid) and Nb-Ti conductors. However recent project developments focus on fully HTS electromagnetic system. It is recognized that operation of the HTS magnet in high magnetic field is reliable at rather low temperature. Thus requirements for the TRT cryoplant are assumed to be similar for LTS, HTS or LTS+HTS variants of its electromagnetic system.

Achievements of the fusion gain factor Q > 1 is a principal goal of the world wide fusion research. Significant progress in understanding the physical processes governing the plasma confinement is based on the experiments of T-10, TFTR, JET, JT-60U, D-IIID, Alcator C-Mod, FTU, ASDEX Upgrade, KSTAR, EAST and others tokamaks provided physical and technological base for construction of the ITER [1] and next step fusion devices.

Evidently rising the magnetic field allows to increase plasma current and, therefore, improve confinement of the tokamak plasma. In parallel higher magnetic field assists to expansion of the plasma pressure and density operation boundaries, facilitates improvement of the MHD stability. Thus modern tokamak projects aimed to operate with reactor-relevant plasmas including ITER, DTT, SPARC was developed for the magnetic field of maximum amplitude achievable within existing technology. Remarkable recent progress in development high temperature rare-earth barium copper oxide (REBCO) superconductors (HTS) opens the way to design a tokamak electromagnetic system with toroidal field at the machine large radius R0 of 8T or higher. One of the key advantages in using the REBCO HTS in a tokamak EMS is that these superconductors manifested necessary functionality at extremely high magnetic field within wide temperature range from 5K to 30K allowing to ease requirements on the cooling system. Higher magnetic field gives the possibility to achieve ignition in the tokamak of essentially smaller size and, therefore, capital cost. Such a compact machine with hign field and fusion gain factor of Q~1 is important step on the way to develop pure fusion as well as hybrid (fusion-fission) reactor.

Tokamak with Reactor Technologies, TRT, is a compact tokamak with high magnetic field provided by HTS EMS aimed to steady state operation with D reactor relevant plasma and to DT operation with amount of tritium and pulse duration limited by nuclear heating of the TFCs. Conceptual design of the TRT is coordinated by Rosatom Institution "Project Center ITER" and carried out by specialists from leading RF institutions involved in ITER activity including Efremov Institute, Kurchatov Institute, TRINITI, Budker Institute, Institute of Applied Physics. Conceptual design and analysis of the TRT plasma performance were carried out with use of existing scientific and technological data bases taking into account recent developments of technologies for the first wall and divertor, auxiliary heating and CD systems and numerical codes including DINA, ASTRA, TRANSMAK, KINX and others.

2. CONCEPT AND MISSIONS OF TRT

TFTR [2], JET [3] and JT-60U [4] experiments demonstrated record fusion power and provided major part of the physical basis for ITER design. According to ITER Physics Basis [1] the fusion performance could be improved by: discovering new operating regimes, increasing of the machine size or of the operating magnetic field. During the ITER engineering design the superconducting material Nb3Sn was chosen as the best to provide, magnetic field of 5.3 T at R0 = 6.2 m to achieve required performance with Q = 10.

There is a consensus in fusion community that ITER will provide the necessary physical basis and technology platform required for creation of the future fusion reactor. Nevertheless, ITER project does not cover all required for fusion reactor technologies as well as does not allow for the most recent progress with high temperature superconductors, liquid metal first wall, divertor and other fusion reactor technologies. In particular, HTS technologies open possibility to design and build high magnetic field compact experimental tokamak – reactor with Q > 1 quickly enough and within reasonable capital cost to make significant contribution in development of the future fusion reactor.

Presented here concept of the Tokamak with Reactor Technologies is based on consistent integration of the currently available and most perspective near future technologies with the aim to maximize the toroidal magnetic field in the machine. Multi-parametric system analysis of the main tokamak EMS components including available and expected HTS, available and perspective construction materials enable to withstand mechanical forces under normal operation and disruptions resulted in determination of the TRT axial magnetic field value equal to 8 Tesla with REBCO HTS magnetic coils operating at 5 – 20 K provided by liquid helium cryogenics. TRT considered as full size quasi-stationary (>100 s discharge duration) plasma prototype of the future fusion neutron source (FNS) for hybrid fusion-fission reactor with Q~1 as well as technological platform to develop technologies for the pure fusion reactor. Real opportunities to develop necessary NBI, ICRH and ECRH systems for plasma heating and current drive add optimistic expectation to extend pulse duration in a reactor relevant plasma to the stationary.

TRT research plan will include several stages of operation starting with non activated hydrogen and helium phase, then operation with deuterium and finally DT within the limitations set by nuclear heating of the EMS components.

TRT concept in a some way follows the so called "high-field path" to fusion energy (Q~B4) represented by T-14 [5], FTU [6], Alcator C-Mod [7] experiments and CIT [8], Ignitor [9] DTT [10] and SPARC [11] projects. Main parameters of the TRT are shown in table 1 in comparison with other machines and projects.

TABLE 1. TRT MAIN PARAMETERS IN COMPARISON WITH OTHER MACHINE AND PROJECTS

	C-Mod	T-14	CIT	Ignitor	DTT	SPARC	TRT	ITER
R_o m	0.67	0.42*	2.1	1.32	2.11	1.85	2.15	6.2
a m	0.22	0.12*	0.65	0.47	0.64	0.57	0.57	2.0
A	3.0	3.3	3.2	2.8	3.3	3.2	3.77	3.1
B_0 T	8.0	12.5*	10.0	13.0	6.0	12.2	8	5.3
I_p A	2.0	1.2*	11.0	11.0	5.5	8.7	5	15.0
λ_{sep}	1.8	1	2.0	1.83	1.8	1.97	1.8	1.85
δ_{sep}	0.4		0.25	0.4	0.4	0.54	0.3	0.48
$\Delta t_{flattop}$ s	1	0.2	5	4	90	10	>100 <10*	1000
$\Delta\Psi$ Wb	8		75	33	33Vs	42	~33	277
$<n_e>$ 10^{20} m^{-3}	2 - 8	8*	3	4.8	1.8	1.4	2	1
τ_E s	~ 0.1	0.06		0.62	0.43	0.77	0.33	
P_{aux} MW	6	3.5	20	24	45	25	45	73
P_{fus} MW	0	3.5*	800	96	~ 0.01	140	1 / 50**	500
P_{sep}/R MW/m	~ 10			16	15	15.7	14 / 16**	15
Q	0	1	infinity	9	0	11	0.01/ >1**	10

* expected T-14 parameters after adiabatic compression,
** TRT in DT plasma.

Primary mission of the TRT is the integration in one single machine the series of key fusion reactor technologies including HTS EMS operating in high magnetic field, metal and liquid metal first wall and divertor, negative ion neutral beams with energy of the order 0.5MeV, megawatt range gyrotrons with frequency 230-260GHz, 5MW ICRH system with frequency range 60-80 MHz, tritium complex, remote plasma control and reactor relevant diagnostics. Second mission of TRT consists in development of the long steady state discharge scenarios in the plasma with reactor parameters and reliable power and particle control. The third mission of the TRT is the development of the tritium technologies and conducting DT experiments with alpha heating within the limits set by nuclear heating of the EMS. Realization of the TRT concept is important step towards creation of the economically attractive fusion reactor and quick start of fusion energy production.

3. MAIN FEATURES OF THE TRT

General view of the TRT main components is shown in Fig 1. Vacuum vessel, ports, magnetic coils and in-vessel components are shown in Fig. 2. Cryostat is the largest component of TRT. Cryostat single wall body will be manufactured from AISI 304L stainless still. According to current preliminary design its volume equal to 1170 m3, height – 11m, diameter – 12 m and mass – 317 t. In addition to mechanical functions cryostat wall operates as thermal barrier between outer space and its internal space where vacuum vessel and high temperature superconducting system are placed.

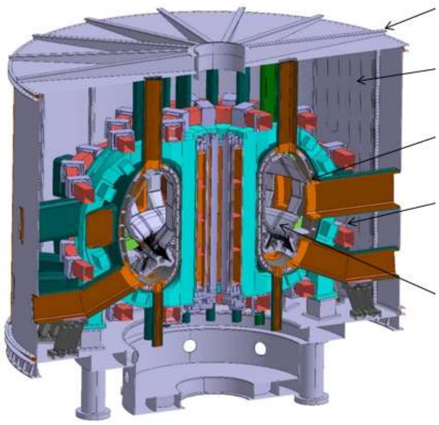

FIG. 1. General view of the TRT in cryostat

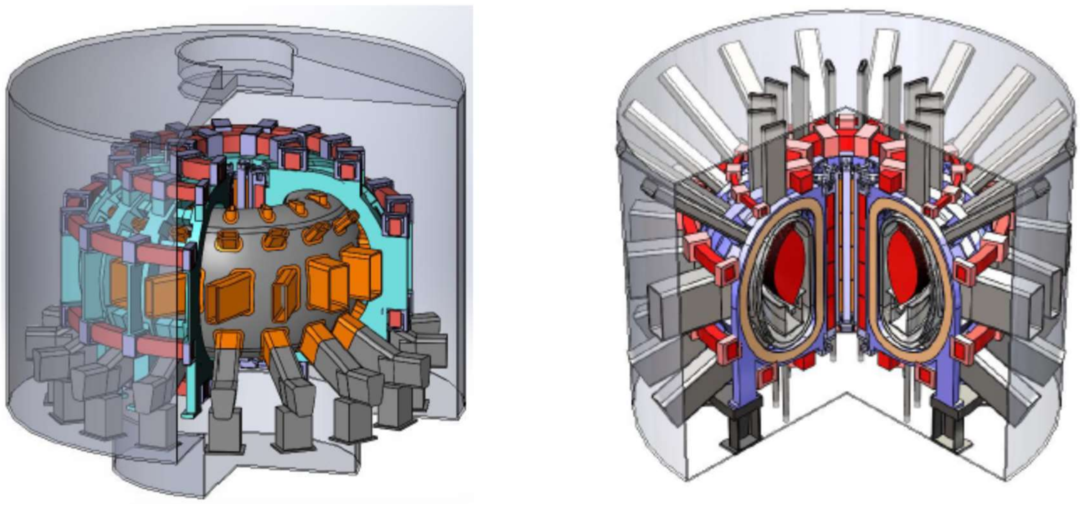

FIG. 2. *TRT vacuum vessel, ports, magnetic coils, in-vacuum vessel components, plasma.*

Electromagnetic system of TRT includes 16 D-shaped toroidal field coils (TFC), 6 poloidal field coils (PFC), 4 sections of central solenoid (CS), 24 correction coils (CC) and 4 horizontal field control coils and 2 coils for swinging the separatrix strike points. To provide TRT long pulse (>100 s) quasi-stationary operation with axial magnetic field equal to 8 T TFC, PFC, CS sections and CC designed from REBCO HTS. Liquid helium cooling provides operation temperature range 5-20 K. To withstand electromagnetic loads during disruptions TRT EMS is designed as single mechanically rigid structure made of special stainless still. 4 horizontal field control coils should be manufactured from very strong to mechanical forces CuCrZr or silver containing cupper. Toroidal field magnet consists of 16 TFCs providing maximum ripple amplitude of 0.5% at the plasma boundary.

Vacuum vessel of TRT has the double wall construction from austenitic stainless still AISI 316LN. Outer and inner walls of VV have thickness 25 mm and connected by 25 mm ribs. Outer diameter of the VV is equal to 664 cm, height 385 cm, mass 170 t. VV cooling will be provided by circulating water with pressure p = 1 MPa and temperature 300C. Stress analysis for VV and passive stabilization coils designs demonstrated that stationary and disruption induced mechanical loads are under construction limits.

To increase radiation shielding provided by VV during high neutron yield discharges the boron water will circulate between VV walls. Total thickness of the VV including placed on its outer surface shielding blocks is equal to 220 mm. Shielding blocks are designed as stainless still boxes with overall thickness 30 mm containing 5.5 mm 10B4C layer and 7.5 mm W layer. According to preliminary calculations such design of VV and shielding blocks will provide HTS TFC long pulse deuterium plasma operation at YnDD ~ 1*1018 n/s and during short pulse (<10 s) deuterium-tritium experiments.

Main parameters of the TRT vacuum vessel water cooling and heating system were determined both for TRT operation regime and VV heating procedure. Hydraulic and thermal calculations for water flow in channels between VV walls and ribs were performed for TRT operation and VV heating regimes up to 1700C using ANSYS CFX 2019 R3 package.

Two options of TRT high vacuum pumping system were preliminary designed. First one includes 32 turbomolecular pumps Pfeiffer ATP2300M with pumping speed 1,7 m3/s for hydrogen, placed in 16 divertor ports by two pumps in each in 16 m distance from machine axis where magnetic field does not exceed 5 mT. Second one includes 6 cryopumps placed in 6 divertor pots 7m away from machine axis outside cryostat. Final decision on selection of high vacuum pumping system will be done at later stage of design.

TRT FW consists of water cooled modules. Each module can be independently replaced by the new one when required. According to design the life time of FW module in TRT is determined by plasma facing material erosion and stress accumulation in FW mechanical supports and is estimated as equal to 10000 plasma discharges with averaged FW heat load of the order of 0.2 MW/m2.

The main functions of the TRT divertor are to accept energy and particle fluxes from plasma, to clean the plasma from impurities produced due to the FW erosion and to provide efficient pumping of the internal volume of vacuum vessel. Due to essentially higher power loads of the order 10-15MW/m2 to the divertor targets compared to those to FW the life time of the TRT divertor is estimated as 5000 full power plasma discharges.

For both FW and divertor the staged approach in design options correlating with the increase of TRT power will be realized. For the first stage FW and divertor designs are based on knowledge and experience obtained during the ITER project development: Be and W will be used as plasma facing materials in FW and divertor, respectively. To be ready for usage liquid lithium on next stages the stainless still tubes and components will be used in water cooling systems for FW from the very beginning and for divertor from second stage.

Then other options including the recent developments of the concept of the swept divertor target with a liquid metal interlayer between the moving armour and motionless heat-sink suggested in [12] should be explored at later stages of operation. Liquid Li protection of the plasma facing elements is one of the principal goals of TRT concept. Various variants of protection including lithium-filled capillary porous system (CPS) are discussed to accommodate the best achievements of the T-11M and T-10 lithium experiments as well as expected contribution from the T-15MD research program. The arrangement of TRT first wall and divertor are shown in Fig. 3.

156 FW modules will be arranged in 10 rows as shown in Fig. 3. Calculated maximum heat loads of the order of ~ 1 MW/m2 expected for rows 1, 6, 8, 9 and ~4 MW/m2 for rows 2, 3 (placed 50 mm from separatrix and play the role of a limiter at the beginning of plasma discharge), 4, 5 (quasi-divertor zone), 7, 10 (placed 70 mm from separatrix and acting as protection limiters). Rows 1 and 10 provide narrow divertor opening to main plasma. Technology for FW module adjustable assembling on preliminary welded to vacuum vessel mechanical structure was developed. FW water cooling system ready to accept 34 MW heat loads have been designed. Thermal analysis demonstrated that TRT water cooling system should keep the plasma facing Be component temperature at the level of about 5000C and never reach limiting 6000C during maximum power TRT operation.

As shown in table 1 due to high auxiliary heating power and compact size TRT will operate at very high values of Psep/R and BtPsep/R for both long time deuterium and < 5 or 10s (depending upon DT neutron yield) short deuterium-tritium regimes of operation. This makes TRT the excellent experimental plant for development of the fusion reactor divertor technology. Experiments with separatrix sweeping by means of additional poloidal coils that are planned for installation in a specially reserved space inside the divertor region is under analysis now. Application of liquid lithium technologies in TRT is under study. Continuous increasing amount of lithium from stage to stage from 100g up to > 1kg is foreseen.

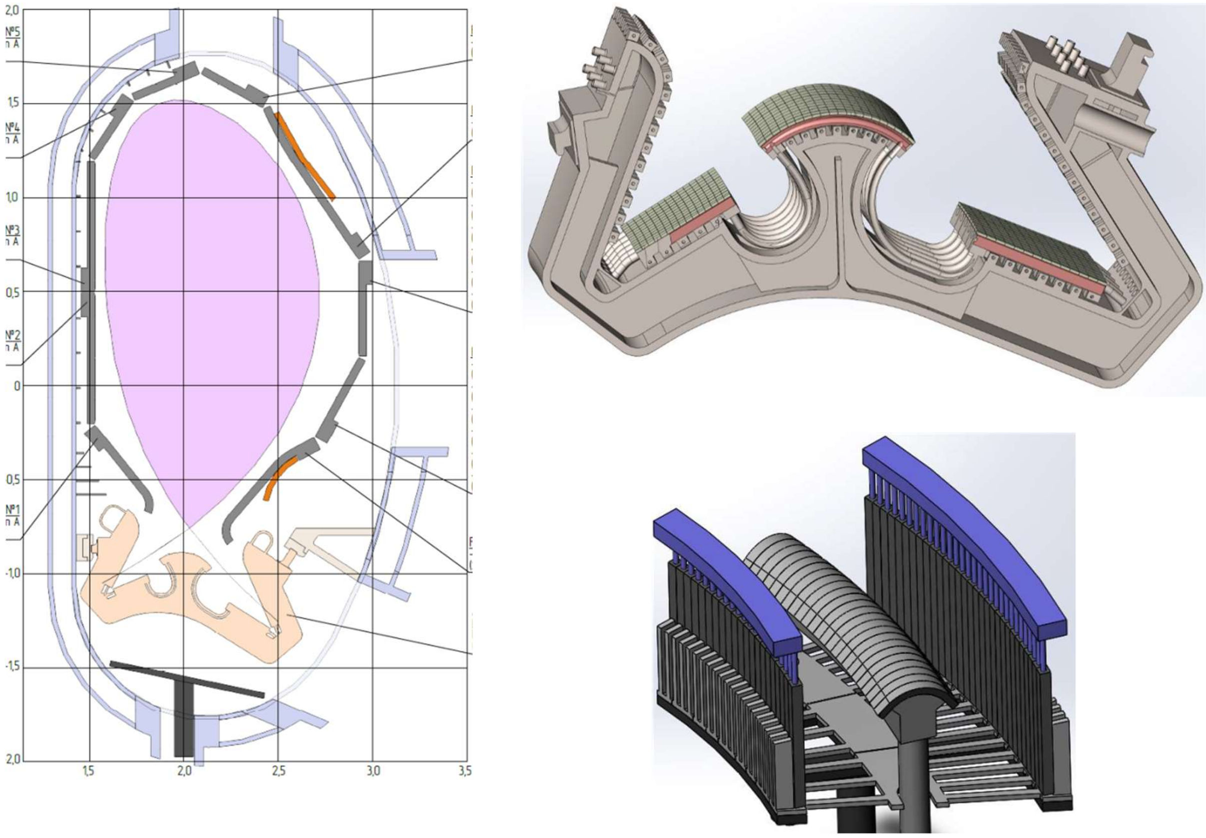

FIG. 3. *TRT FW (left) and divertor ITER-like (up) and alternative (down)*

TRT auxiliary heating and current drive system comprises negative ion based neutral beams of 25MW with beam energy 300-500keV, ECRH system with 10-12 gyrotrons with frequency 230 (260) GHz and total power of 10MW and ICRH system of ~5MW with frequency range of 60 - 80 MGz. As additional option traveling way antenna to launch helicon waves of 1-1.2 GHz of few MW total power is considered. TRT VV has 3 specially designed equatorial ports to provide tangential neutral beam injection with R_{tg} = R0-a/2 = 1865mm. 6 injectors (2 for each ports) are designed in Budker Institute. Gyrotrons for frequency 230GHz are designed in Applied Physics Institute (Nizniy Novgorod) with use of technologies and experience gained in development of the 177GHz gyrotrons for ITER. EC waves from 12 gyrotrons in TRT should enter the plasma through the one equatorial port. Multi-mirror system in the port should provide online control on the EC power deposition profile. The synergetic effect between NB and EC heating and CD is the most promising in development the algorithms for the kinetic control of the plasma.

4. OPERATION REGIMES OF THE TRT

Analysis of the plasma initiation and current ramp up stages revealed that projected maximum plasma current of Ip=5MA value can be reached within <30s with use of the ECRH. 4-5Wb of the inductor magnetic flux remaining after the current ramp up was found to be sufficient to provide projected pulse duration of order 100s at high plasma density of ne~$2*10^{20}m^{-3}$ if full auxiliary power of P_{aux}=35MW is used. However, further analysis shown that high current, high density regime is hardly compatible with ITER-like divertor design. Decreasing the plasma density to the $1*10^{20}m^{-3}$ and plasma current to 4MA opens wide operation domain to explore stationary (fully noninductive) scenarios with reactor relevant plasmas.

Important finding of the TRT scenario development is that for 50/50 DT plasma with P_{LH}~15MW< P_{aux}<30MW fusion gain factor Q~1 achieved at moderate plasma densities $0.5<n_{e20}<1.5*10^{20}m^{-3}$ (see Fig.4 - left). I.e. even at the lower limit of P_{aux}=15MW which close to the threshold LH transition power, TRT can operate at ~0.17MW/m2 of neutron load, which is sufficient for the FNS in the fusion-mission hybrid reactor.

NBI accompanied by ECRH with controllable location of the deposited power was shown to provide steady-state MHD stable 4MA scenarios. In turn, preliminary plasma edge stability analysis resulted in optimistic predictions on ELM regimes in expected operation scenarios. Beam driven Alfven instabilities according to preliminary analysis can be dangerous for the fast ion confinement in TRT and should be accounted for in the future optimization of operation scenarios.

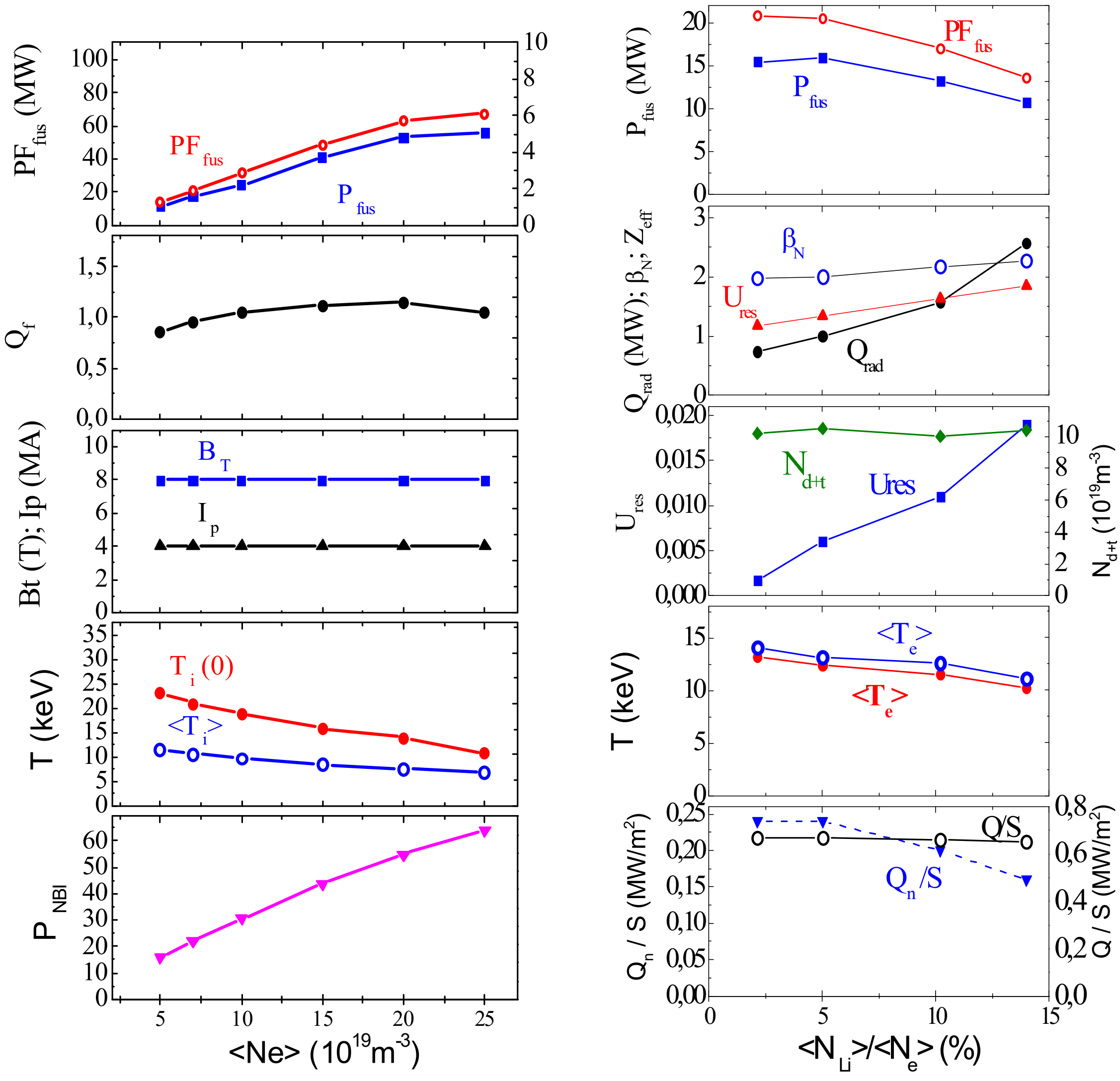

FIG. 4 *TRT plasma performance at different plasma density (left) and Li concentration (right). P_{fus} and PF_{fus} at the top pannels stand for fusion reactivities without and with allowing for the beam - target reactions, respectively*

The presence of Li as a main plasma impurity was found to be tolerable up to the concentration of about n_{Li}~15% (see Fig.4 - right). Neutron flux necessary for FNS with rise of the Li concentration can be kept constant by the corresponding increase of the DT mixture density. In simulations Li came to the plasma from the wall. Then further increase of the Li contamination was accompanied by overcooling of the plasma edge, contraction of the plasma current and deterioration of the MHD stability.

As a whole preliminary analysis of the TRT operation scenarios manifested that technical characteristics of TRT altogether with its auxiliary heating and current drive systems open wide operation window for experiments with really reactor relevant plasmas.

5. SUMMARY

Conceptual design of the Tokamak with Reactor Technologies - TRT revealed that compact (R/a=2.15/0.57) tokamak with high (B_t=8T) magnetic field with HTS REBCO electromagnetic system, powerful auxiliary heating/CD complex and advanced strategy in FW and divertor technology development opens the unique possibility to integrate most important technologies both for pure fusion and hybrid fusion-fission reactor in a one single machine. Expected discharge characteristic in TRT allow wide operation window to explore steady MHD stable scenarios with (for DT mixture) fusion gain factor of order unity providing neutron flux to the wall > 0.2MW/m^2, which is sufficient for the fusion neutron source in the fusion-fission hybrid reactor. TRT experimental program with hydrogen, helium and deuterium plasmas is aimed to develop the stationary operation regimes necessary for the economically attractive reactor, i.e. the regimes with reliable controllability of the plasma parameters including their radial profiles and keeping plasma wall interaction at the constant tolerable level.

ACKNOWLEDGEMENTS

The work was supported by State Atomic Energy Corporation "Rosatom", agreement $N^0$313/1671-D from September 5, 2019.

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