

Fusion-fission hybrid reactor facility: neutronic research^{*}

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Abstract

The authors investigate the neutronic characteristics of the operating mode of a hybrid nuclear-thermonuclear reactor. The facility under study consists of a modified core of a high-temperature gas-cooled thorium reactor and an extended plasma neutron source penetrating the near-axial region of the core. The proposed facility has a generated power that is convenient for the regional level (60–100 MW), acceptable geometric dimensions and a low level of radioactive waste.

The paper demonstrates optimization neutronic studies, the purpose of which is to level the resulting offsets of the radial energy release field, which are formed within the fuel part of the blanket during long-term operation and due to the pulsed operation of the plasma D-T neutron source.

The calculations were performed using both previously developed models and the SERPENT 2.1.31 precision program code based on the Monte Carlo method. In the simulation, we used pointwise evaluated nuclear data converted from the ENDF-B/VII.1 library, as well as additional data for neutron scattering in graphite from ENDF-B/VII.0, based on the $S(\alpha, \beta)$ formalism.

Keywords

Fusion-fission hybrid reactor facility, plasma D-T-neutron generator, neutronic research

Introduction

Thermonuclear research is being conducted on an interstate scale and is aimed at the prospect of entering industrial energy production after 2050. On this path, in 2020, researchers at the Korea Institute of Fusion Energy (KFE) managed to achieve plasma confinement in a toroidal magnetic trap (KSTAR tokamak (Nield 2020)) for 20

seconds at a temperature of 100 million degrees (data for December 2020), which today is an absolute record for the implemented set of parameters. The result achieved by the scientists suggests that the international program for the construction of the ITER tokamak will be completed with the production of such an amount of energy from plasma that exceeds the energy costs for obtaining and confining plasma.

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Power production plants using tokamaks will be of exceptionally large size and capacity; they will be built in the distant future. Our research is focused on the prospect for the practical use of thermonuclear power in a shorter period (Arzhannikov et al. 2019a, 2019b, 2020, Prikhodko and Arzhannikov 2020, Yang et al. 2021). The purpose of our research is to create a subcritical fusion-fission facility, the concept of which is proposed and developed by the Budker Institute of Nuclear Physics of the Siberian Branch of the Russian Academy of Sciences (Novosibirsk), Tomsk Polytechnic University (Tomsk) and Russian Federal Nuclear Center – Zababakhin All-Russia Research Institute of Technical Physics *a.k.a.* RFNC–VNIITF (Snezhinsk).

The reactor facility under study is a hybrid reactor, the core (blanket) of which consists of an assembly of unified fuel elements of a high-temperature gas-cooled thorium reactor (HGTRU, Tomsk Polytechnic University, Tomsk (Shamanin et al. 2015, Shamanin et al. 2018, Bedenko et al. 2019a, 2019b) and an extended magnetic gas-dynamic trap (GDT) penetrating the near-axial region of the core (the Budker Institute of Nuclear Physics of the Siberian Branch of the Russian Academy of Sciences (Novosibirsk) (Beklemishev et al. 2013, Anikeev et al. 2015, Gas-Dynamic Multiple-Mirror Trap 2021). This hybrid nuclear-thermonuclear reactor has a generated power that is convenient for the regional level (60–100 MW), acceptable geometric dimensions, a small amount of spent fuel and a low level of radioactive waste in comparison with common LWRs.

The paper describes optimization neutronic studies, the purpose of which is to level the resulting offsets of the radial energy release field, which are formed within the fuel part of the blanket during long-term operation and due to the pulsed operation of the plasma D-T neutron source. The calculations were performed using the SERPENT 2.1.31 precision program code based on the Monte Carlo method (Leppaanen et al. 2015, NEA 2021).

Materials and methods

Neutronic model of the facility

The reactor facility under study (Fig. 1) consists of a plasma D-T neutron source (PNS) (Beklemishev et al. 2013, Anikeev et al. 2015, Gas-Dynamic Multiple-Mirror Trap 2021) and a blanket power generating part, which is based on the concept of the core of a multi-purpose high-temperature low-power gas-cooled thorium reactor unit (Shamanin et al. 2015) with a modified near-axial region under the PNS. This modified core, in fact, is a blanket composed of regular hexagonal graphite fuel elements containing pellets filled with microencapsulated thorium and plutonium oxides in cylindrical channels (Fig. 2). If the blanket is arranged in this way, then its dimensions, and hence its performance, can be varied by changing the number of fuel and non-fuel elements installed in it.

The PNS (a scheme of the computational model is in Fig. 2d) of the facility under study, located in the near-axial region, is a cylindrical vacuum chamber in which the magnetic field holds the high-temperature plasma. The cylindrical chamber designed to generate D-T neutrons corresponds in diameter and length to the dimensions of the near-axial region of the blanket with nuclear fuel.

The plasma pinch zone (Fig. 2a, d) or the neutron production zone is a volume isotropic monoenergetic neutron source with intensity $I_{v_6} = 1.76 \times 10^{12} \text{ n} \times \text{s}^{-1} \times \text{cm}^{-3}$ ($5.76 \times 10^{17} \text{ n} \times \text{s}^{-1}$ from the entire plasma volume) (Prikhodko and Arzhannikov 2020) and a longitudinal profile (Fig. 3a) for the optimal solution v6 from (Prikhodko and Arzhannikov 2020). This optimal longitudinal profile of the neutron yield density was used in calculations to determine the power distribution of the flux density of neutrons (Fig. 3b) coming from the PNS into the blanket part of the facility. The spatial (Fig. 3a) and power (Fig. 3b) distributions of neutrons are input data sets for neutron-physical calculations using the 3D model (Fig. 2a).

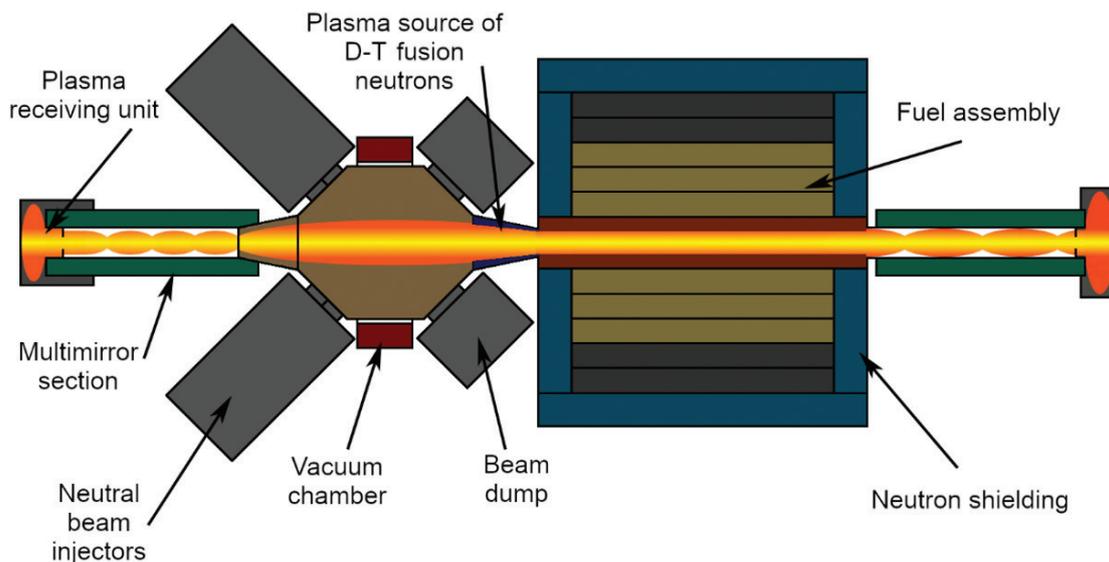


Figure 1. Conceptual design of the fusion-fission hybrid reactor facility (Arzhannikov et al. 2020).

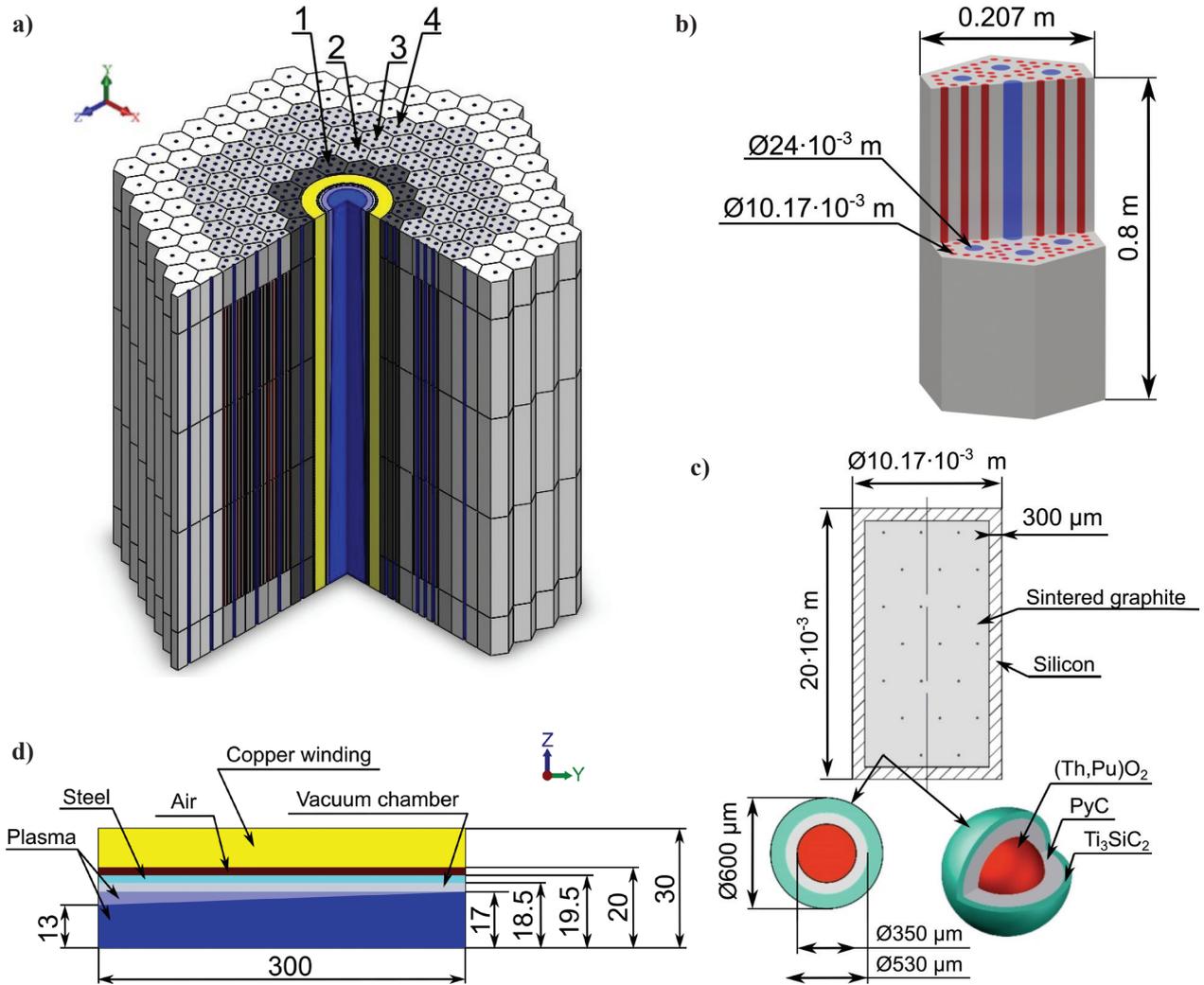


Figure 2. Scheme of the design solution for the fusion-fission hybrid reactor facility **a)** 3D design model of the facility, including the blanket with Th-Pu fuel and the extended PNS (1–4 are the numbers of rows with fuel and non-fuel elements); **b)** fuel element of a unified design (Shamanin et al. 2015); **c)** fuel pellet (Bedenko et al. 2019b, Shamanin et al. 2015); **d)** computational model of the magnetic trap (dimensions are given in centimeters).

It should be noted that in the considered ‘plasma neutron source – subcritical blanket’ configuration, the high-temperature plasma pinch is formed in a repetitively pulsed mode and propagates from the near-axial region over the entire multiplying part in time correlation with the PNS.

The simulation results (Fig. 4) showed that the effect of the repetitively pulsed operating mode of the PNS is most noticeable in the first rows of the fuel elements adjacent to the PNS (Fig. 4a, layer 5 is included in the computational region of the second row of the fuel elements).

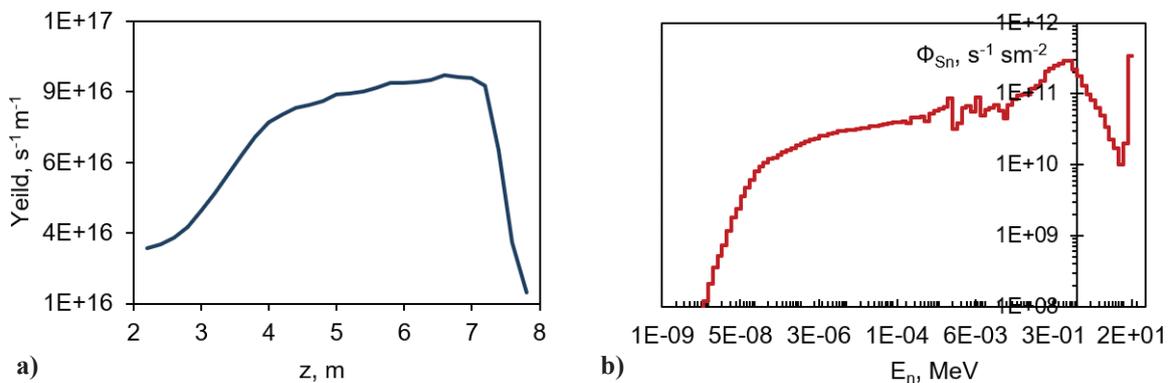


Figure 3. Spatial and power distributions of the neutron flux density: **a)** longitudinal neutron yield profile $I_N(z)$ per running meter of the plasma column (Prikhodko and Arzhannikov 2020); **b)** power distribution of the neutron flux density.

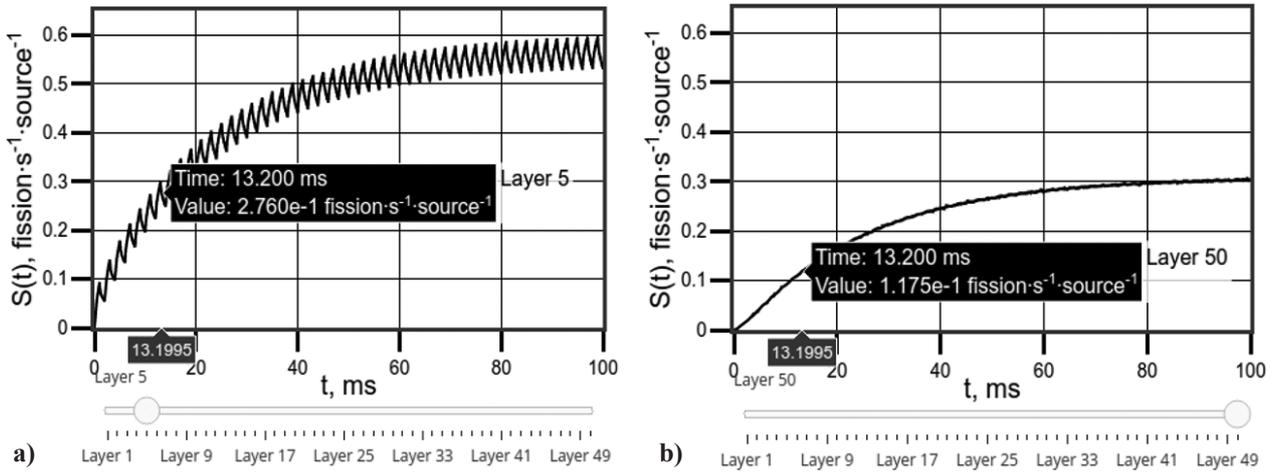


Figure 4. Fission rate in the blanket powered by the pulse PNS. The numbers of layers indicated on the scale are the calculated areas of equal volume into which the blanket part of the facility is divided, in order to display more accurate data on the fission rate in the fuel elements 2–4.

At the periphery of the fuel part (Fig. 4b, layer 50 is included in the computational region of the fourth peripheral row of the fuel elements), the effect of neutron emission nonstationarity manifests itself very weakly and, at a certain pulse repetition rate, such nonstationarity can be neglected in all the fuel elements. This result made it possible to replace the pulsed neutron source with a quasi-stationary one in further optimization neutronic studies.

Note that reaching the steady state in the entire multiplying region is observed in the time interval from 100 ms to 1 s. In a second, the total number of fissions in the blanket increases to 20 (per one neutron coming from the PNS into the blanket in the radial direction). This value does not change any further, providing heating of the blanket at a rate of no more than $10 \text{ K} \times \text{h}^{-1}$ with a constant neutron emission from the PNS at a level of $5.76 \times 10^{17} \text{ n} \times \text{s}^{-1}$, which meets the requirements of thermal engineering reliability during a cold startup.

Computational methods

Neutronic calculations were performed using the SERPENT 2.1.31 precision program code based on the Monte Carlo method (Leppanen et al. 2015). In the simulation, we used pointwise evaluated nuclear data converted from the ENDF-B/VII.1 library (NEA 2021), as well as additional data for neutron scattering in graphite from ENDF-B/VII.0, based on the $S(\alpha, \beta)$ formalism (SERPENT 1.1.0 2021). In each calculation, 1×10^9 histories were played, which made it possible to ensure the accuracy of the desired solution equal to 0.01%, and also to take into account the nonlinearity in the intensity of neutron multiplication in the blanket part of the facility. Note that the number of neutrons played in the system does not correspond to the real number of neutrons emitted by the PNS; for this reason, the calculation result is normalized to one neutron emitted by the source per unit time (fission $\times \text{s}^{-1} \times \text{source}^{-1}$).

Results and discussion

The neutronic optimization of the facility was performed by profiling the power density along the radius of the fuel part of the blanket by changing the content of the Pu fraction. In order to reduce the power density in the near-axial region, the first row of the fuel elements, even before profiling, was immediately replaced by graphite elements with holes for the helium coolant (Fig. 2a). The fuel elements of rows 2–4 (Fig. 2a) are loaded with fuel pellets with a volume fraction of the dispersed phase $\omega = 17\%$ (Fig. 5c). The calculated peak power density values for the fuel elements before and after profiling are shown in Fig. 5a, b, respectively (as a percentage of the total power).

As expected, the most power-intensive part is in the first row adjacent to the PNS. The calculation showed that

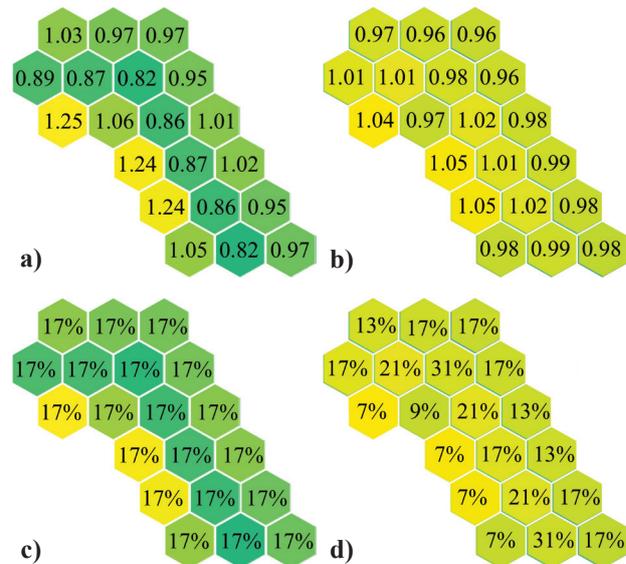


Figure 5. Loading and power density scheme (1/3 of the blanket), in percent, before and after profiling the fuel part of the facility.

the maximum peak power density of an unprofiled blanket reaches a level of 1.25% (Fig. 5a).

After profiling (Fig. 5b), the radial power density profile became more uniform, while the non-uniformity coefficient (the ratio of the maximum power density in the element to the total power density in the entire blanket) was reduced to 1.05; the loading scheme obtained in this way is shown in Fig. 5d.

The subcriticality value required for such systems is achieved through the use of burnable poisons (BP). Table 1 summarizes the five most successful reactivity compensation options. These results demonstrate the solution of a conditionally critical problem, i.e., the solution was obtained for $P_{th} = \text{const}$ and in the absence of additional neutron generation in the plasma source.

Table 1. Calculation results for various reactivity compensation options of the facility blanket

Calculation option	H. met. mass, kg	Pu mass, kg	BP mass, kg	Exposure time (250 MW×day/kg), g
01_non-profiled blanket (Core)	290.77	147.57	–	3.32
02_profiled blanket (Core)	302.33	153.44	–	3.45
03_ZrB ₂	305.05	154.82	5.23	3.48
04_Gd ₂ O ₃ Hom	277.12	157.76	22.86	3.16
07_Er ₂ O ₃ Hom	238.34	160.48	56.18	2.72
09_HfO ₂ Hom	235.96	157.85	57.73	2.69
11_Pa-231Hom	233.78	157.41	76.04	3.53

The burnable poison was used in two placement options (Table 1): homogeneously (in the composition of fuel microcapsules, see the calculation options: 04_Gd₂O₃Hom, 05_Er₂O₃Hom, 06_HfO₂Hom and 7_Pa-231Hom) and heterogeneously, as a micron layer on the fuel pellet surface (see the calculated option: 03_ZrB₂).

The options 01 and 02 are non-profiled and profiled fuel parts of the facility blanket, respectively. The option 03_ZrB₂ is a heterogeneous way of placing the burnable absorber, representing a technological solution proposed in (Linnik et al. 2011). For the options 04_Gd₂O₃Hom, 07_Er₂O₃Hom, 09_HfO₂Hom, and 11_Pa-231Hom, a homogeneous placement of the burnable poison was used. In these calculations, the BP was placed in microcapsules of the fuel pellets by reducing the content of Th in them, thus the mass of the fissile material remained the same (Table 1).

An analysis of the results given in Tab. showed that the best reactivity compensation options, in terms of neutronic characteristics, are 03_ZrB₂ and 07_Er₂O₃Hom (Table 1 and Fig. 6).

In further calculations, the 03_ZrB₂ option was used, since the technology for applying such coatings was developed at the Tomsk Polytechnic University (Linnik et al. 2011); moreover, when ZrB₂ is used, the boron run-out and the released reactivity are significantly lower in comparison with the same technological solution in the basic HTGR configuration (Shamanin et al. 2015, 2018). In addition, the mass of the BP is minimal from all the options presented in Table 1, and the fuel campaign duration

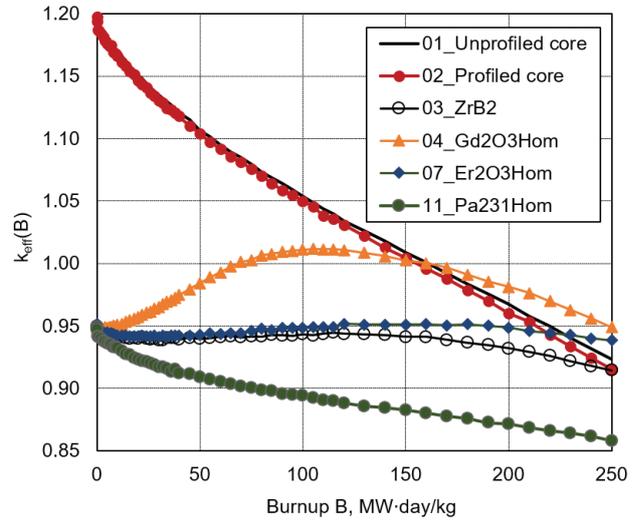


Figure 6. Fuel burnup dependence of the blanket multiplication properties.

is 3480 effective days with a burnup of 250 MW×day per kilogram of heavy metals.

Note that the system of rods intended for control and emergency protection is not calculated in the configuration under study, since the facility is in a subcritical state throughout the entire operating cycle (Fig. 6, calculation options: 03_ZrB₂, 04_Gd₂O₃Hom and 11_Pa-231Hom), and all the control of its work is carried out by the neutron flux from the PNS.

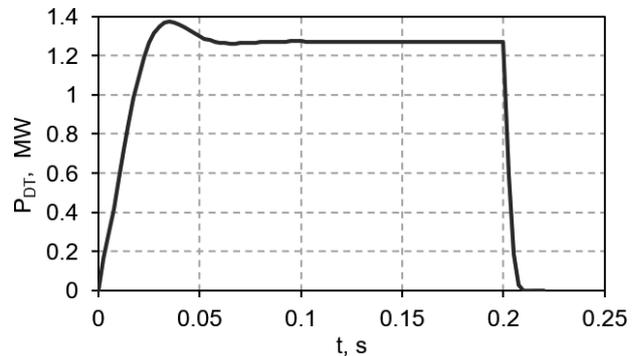


Figure 7. Time evolution of the power released in D-T neutrons. The injection turns on at the moment $t = 0$ s and turns off after 0.2 s.

It should also be noted that the use of PNS as an additional neutron source increases the nuclear safety of the facility, since when the injection of neutral atoms is turned off, the neutron generation drops by about a factor of two over the first 2.5 ms and another 20 times over the next 5 ms (Fig. 7).

This result indicates that the decrease in the generation of additional neutrons in the blanket proceeds much faster than it occurs in the core of a conventional reactor.

Conclusion

The paper describes the neutronic and thermophysical optimization of the operating mode of the facility:

- radial power density profiling in the blanket fuel part was performed by locally changing the content of the Pu fraction in the blanket volume, the loading scheme obtained in this case is illustrated in Fig. 5d; and
- the materials of burnable poisons were selected to compensate for the excess reactivity in the blanket part and to organize the possibility of controlling it in conjunction with the PNS (Table 1 and Fig. 6). Placing the burnable poison (ZrB_2 , Gd_2O_3 , B_4C , Er_2O_3 , HfO_2 , ^{231}Pa) was considered in two options: (1) homogeneously, when it was a part of fuel microcapsules, and (2) heterogeneously, when it was used as a micron layer on the fuel pellet surface.

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