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Development of Technologies for the Fusion-Fission Hybrid Systems

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and the group of developers of the hybrid fusion-fission system

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Second Technical Meeting on Long Pulse Thermonuclear Device Operation

TOPIC : RAMI (Reliability, Availability, Maintainability, Inspectability) and Nuclear Technologies for Long Pulse Operation

Participants

Specialists from the following organizations participated in the program for developing hybrid fusion-fission systems.

- NRC “Kurchatov Institute”, Moscow, Russia
- Efremov Institute, Sankt-Petersburg, Russia
- Bochvar Institute, Moscow, Russia
- NIKIET, Moscow, Russia
- Peter the Great SPbPU, St. Petersburg, Russia

Introduction

Two ways of fusion energy development

- 1. Creation of a fusion power plant
- 2. Creation of a hybrid fusion-fission system

At this meeting, most of the projects presented relate to the first way.

This work presents the second direction.

Fusion device is not used as a source of energy but mostly as a a source of neutrons.

Neutrons can be used for different purposes.

Biology, medicine, materials, <u>nuclear fuel production</u>, transmutation of nuclear energy waste, energy from nuclear blanket, others.

In our case - <u>nuclear fuel production</u> from natural transuranic isotopes.

Hybrid fusion-fission system includes fusion neutron source (FNS) and nuclear blanket.

Introduction

- Environmentally acceptable nuclear energy is an important component of the world's energy system today.
- Difficulties associated with the production of nuclear fuel, processing and disposal of radioactive waste from fission reactors limit its development.
- Integration of technologies can ensure long-term and sustainable development of the world energy system.
- One of the most promising ways to solve these problems is fusion neutron sources (FNS), which use the fusion reaction of hydrogen isotopes: D and T.
- Developing a hybrid fusion-fission reactor system gives the opportunity of using fusion neutron fluxes to produce fuel nuclides, such as U-233 (Thorium-Uranium nuclear fuel cycle).

$$Th^{232} + n_0 \rightarrow Th^{233}\ (\tau_{½} = 22\ \text{min}) \xrightarrow{\beta^-} Pa^{233}\ (\tau_{½} = 27\ \text{days}) \xrightarrow{\beta^-} U^{233}$$

Introduction

- The project of hybrid facility tokamak is aimed at achieving quasi-steady-state operating modes with a load on the first wall of up to ~0.2 MW/m^2 neutrons with an energy of 14 MeV and a neutron fluence during the life cycle of the facility of ~2 MW • year/m^2 (10 years of continuous operation).
- Hybrid technologies developed by the National Research Center "Kurchatov Institute" includes: development of experimental compact neutron source (FNS-C), including the development of technologies and testing of materials.
- The development of prototypes of various versions of a nuclear blanket, containing raw and nuclear materials.
- In connection with the expected results of experiments on the T-15MD tokamak, it seems appropriate to consider the possibility of creating on the scale of this facility.

Modern activities

- Design tasks for the period 2021-2024 are focused on developing new plasma modeling tools and discharge scenarios, improving the performance of key systems, and integrating FNS components that implement upgraded and new technical solutions.
- These include the first wall, divertor, neutral injection system, ECR heating and current drive, heat exchange, thermonuclear and nuclear fuel cycles, lithium technologies [1, 2].
- **Important**: hybrid system produces less minor actinides and radioactive waste than nuclear reactors.
- The carried out economic assessments showed that cost of fissile nuclides produced using a hybrid blanket could potentially be cost-effective.

REFERENCES

[1] Yu.S. Shpanskiy , B.V. Kuteev . Development of basic thermonuclear technologies of the fusion-fission hybrid facility for testing materials and components // In Program, Book of Abstracts & Conference Materials of the 29th IAEA Fusion Energy Conference, London, Great Britain, 16-21 October 2023. p. 694.

[2] E. Dlougach , M. Kichik , Beam Transmission (BTR) Software for Efficient Neutral Beam Injector Design and Tokamak Operation. Software 2023, 2, 476-503 https://doi.org/10.3390/software2040022

PROJECT PURPOSE

Implementation of an R&D program justifying the construction of a hybrid reactor facility containing a hybrid blanket based on a stationary tokamak

- with a DT fusion power of more than 30 MW (this corresponds to the generation of ~$1x10^{19}$ neutrons/s)
- fission power up to 500 MW.

Development of the effective technologies for fuel supply for nuclear power engineering based on fission reactors that use the conversion of the natural isotop Th-232 into fissile U-233 by a flow of fusion neutrons.

To implement this task, the following is planned:

- development of basic technologies for a tokamak with a hybrid blanket (including lithium ones),
- development of a set of experimental facilities,
- development of an experimental compact neutron source FNS-C for testing materials and components of hybrid systems.

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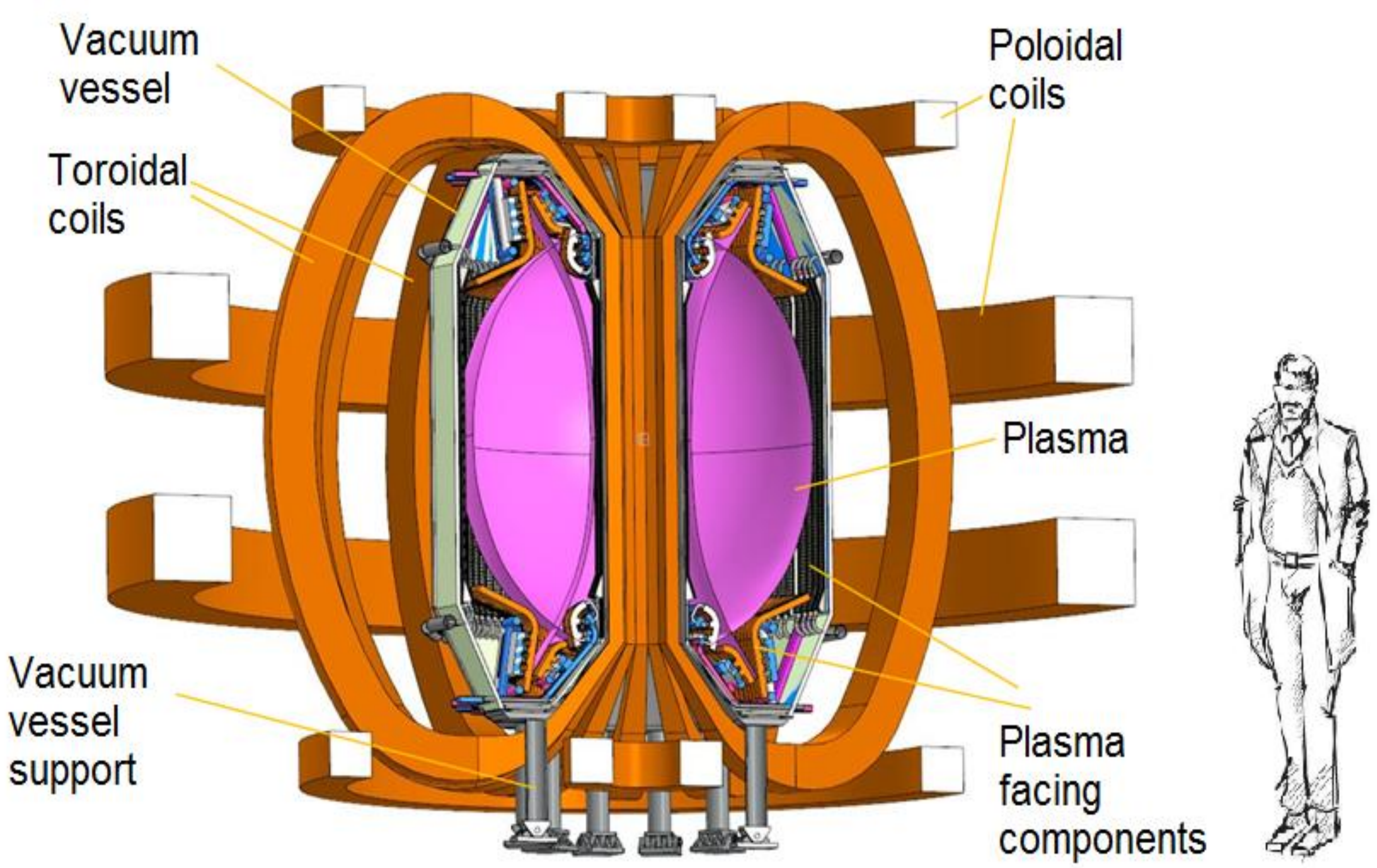
General view of the compact tokamak TIN-K (FNS-C) fasility (2023)

The purpose of TIN-K (R= 0.53 m; a = 0.3 m) is to develop technologies and study the behavior of materials under conditions of neutron load up to 0.25 MW/m^2



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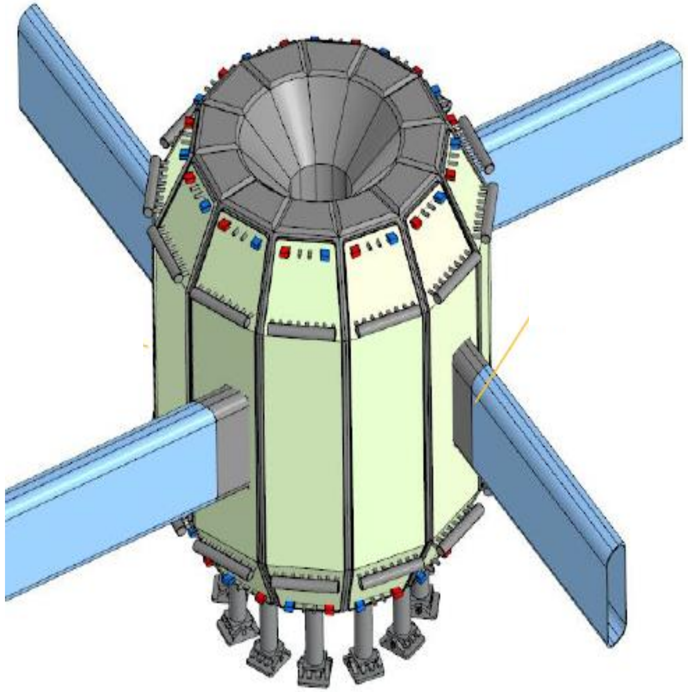
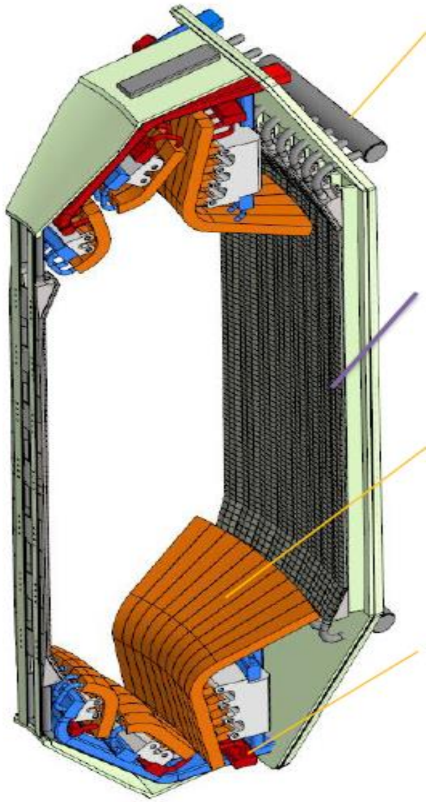
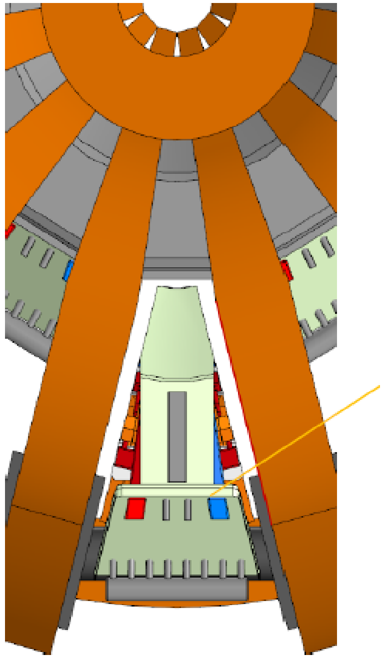
Components of TIN-K (FNS-C)



Vacuum vessel with NBI ports

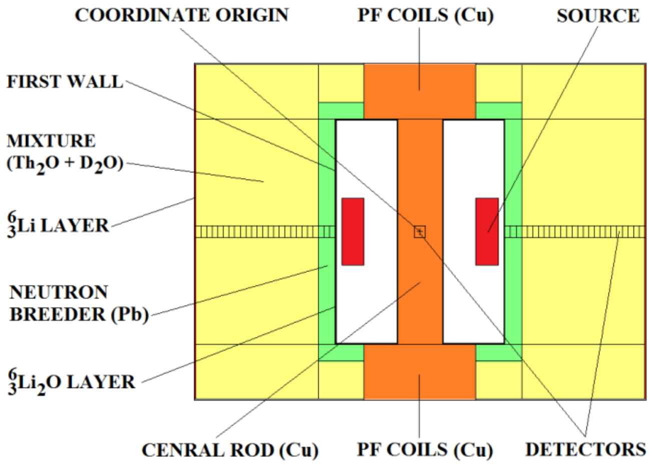


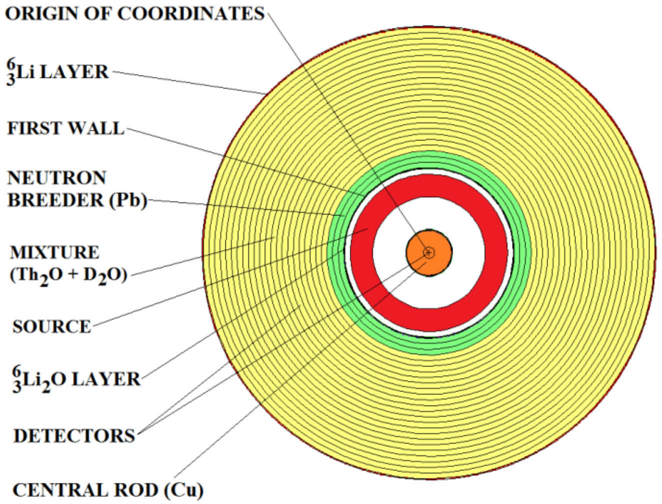


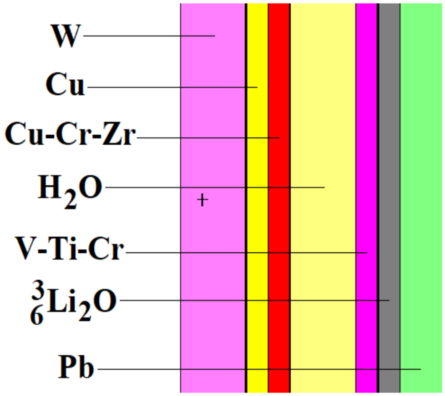
Replacement of D-shaped sectors of the PFC through radial mega-flanges

Concept of a solution thorium-uranium blanket for FNS-C

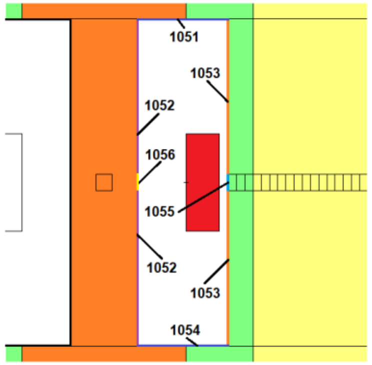
Neutron-physical calculation model







Vertical section of the calculation model in the area of the first wall



Detector placement in the first wall for neutron load calculation

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Concept of a solution thorium-uranium blanket for a FNS-C

The aim of the work: to demonstrate the possibility of producing nuclear fuel

Results of neutron-physical analysis

- Analysis of the results shows that in a blanket with (0% ^{233}U) at a reaction rate of ^{232}Th (n , γ) equal to 9.089×10^{-1} (reactions per neutron), the production of ^{233}U will be 1.968×10^{1} kg/year for a nuclear fusion power of 5 MW.

- In the isotope ^{6}Li with a mass of 331.2 kg, the rate of tritium generation reaction is 3.299×10^{-2} . In lithium oxide $^{6}Li_2O$ with a mass of 23.46 kg (the content of ^{6}Li by mass is 42.86%) – 4.643×10^{-1} 1/s. The total reaction rate (n,t) is 4.973×10^{-1} 1/s. Tritium production is 1.355×10^{-1} kg/year.

- The total rate of reactions (n ,2n) and (n ,3n) in $^{232}ThO2$ is $3.678 \cdot 10^{-3}$ 1/s, the rate of fission reaction (n , f) is $1.116 \cdot 10^{-3}$, reactions (n, γ) – $9.089 \cdot 10^{-1}$. The average yield of neutrons per fission of the ^{232}Th nucleus is 3.081.

- When adding 1.37% ^{233}U to the solution blanket, the production of ^{233}U will be 2.132×10^{2} kg/year. The production of tritium is 4.700×10^{-1} kg/year. The total rate of reactions (n ,2n) and (n ,3n) in $^{232}ThO_2$ is 7.052×10^{-3} , the rate of the fission reaction (n , f) is 1.562×10^{-2} the reaction (n, g) is 9.850. The average neutron yield per fission is 2.489.

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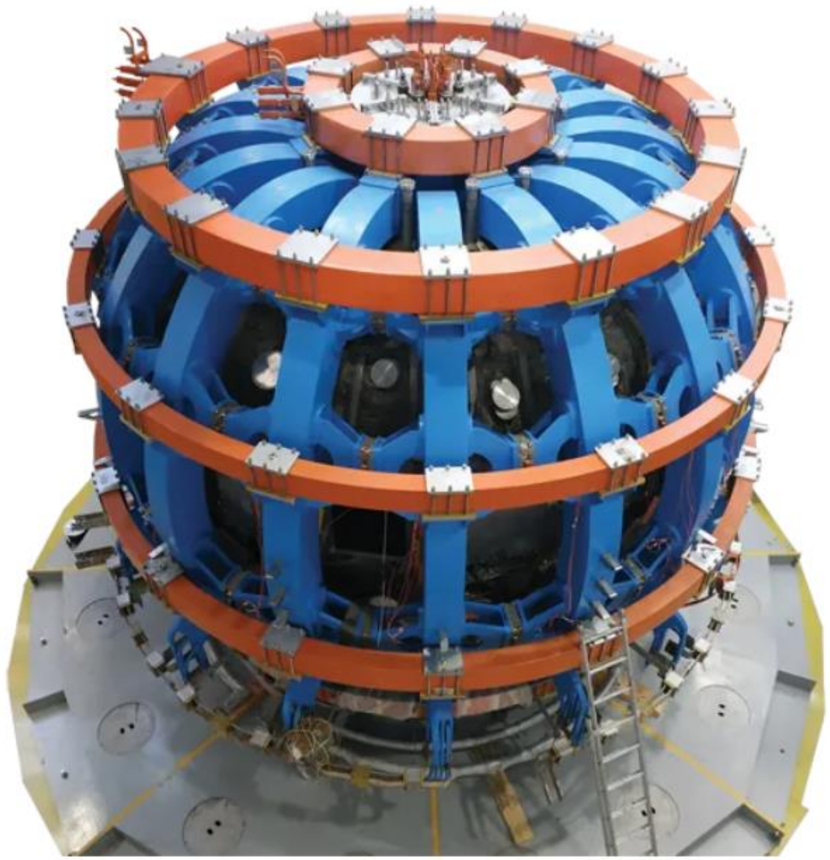
General view of the experimental compact tokamak based on the T-15 MD



In 2024-2025 –
Systematic research on the formation of the appearance of an experimental compact tokamak based on a thermal conductor

T-15 MD - Prototype of FNS-C

Major radius R_0, m . . .	1.48
Minor radius a, m . . .	0.67
Aspect ratio, A. . .	2.2
Toroidal magnetic field B_T, T	2.0
Plasma current Ip, MA . . .	2
Elongation, k95 . . .	1.7—1.9
Triangularity, δ95 . . .	0.3—0.4

The purpose of the experimental compact tokamak is

- demonstration of the possibility of producing nuclear fuel,
- development of technologies,
- study of the behavior of materials under conditions of neutron load of the fusion spectrum

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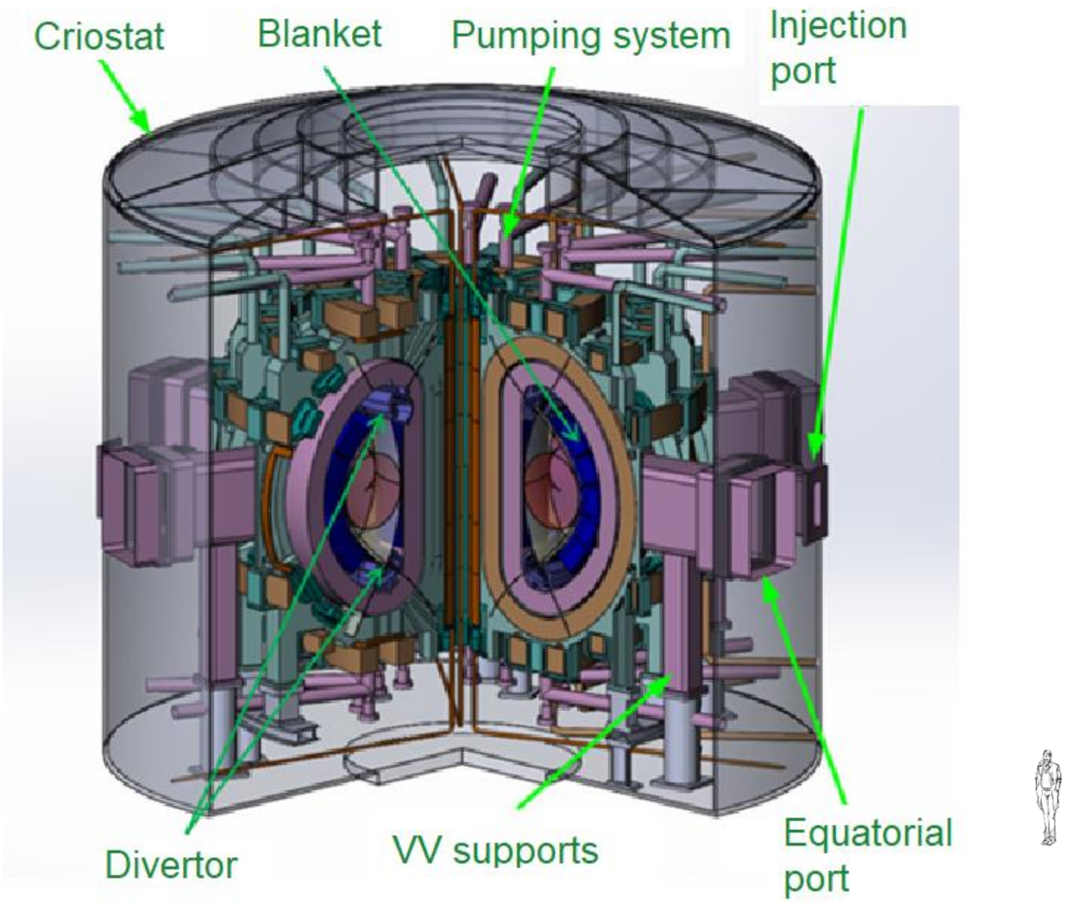
Hybrid facility (2017-2023)



Basic installation parameters:

- Major radius R = 3.2 m
- Minor radius a = 1 m
- Magnetic field B on the axis = 5 T
- Plasma current I_{pl} = 5 MA

•Purpose of the facility:
Conducting comprehensive tests of the designs and technologies of a steady-state tokamak – a neutron source.

•The main objective is to demonstrate the possibility of producing fuel for thermal nuclear reactors - U - 233 from the natural isotope Th - 232

Working parameters:

DT fusion power 40 MW
n- source intensity > 10^{19} n/s
fission power 400 MW
electrical power 200 MW
thermal power 700 MW

With these parameters, the production of U-233 will be up to 1200 kg for 1000 days operation (the operating time depends on the design and composition of the hybrid blanket)

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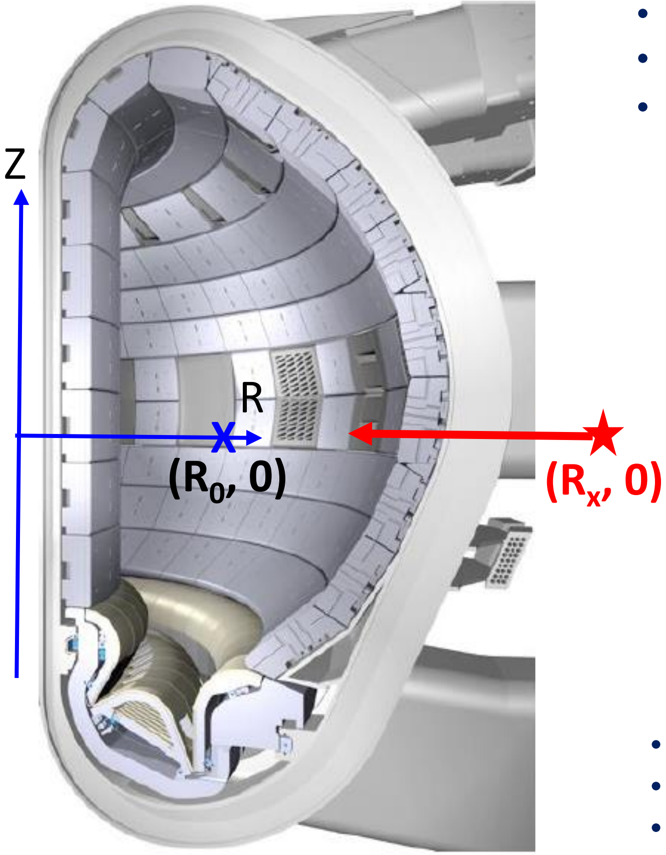
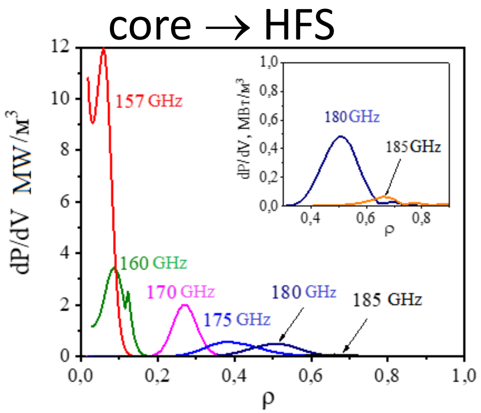
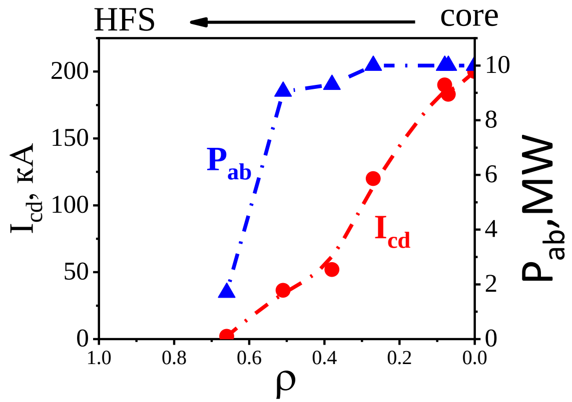
ECR heating and current generation in a hybrid installation



- O1 – ordinary wave at the first harmonic of the ECR
- Parallel beam; Wavefront curvature radius: 10 m
- Equatorial port entry with LFS R_x =4.25 m ; Z=0
- Wave input angles f_T – var ; f_p =0 ^{o}

Absorbed power profiles and current generation





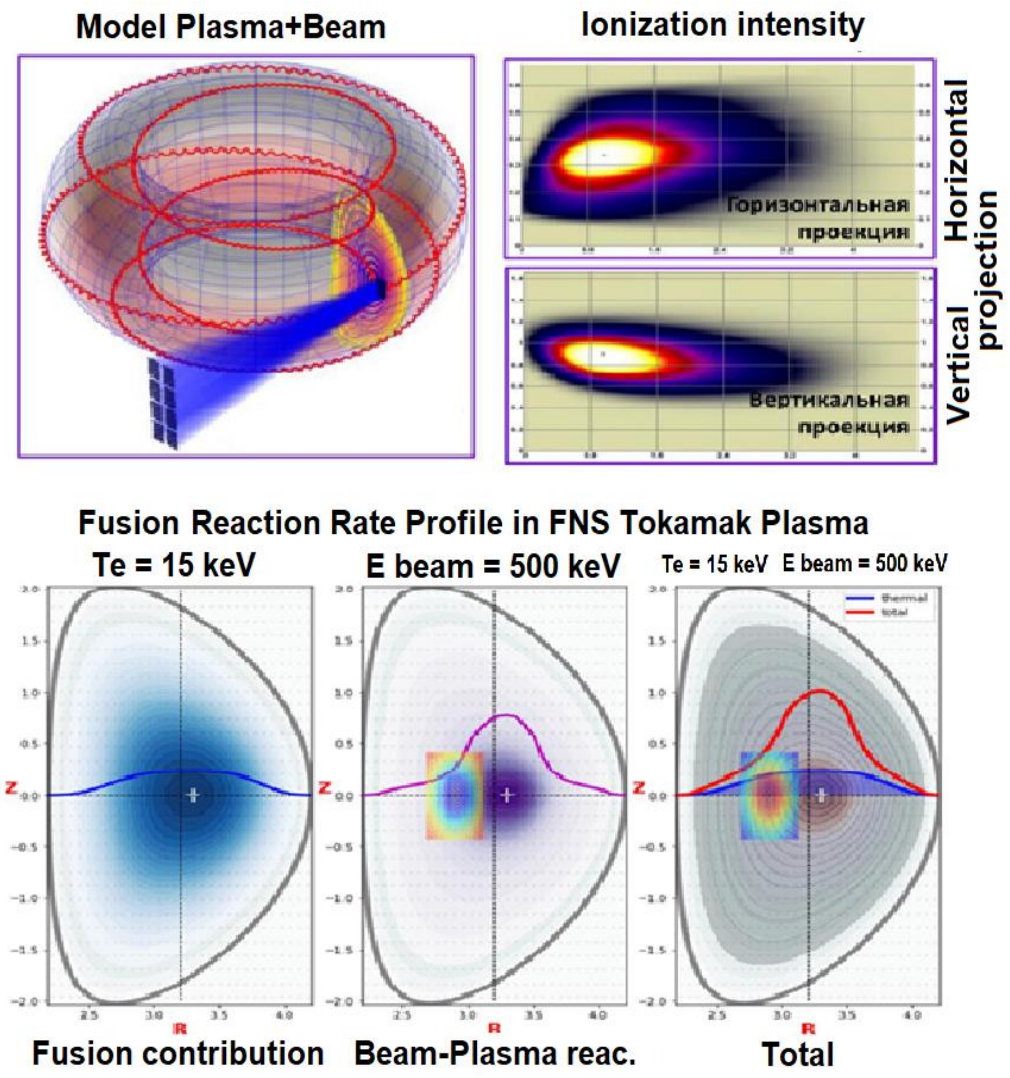
Non-central contribution f_T =20 ^{o}

- Source frequency f= 140 - 180 GHz – existing frequency range
- Possibility of using tunable gyrotrons operating at different frequencies
- The power of the source is not less than 1 MW
- The system operation duration ranges from several hours (in pulse mode) to continuous operation

Analyses results I_{cd} = 200 kA at P_{ab} = 10 MW , T_e (0) = 5 keV = >

I_{cd} = 240 kA at P_{ab} = 6 MW , T_e (0)= 10 keV

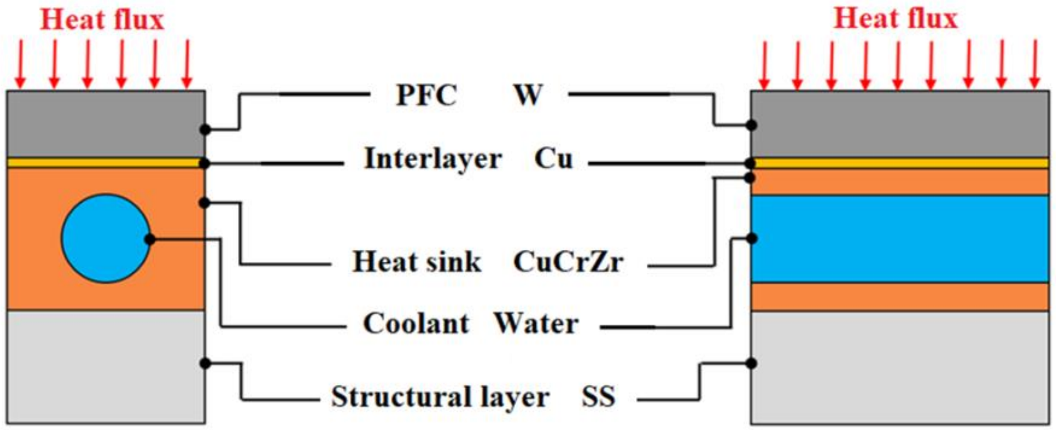
Neutral Beam Heating and Current Drive in the FNS Tokamak

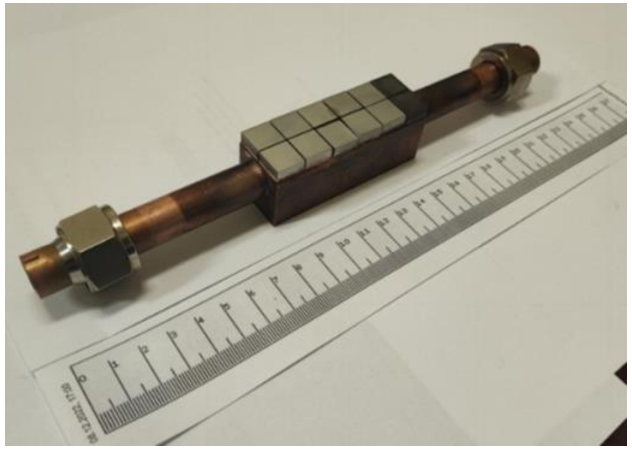


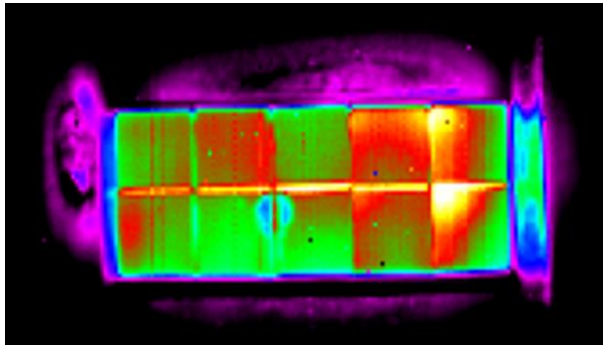
- 2-component operating modes (beam + thermal plasma).
- Intensive generation of neutrons (10^{18} -10^{19} n/s) at moderate costs and installation sizes.
- Beam-controlled steady-state operation and release of thermonuclear fusion power.
- Low plasma temperature relative to pure thermonuclear fusion .
- Low power gain, high neutron yield, plasma rotation and non-inductive current.
- High sensitivity to injection energy, geometry and aiming.
- Flexible control of the distribution of fast ions in phase space by adjusting the NBI parameters.

EXPERIMENTAL STUDY OF the FIRST WALL PROTOTYPES (Efremov Institute)

Prototypes of multilayer plasma-facing elements operating at a thermal load of 2.5 MW/m^2 (for the first wall) and 10 MW/m^2 (for the divertor target), compatible with lithium at temperatures up to 400 °C and neutron







Schematic diagram of multilayer elements (a), a prototype that has undergone thermal testing (b), thermal imaging at a load of 34.72 MW/m^2, T max (W) = 2525°C (c)

A total of 770 loading cycles were carried out with an increase in the absorbed power density from 2 to 35.56 MW/m^2

At a load value of 34 MW/m^2 – no violation of the tightness of the cooling channel (but not PFC).

Berylium vs Tungsten

- A comparative analysis of radiation damage to the tokamak first wall of a hybrid facility was conducted when replacing the beryllium coating with tungsten.
- The results of neutron-physical analyses showed that tungsten is better material for the first wall.
- However, its increased brittleness must be taken into account when exposed to high-energy neutrons for a significant period of time.

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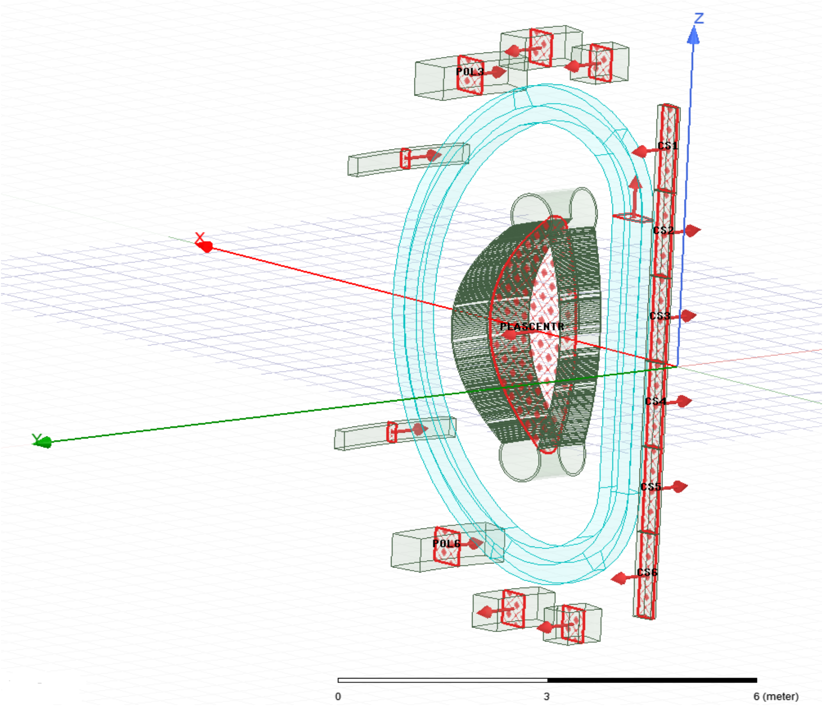
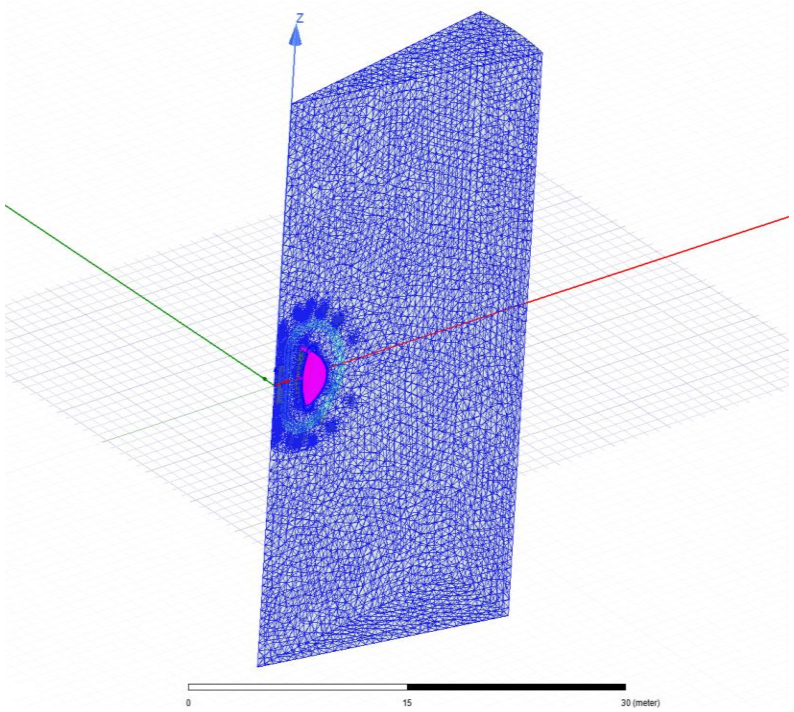
Calculation of electromagnetic forces in the first wall of the large quench

- Analyses were carried out using the finite element method in axisymmetric 3D transient approach of the electromagnetic force impact of thermal quench (TQ) of the plasma current on the first wall of the FNS facility.
- Changes in the magnetic properties of the plasma (5% increase of the toroidal magnetic flux in the plasma) during the initial period of disruption (1 ms) were taken into account, leading to generation of poloidal currents in the wall.
- The subsequent breakdown of the plasma current to zero (up to 40 ms), induse of toroidal currents in the wall, was also taken into account





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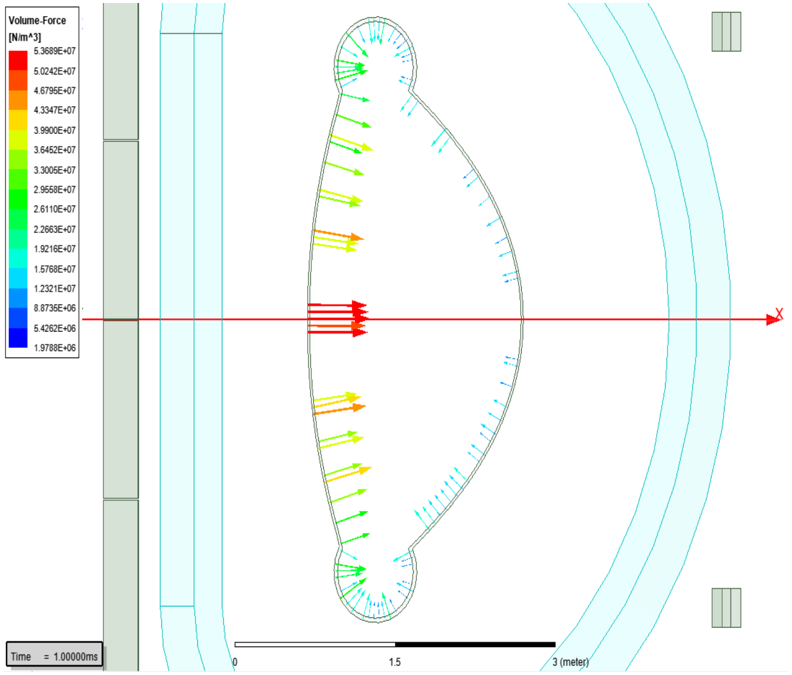
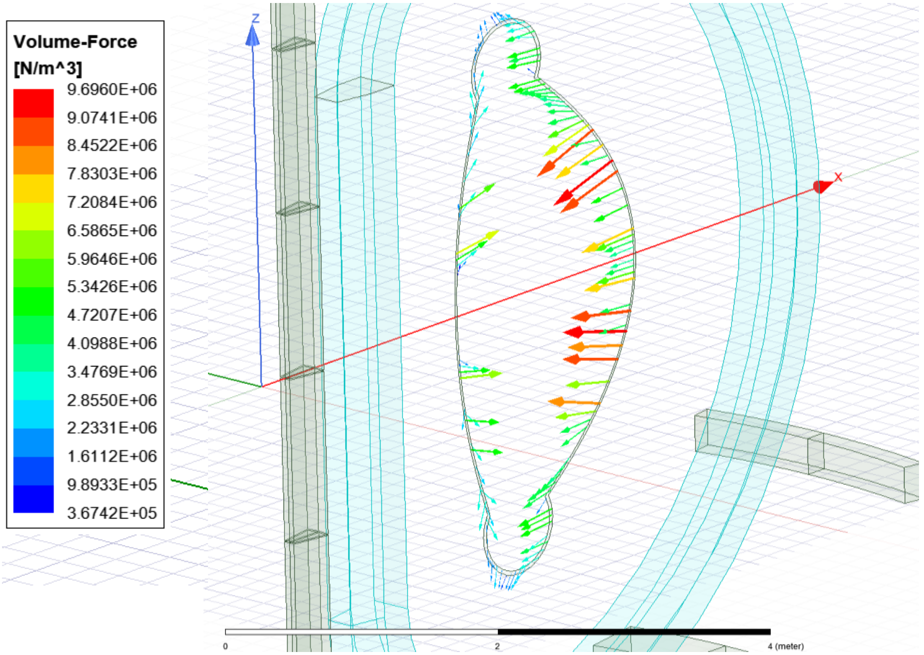
Calculation of electromagnetic forces in the first wall of a hybrid facility





Vectors of volumetric forces (for 1 ms with a sharp change in the magnetic properties of the plasma) (max 54 MN/m^3) in the inner wall locally at the moment and for 40 ms in the poloidal section of the first wall (max 9.7 MN/m^3).

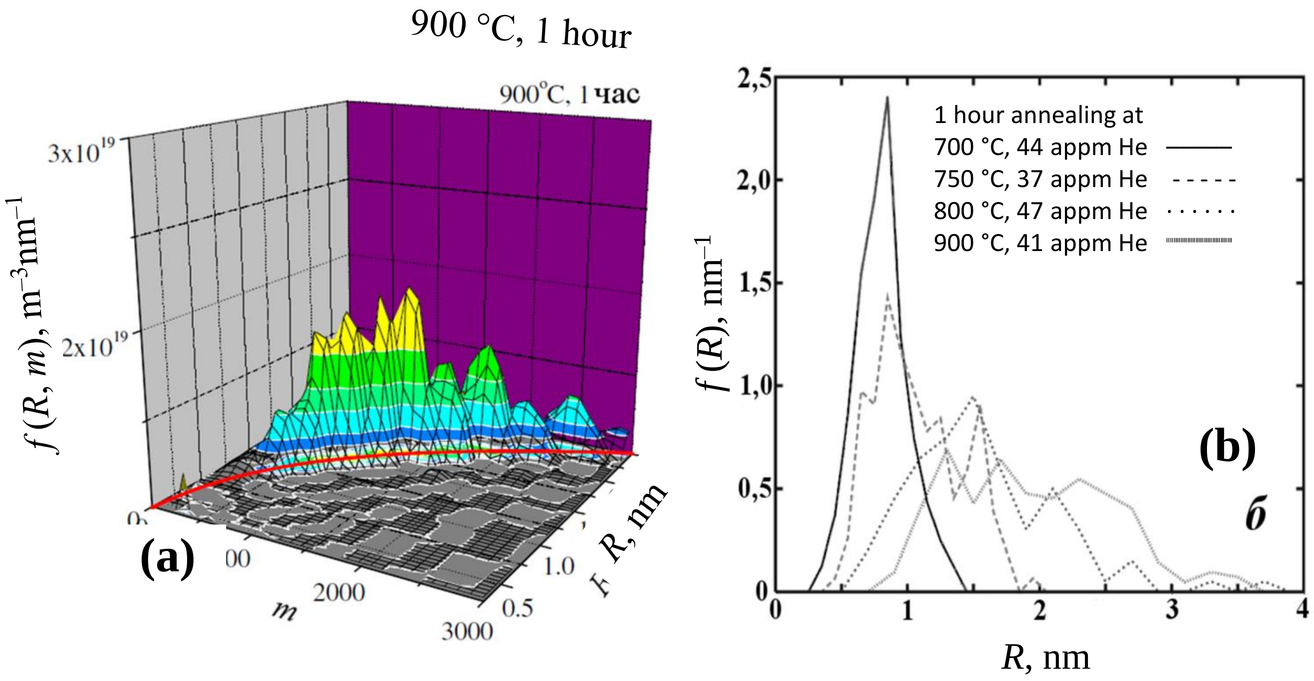
The considered quench, suffisiently different values of the poloidal components of magnetic induction grouth, causing different values of the volume forces in them. Dynamic structural analyses showed that thermomechanical stresses in the first wall slightly exceed the allowable values.

The results can be used as input data for the strength non-steady-state dynamic analysis of the structural elements of the first wall of the FNS.

Materials science work

Formation and growth of He bubbles in metal irradiated with fast neutrons and He ions

The grouping method for cluster dynamics calculations used in this work demonstrated high accuracy and conservation of both the bubble density and the number of point defects stored in these bubbles or voids (swelling). Figure shows an example of the operation of this method as applied to the problem of bubble evolution during annealing after irradiation of austenitic stainless steel. It has been confirmed that the most thermodynamically favorable state for growing bubbles in metals and alloys during annealing is a state close to mechanical equilibrium. Thus, physical validity of theoretical models of He bubbles evolution during annealing, based on this assumption, has been confirmed.



(a) size-distribution function for He bubbles after annealing for one hour at 900 °C and
(b) its temperature dependence

Conclusion

The main goal is to create a pilot industrial FNS facility

The goal in the near future is to build a set of experimental facilities
as well as an experimental compact fusion neutron source

- to study the process of obtaining uranium-233 from natural thorium-232 by irradiating samples containing thorium-232 with a flux of thermonuclear neutrons;
- for testing technologies; steady-state and neutron;
- for testing materials and components of hybrid systems.

THANK YOU FOR

YOUR

ATTENTION