# The concept of Tritium Fuel cycle for a Tokamak based fusion neutron sources in the Russian Federation

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**Abstract**

The paper describes contemporary development results and novel applications of the integrated model for simulating processes in the plasma fueling cycle systems (FC) of DEMO-FNS tokamak. Interaction of FC with core and divertor plasmas employs the SOLPS, ASTRA, and FC-FNS codes. The range of operation parameters is found for the core plasma, in which the tritium fraction is controlled by pellet injection with a different T/D isotopic composition. Impact of neutral beam heating and fueling by recycling and neutrals flow from divertor was accounted for as well. The neutral beams with different D/T composition and the core plasma were studied, as well as the total inventory at the facility site and the localization of tritium in FC. The influence of HFS and LFS pellet injection on the plasma core fueling and ELM triggering in the DEMO-FNS is considered. These results will be used for further optimizing the FC of the FNS-C neutron source and hybrid reactor facility HRF, which are considered as a part of the comprehensive program of the State Corporation Rosatom "Development of engineering, technology and scientific research in the field of using atomic energy in the Russian Federation for the time period up to 2030" and further up to 2040.

1. INTRODUCTION

The National Research Center "Kurchatov Institute" is planning to carry out a broad applied research within the framework of the federal project "Development of technologies for controlled fusion and innovative plasma technologies".

The current period tasks include substantiation of DT-fuel cycle and choice of hybrid blanket technologies, as well as engineering design of the FNS-K fusion neutron source and hybrid reactor facility (HRF) for testing steady state technologies, materials and components [1].

Timely provision of specialized stands and qualified personnel will make it possible to effectively implement the project for the construction of a HRF with the thermal power of up to 700 MW.

A mockup facility of the tritium-deuterium fuel cycle (FC) should be designed, developed and put into operation for research aimed at determining and clarifying the system operational parameters, productivity, amount of tritium at the site and other characteristics of FC systems supporting the projects DEMO-FNS [2] and FNS-ST [3] as the basis for the design of FNS-C and HRF. It is planned to justify the safety of technological systems and train personnel for the operation of these systems in the FFHS with tritium storage at the facility site from 0.1 to 2.0 kg. At later stages, engineering projects of the FC systems for FNS-C and HRF should be developed, mock-ups are to be produced and their joint operation carried out, in accordance with the research program. Research and selection of the optimal technologies for tritium breeding and extraction in blankets, improvement and optimization of specific fuel cycle technologies, justification of the radiation safety of the FC-facility.

Calculations of hydrogen isotope fluxes in the facility systems have been carried out to the architecture of fuel systems was chosen, being based on the results obtained, Candidate technologies for operation with tritium were selected, and complex for neutral beams injection were developed [4-7]. This report is devoted to a description of the recent results for D-T FC of the FNS-ST and DEMO-FNS projects under various plasma scenarios.

2. NEUTRON FLUX AND TRITIUM INVENTORY VS D/T- COMPOSITION OF THE CORE PLASMA AND NEUTRAL BEAMS

The injection of beams of fast atoms of hydrogen isotopes into the tokamak plasma considerably affects the plasma isotopic composition and the neutron yield [4, 5, etc.]. For FNS-C/FNS-ST compact tokamaks, the effect of the beam composition on the neutron yield is most significant [8]. Therefore, the NBI system is considered currently as the key subsystem of the FC. Accordingly, the self-consistent calculations of the fueling hydrogen isotope fluxes in the core and divertor plasmas and FC systems are performed using the developed FC-FNS electron model.

Simulation of FNS-ST operating modes with heating by D+T-, D- or T-injectors 6 MW for various values of the main plasma density and ion diffusion coefficients made it possible to find the neutron flux as a D+T fusion reaction function (accounting for the beam fraction) vs the tritium fraction in the plasma core. The fuel flow components in the facility fuel cycle systems were calculated by SOLPS, ASTRA, and FC-FNS codes, being provided by various injection systems - pellet injectors, fast atom beams, and gas puff to the vacuum chamber. It was shown that the plasma core fueling by neutral beams and neutral fluxes from the divertor is not sufficient to maintain the required plasma density level.

In the window of working parameters the plasma core density *ne* and particles diffusion coefficient *D/χе* its D/T composition are calculated, at which the maximum neutron yield is expected. It is shown that with the chosen approach to the fuel cycle architecture (with partial gas exhaust separation from the vacuum chamber – 20-60%) for all considered beam regimes, the neutron yield is limited by the plasma core isotopic composition that depends on the divertor plasma composition. An approach was considered for controlling the isotopic composition in a divertor plasma using monoisotopic gas puffing into the vacuum chamber being based on the hydrogen isotopes separation of full exhaust gas flow pumped out from the divertor.

It was shown that for the D+T beam the neutron yield of 4,5-5,5·1017 1/s is achievable at the plasma core density *ne*= 7,0-8,5·1019 1/m3 and *D/χе*= 0,2-0,4 - that is good for particles retention in the plasma. For the D-beam, the highest neutron yield 4,5-5,5·1017 1/s is achieved in the density range *ne*= 7,0-10,0·1019 1/m3 in modes with *D/χе*= 0,4-0,6. An increase of the neutron yield of 4,5-6,0·1017 1/s occurs for the T-beam together with a decreasing the T fraction in the plasma core, that is, *ne*= 9,0-10,0·1019 1/m3 and *D/χе*= 0,4-0,6.

For the proposed new approach to the FC architecture (with separation of the entire gas flow evacuated from the vacuum chamber) the highest neutron flux (up to 6,5·1017 1/s) for T-beam corresponds to the region of higher density and better confinement (*ne* > 8,5·1019 m-3 and *D/χе*< 0,4). Application of this approach to the pure D-beam makes it possible to raise the neutron yield up to 7,0·1017 1/s. In this case, the region of lower density and worse confinement (*ne* < 8,5·1019 m-3 и *D/χе*> 0,3) corresponds to the highest values of the neutron flux.

The required Т inventory at the facility site (in the fuel cycle system) for Т- and D-beams was calculated for regimes with the maximum neutron yield. The T2 total amount in the FC of the facility will be in all considered cases in the range of 150-250 g, but it will be distributed differently between the FC systems (at the facility site) - see Table 1. FC systems optimization with the goal to reduce the Т inventory is a promising task that is not yet fully resolved to date.

TABLE 1. TRITIUM LOCATIONS IN FC SYSTEMS AND THE TOTAL INVENTORY FOR VARIOUS NBI GAS SUPPORT SCENARIOS (in gram)

|  |  |  |  |  |
| --- | --- | --- | --- | --- |
| Beam isotope composition | Tritium plant, g | NBI-system, g | Fueling systems and vacuum chamber, g | Total inventory, g |
| D+T (current FC architecture) | 25 | 20 | 180 | 180-220 |
| D (current FC architecture) | 30 | 0 | 160 | 160-190 |
| D (full exhaust separation) | 35 | 0 | 180 | 215 |
| T (current FC architecture) | 40 | 40 | 40 | 145-170 |
| T (full exhaust separation) | 100 | 40 | 30 | 175 |
| T (NBI gas support optimisation) | 165 | 15 | 30 | 215-220 |

For example, with separate gas support for the ion source (T2) and neutralizer (D2), the T inventory in the hydrogen isotope separation system will increase up to 160 g (relative to the variant with T2 inlet into all elements of the injector and partial gas separation to remove the D2 impurity). This leads to an increase of the T2 inventory in the FC by 40 g within the working window. Meanwhile, such procedure reduces the T2 inventory in injectors by a factor of three. This should be taken into account in comprehensive assessment of various options for the gas isotopic composition choice in heating injectors. It should be noticed that in order to make a balanced decision implementing a particular scenario the T2 inventory in the FC is not the only criterion. We should also take into account several subsystems with significant T2 inventory, as well as technological difficulties in implementing the T-beam injection, operating the additional heating system and etc.

3. FUEL COMPONENT FLUXES IN DEMO-FNS FUEL CYCLE SYSTEMS AND TRITIUM INVENTORY WITH ALLOWANCE CONVECTIVE ELMS CONTROL

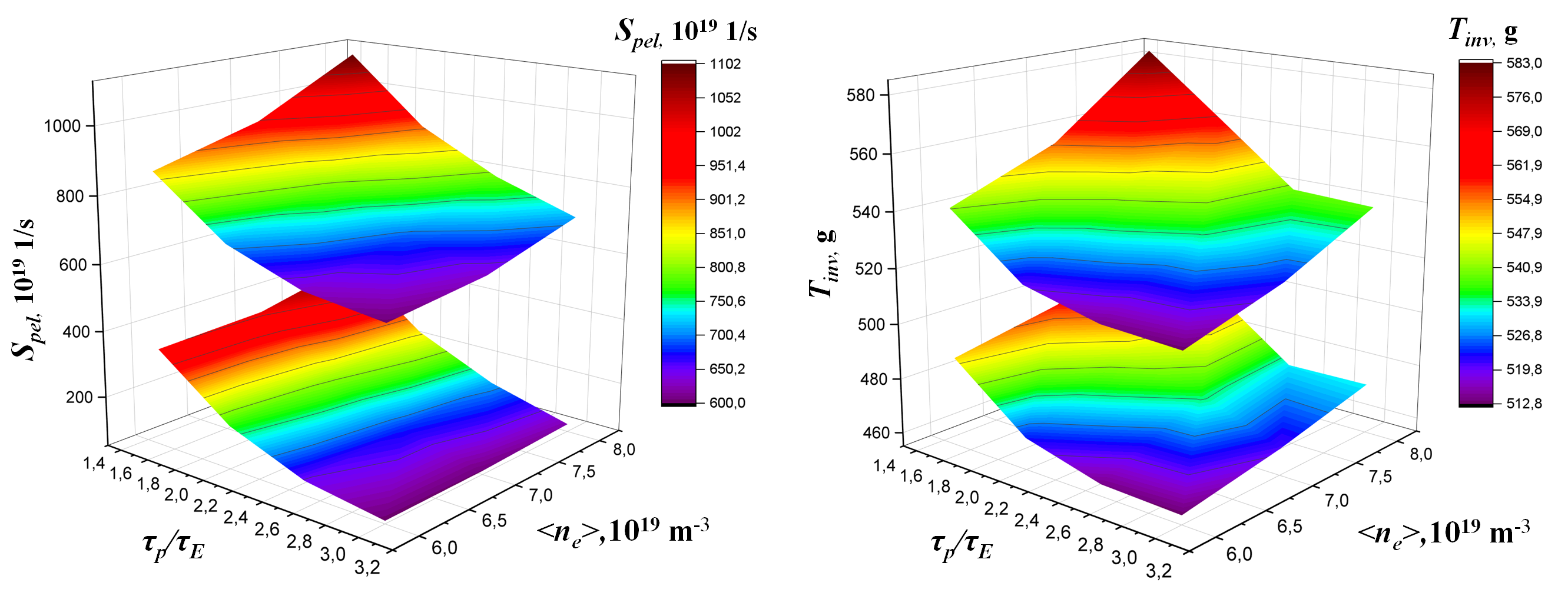
In [5] we considered the effect of pellet injection (from the HFS for the core plasma fueling and the LFS for driving ELMs) on the fluxes of D and T particles in the FC systems of DEMO-FNS. The further development is the implementation of ions, instead of electrons, in particle transport equations in ASTRA. This allows a more accurate estimation of the partial confinement times for ions originating from different sources (NBI, pellets, gas puff, recycling) forming FC-FNS.

As a result of simulations performed using the ASTRA + SOLPS and FC-FNS codes, it was shown that in the parameter ranges 1.5 < τ*p*/τ*E* <3.0 and 〈*ne*〉 = 6.0–8.0 × 1019 m–3, the injection of pellets of different isotopic composition can provide the required fraction of tritium = 0.5 in the core plasma of DEMO-FNS. In the operating window of plasma parameters, the fluxes of the fuel components were analyzed as functions of the τ*p*/τ*E* and 〈*ne*〉. For the considered scenarios the neutron flux turns out to be about 1.2·1019 1/s.

It is shown that taking into account the convective ELMs (the increased particle flux D+T to plasma core as a source from pellets *- S*pel rises), the required capacities of the systems for pellet injection and hydrogen isotopes separation (and some other FC systems) increased by up to an order of magnitude as compared to the estimates made previously.

Calculations of the hydrogen isotope inventories in the FC systems showed that the starting inventory (including reserve supplies) should be from 500 g (for the D beam) to 800 g (for the D+T beam) of T2 and up to 850 g of D2. Figure 1 shows the total particle fluxes to plasma core as a source from pellet, as well as the tritium inventory in FC systems for natural and convective ELMs cases, given for the D-beam.

The reserve supplies required to be able to work temporarily with the tritium breeding systems closed (for repair or maintenance) make 80–120 g of T2. The long-term storage should hold up to 350 g of T2 (TBR = 1.2, extraction of accumulated tritium once a year). Therefore, all systems of the facility will contain from 900 (for D-beam) to 1200 g of T2, taking into account the decay of accumulated T and including tritium in the long-term storage.



*FIG. 3. Total particle fluxes D+T to plasma core as a source from pellets Spel, as well as the tritium inventory Tinv in FC systems for natural and convective ELMs cases, given for the D-beam.*

Note that an increase in the fuel fluxes through the injection systems, hydrogen isotopes separation systems (and some other systems) in the case of the convective ELMs causes no noticeable increase in the T inventory in the fuel cycle. The hydrogen isotopes inventory at the facility site in this case will reach values from 1000 (for D-beam) to 1400 g T2 and up to 1400 g D2.

3. CONCLUSION

Concept of DT-FC is developed in RF for tokamak based fusion neutron sources. Analysis of the FC operation is performed on the basis of advanced SOLPS, ASTRA, and FC-FNS codes taking into accounting for interaction of gas flows with plasma. Effects of specific gas mixture flows on plasma optimization and minimizing the T inventory at the site are treated in details.

For the considered cases of heating beam isotope composition, corresponding to operation conditions of FNS, a neutron flux up to 6,5·1017 1/s can be received for the compact tokamak based neutron source FNS-C. The tritium inventory in the fuel cycle systems of the facility in all considered cases will be in the range of 150-250 g, but will be distributed differently between the systems at the on-site. Choosing the plasma core and heating beam composition, it is probably necessary to minimize the tritium inventory in the vacuum chamber and injection systems adjacent to it - in this case, an increase in its inventory in the heating beams gas supply system and hydrogen isotope separation system can occur. The tritium inventory in the beam injectors can be reduced by various gas compositions introducing to the ion source and neutralizer. This will lead to an additional increase of the tritium inventory in the "tritium plant", however, taking into account that it will be located in a separate building on site, such decision may well be justified.

Calculations performed for core plasma with ions equations instead of electrons showed smaller values of fuel particle fluxes for plasma fueling. This changes the balances in plasma by particle sources and reduces the performance requirements of injection systems. For the considered plasma core scenarios, the neutron flux turns out to be about 1.2·1019 1/s for DEMO-FNS tokamak based facility. The starting inventory on site (including reserve supplies) should be about 500 g of T2 and up to 850 g of D2 for the D-beam. In the case with allowance convective ELMs control at the facility will contain up to 900 g of T2 for D-beam, taking into account the radioactive decay and including in the long-term storage. Further development should be the profiles calculation from pellets ablation in plasma, as well as FC systems optimizing to reduce tritium inventories and the fuel mixture processing time.

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