# 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 (FC) systems of tokamak based fusion neutron source (FNS). The integrated model of the 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 the neutral flow from the divertor is allowed for as well. The neutral beams with different D/T composition and the core plasma are studied, as well as the total inventory at the facility site and the localization of tritium in the FC. The influence of HFS and LFS pellet injection on the plasma core fueling and ELM pacemaking in DEMO-FNS is considered. These results will be used for further optimization of 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

One of the scenarios for the nuclear power development in the Russian Federation is associated with an open uranium-thorium (233U - 232Th) fuel cycle. This scenario was also considered from the beginning of the nuclear energy development, but its practical implementation was hampered by the absence of external neutron sources, since there are not enough neutrons to produce 233U within the uranium-plutonium cycle. If the functions of such source are performed by the fusion neutron source (FST), then such a concept will receive an undeniable advantage. In this scenario, the development of nuclear energy will be accompanied by the creation of reactors with thermal neutrons (TR) [1].

The main advantage of this approach is that there is no need to separate SNF for fuel nuclides extraction - spent fuel elements will be removed from the reactor core and sent for long-term storage. In the uranium-thorium cycle, chemical processing must also be carried out to separate 233U from the raw nuclide. Due to the significantly lower activity of the fusion-fission hybrid system (FFHS) blanket contents, such reprocessing can be accompanied by significantly lower losses of radioactive materials - due to the complete absence of minor actinides and an insignificant fraction of fission products compared to SNF from fast reactors (FR). In view of the fact that the Th reserves are much greater than the reserves of natural U, such a concept will ensure a longer development of nuclear energy.

The tritium (3H) production as a FNS fuel can be provided in thermal reactors, where the 6Li isotope will be used as a burnable absorber. When optimizing the TR core, it is possible to ensure the production of a sufficient tritium amount to close the TR+FNS tandem fuel cycle for fusion fuel.

The development of the FFHS is being actively discussed at the National Research Center "Kurchatov Institute" for over 10 years. These R&D work was associated with fusion neutron source technical design and neutrons generation optimization in FFHS and the use of these neutrons for nuclide transmutation and breeding. It should be noticed that the tritium breeding in FFHSs for start-up loading of future thermonuclear power plants opportunities were considered also.

Current views including those reflected in the Federal project “Development of Fusion and Innovative Plasma Technologies” (which is 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") foresee the feasibility of industrial FFHSs by 2050.

2. TRITIUM FUEL CYCLE FOR A FUSION NEUTRON SOURCE DEVEPLPMENTS

The National Research Center "Kurchatov Institute" is planning to carry out a broad applied research within the framework of this federal project. Previously, the conceptual design of the FNS fuel cycle was carried out. The fuel cycle design for FNS was selected and described in detail, and the differences from fusion reactor were analyzed [2, 3]. Calculations of hydrogen isotope fluxes in the facility systems was carried out for the chosen fuel systems design, based on the previous results obtained. Candidate technologies for operation with tritium are selected, and a complex for neutral beams injection (NBI) was developed [2-5]. The current period tasks include substantiation of the DT-fuel cycle and the choice of hybrid blanket technologies, as well as the engineering design of the FNS-С compact fusion neutron source and hybrid reactor facility (HRF) for testing the steady state technologies, materials and components [6]. Timely provision of specialized test benches and qualified personnel will make it possible to implement efficiently the project for the construction of a HRF with commercially interesting thermal power.

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 the FC systems supporting the DEMO-FNS [7] and FNS-ST [8] projects as the basis for the design of FNS-C and HRF. It is planned to justify the safety of the technological systems and to train personnel for the operation of these systems in the HRF with the tritium inventory at the facility site up to 1 kg. At later stages, the engineering projects of the FC systems for FNS-C and HRF should be developed, the mock-ups 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 will be done.

This report describes the recent results for the FC of the FNS-ST and DEMO-FNS projects under various plasma scenarios.

3. 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 [2, 3, etc.]. For the FNS-C/FNS-ST compact tokamaks, the effect of the beam composition on the neutron yield is most significant [9]. 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 FC-FNS computer model developed.

Simulation of FNS-ST operating modes with heating by D+T, D or T injectors with 6 MW for various values of the main plasma density and ion diffusion coefficients makes it possible to find the neutron flux (including the beam-plasma reactions) as a function of the tritium fraction in the plasma core. The fuel flow components in the facility fuel cycle systems are calculated by the SOLPS, ASTRA, and FC-FNS codes, being provided by various injection systems - pellet injectors, fast atom beams, and gas puff into the vacuum chamber. It was shown [9] 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 working window of the plasma core density *ne* and particle to heat diffusivity ratio *D/χе*,the D/T composition of the plasma is calculated, at which the maximum neutron yield is expected. It is shown that with the chosen approach to the fuel cycle architecture (with partial separation of the gas exhausted from the vacuum chamber – 20-60%), for all considered beam regimes, the neutron yield is limited by the isotopic composition of the plasma core, which depends on the divertor plasma composition. A new approach is proposed for controlling the isotopic composition in the divertor plasma using monoisotopic gas puffing into the vacuum chamber, which relies on the hydrogen isotopes separation in the whole exhaust gas flow pumped out from the divertor.

It was shown [9] 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, at good particles retention in the plasma. For the D beam, the highest neutron yield of 4.5-5.5·1017 1/s is achieved in the density range *ne*= 7.0-10.0·1019 1/m3 if *D/χе*= 0.4-0.6. An increase of the neutron yield to 4.5-6.0·1017 1/s occurs for the T beam by 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 the 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 is calculated for the regimes with the maximum neutron yield. The total T2 amount in the FC of the facility is, in all the cases considered, in the range of 150-250 g, distributed unevenly among 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 fully resolved yet.

TABLE 1. TRITIUM INVENTORY IN DIFFERENT FC SYSTEMS AND THE TOTAL INVENTORY FOR VARIOUS NBI GAS SUPLY 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 feed optimization) | 165 | 15 | 30 | 215-220 |

For example, with separate gas feeds 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 feed to all injector components 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 a procedure reduces the T2 inventory in the injectors by a factor of three. This should be taken into account in comprehensive assessment of various options for the choice of the isotopic gas composition in the heating injectors. Note that for making a balanced decision on implementation of 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.

4. FUEL COMPONENT FLUXES IN DEMO-FNS FUEL CYCLE SYSTEMS AND TRITIUM INVENTORY WITH ALLOWANCE FOR ELM PACEMAKING

In [3] 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 consideration of ions, instead of electrons, in the particle transport equations in ASTRA. This allows a more accurate estimation of the partial confinement times for the ions originating from different sources (NBI, pellets, gas puff, recycling).

As a result of simulations performed using the ASTRA + SOLPS and FC-FNS codes, it is shown that in the parameter ranges of 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 are 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 the plasma core as a source from pellets *S*pel rises), the required capacities of the systems for pellet injection and hydrogen isotope 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 show that the initial inventory (not including the 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 the plasma core as a source from pellets, as well as the tritium inventory in the FC systems for the natural and driven ELMs cases, given for the D beam.

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



*FIG. 3. Total, D+T, particle fluxes to the plasma core as the source from pellets Spel, as well as the tritium inventory Tinv in FC systems for the natural (down) and driven (up) ELM 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 isotope inventory at the facility site in this case will reach the values from 1000 g (for the D beam) to 1400 g (for the D+T beam) of T2 and up to 1400 g D2.

5. CONCLUSION

In the Russian Federation, a concept is being developed for the AE development with a fusion neutron source based on a tokamak for a fuel nuclide breeding in a FFHS blanket and "closing the fuel cycle" of thermal reactors (in the 233U - 232Th fuel cycle). The main advantage of this approach is that there is no need to separate SNF for fuel nuclides extraction. Due to the significantly lower activity of the fusion-fission hybrid system blanket contents, such concept can be accompanied by significantly lower losses of radioactive materials compared to using fast reactors and processing SNF from them. be provided in thermal reactors, where the 6Li isotope will be used as a burnable absorber.

It was shown that in order to create an energy efficient combination of TR+FFHS, it is necessary to ensure the fuel nuclide production in an amount that is sufficient for the operation of the TR and also to reduce the electricity consumption for the FFHS (taking into account the thermal energy produced in the blanket to electricity conversion). To do this, R&D should be carried out to intensify the neutrons source production and the fuel nuclide breeding in blanket. These studies are carried out within the federal program framework "Development of technologies for controlled fusion and innovative plasma technologies".

The concept of the DT fuel cycle for the tokamak-based fusion neutron sources is being intensively developed in the Russian Federation. Investigation of the neutron source operating modes and the fusion plasma isotope composition optimization is the most important task from the facility operation safety point of view. Analysis of the FC operation is performed on the basis of the advanced SOLPS, ASTRA, and FC-FNS codes, taking into account the interaction of the gas flows with the plasma. Effects of specific gas mixture flows on plasma optimization and minimizing the T inventory at the site are treated in detail.

For the considered cases of the heating beam isotope composition, corresponding to the operational conditions of the FNS, a neutron flux of up to 6.5·1017 1/s can be expected in the compact tokamak-based neutron source FNS-C. The tritium inventory in the fuel cycle systems of the facility in all the cases considered is in the range of 150-250 g, distributed unevenly among the systems on site. When choosing the plasma core and heating beam composition, it is desirable to minimize the tritium inventory in the vacuum chamber and injection systems adjacent to it. In this case, an increase in the T inventory in the heating beams gas supply system and the hydrogen isotope separation system can occur. The tritium inventory in the beam injectors can be reduced by separate feeds of gas with different composition for the ion source and neutralizer. Although this will cause an additional increase of the tritium inventory in the "tritium plant", such a decision may well be justified taking into account that it will be located in a separate building on the site.

Calculations performed for the core plasma with particle transport equations for ions instead of electrons shows lower values of the fuel particle fluxes for plasma fueling. This affects particle balance in the plasma and reduces the performance requirements on the injection systems. For the plasma core scenarios considered, the neutron flux is about 1.2·1019 1/s for the DEMO-FNS tokamak-based facility. The initial inventory on site (including the reserve supplies) should be about 900 g of T2 for the D beam. In the case with ELM pacing it increases up to 1000 g of T2 for the D beam, taking into account the radioactive decay and including the long-term storage. Further development should be the profile calculation from pellet ablation in the plasma, as well as optimization of the FC systems to reduce the tritium inventories and the fuel mixture processing time.

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