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Neutronics Analysis of Blankets for Hybrid Fusion Neutron Source

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Abstract
Modern challenges of nuclear power are concerned a potential shortage of nuclear fuel for fission reactors in the case of intensive development of nuclear industry, insufficient development of economic and safe fast reactors, underutilization of the closed fuel cycle, spent nuclear fuel reprocessing, nuclear power safety, nonproliferation of weapon-grade materials. Expected progress in solving these problems is associated with development of a hybrid fusion neutron source (FNS) with a subcritical blanket, using the closed nuclear fuel cycle. In this work it is investigated capabilities of hybrid fusion neutron source blankets in generation of ^{233}U, ^{239}Pu, and ^{3}H nuclides, reprocessing of spent nuclear fuel, electricity and heat production. The basic kinds of the blankets are considered. There are a uranium and thorium solid-state blanket, a blanket with the use of heavy water solutions of salts, and oxides of uranium and thorium, a molten lead and a molten salt blanket. The structure and geometrical parameters of the blankets, the moderators, and the fertile materials are optimized to obtain the ultimate nuclear fuel yield at the minimal accumulation of radiation toxic wastes. The neutronics comparative analysis of the different blanket models is presented. In the result, the optimal FNS blanket models have chosen.

Keywords: nuclear fuel generation, hybrid fusion neutron source, solid-state blanket, aqueous blanket, molten salt blanket, uranium-238, plutonium-239, thorium-242, uranium-233, tritium.

1. Introduction.

Nuclear fuel production, and reprocessing of spent nuclear fuel are related to the most actual modern problems. A fusion neuron source (FNS) has a special role in their solution.

Modern thermal reactors do not provide reproduction of burned fuel nuclides. Its conversion ratio is approximately equal to 0.6. The doubling time of fast reactors is 10-20 years. A number of fast reactor projects (for example, BREST) does not provide additional fuel production, and demands significant fuel nuclide stocks for an initial loading and maintenance of the closed fuel cycle. Breeder reactors are capable to produce effectively neutrons and fuel nuclides. However, for achievement of their criticality, an initial loading of fuel nuclides is required that demands either withdrawal of these nuclides from the uranium-plutonium fuel cycle, or their generation with the use of external neutrons. Electronuclear systems for power applications, and micro explosions for laser ignition of thermonuclear targets are technologically undeveloped.

For the nuclear power industry development, it is required an effective nuclear fuel manufacturer, which is safe from the viewpoint of severe accidents, negative impact on the environment, and proliferation of weapons-grade materials. Another key problem of nuclear power is utilization of spent nuclear fuel. A possible solution of these problems is development of a hybrid fusion neutron source with a subcritical blanket, using the closed nuclear fuel cycle.

Therefore, design of a hybrid FNS "fusion-fission" system for manufacture of artificial nuclear fuel, and nuclide transmutation is actual [1-11]. Also it is desirable to create FNS capable to generate nuclear energy for electricity and heat production. To solve this problem, it is necessary to provide k_{eff} which is equal to 0.95.

2. Objectives

At a stage of the outline designing, a hybrid source of thermal neutrons with neutron multipliers on the basis of Be and Pb was considered as one of the basic variants of FNS realization. Additional multiplication of neutrons was performed by fission reactions induced by 14 MeV neutrons interacting with nuclei of ^{232}Th or ^{238}U, and by fission reactions induced by thermal neutrons interacting with nuclei of ^{233}U or ^{239}Pu.

To estimate FNS opportunities in production of fissile isotopes ^{233}U and ^{239}Pu, also in ^{239}Pu reprocessing, the FNS neutronics analysis was performed for blankets with the use of heavy water solutions of salts, and oxides of uranium and thorium, solid-state and molten salt blankets.

The purpose of the calculations was optimization of a composition, quantity, and geometrical parameters of FNS materials generating and absorbing neutrons for effective production of ^{233}U or ^{239}Pu isotopes. Also simultaneous tritium production was assumed. Capabilities of spent fuel reprocessing, and criticality analysis to produce electricity were also taken into account. FNS nuclear synthesis power was from 1 to 10 MW.

In the result of the optimization, it is desirable to increase the yield of fissile isotopes up to 1 and more nuclide per 1 D-T neutron.

The MCNP-4 code [12] with the point cross-section libraries generated from the FENDL-2 [13] and ENDF/B-6 [14] files were used for the calculations.

3. Results of neutronics analysis

^{239}Pu is produced from ^{238}U in the result of the reactions (Figure 1)

$^{238}U(n,\gamma)$ $^{239}U(\beta\text{-decay}, T_{1/2}=23.45 \text{ min.})$ $^{239}Np(\beta\text{-decay}, T_{1/2}=2.356 \text{ days})$ ^{239}Pu.

The reaction chain for the production of ^{233}U from ^{232}Th is

$^{232}Th(n,\gamma)$ $^{233}Th(\beta\text{-decay}, T_{1/2}=23.5 \text{ min.})$ $^{233}Pa(\beta\text{-decay}, T_{1/2}=26.967 \text{ days})$ ^{233}U.

The reaction for the production of ^{3}H from ^{6}Li и $^{6}Li_2O$ is

$^{6}Li\ (n,^{4}He)^{3}H$.

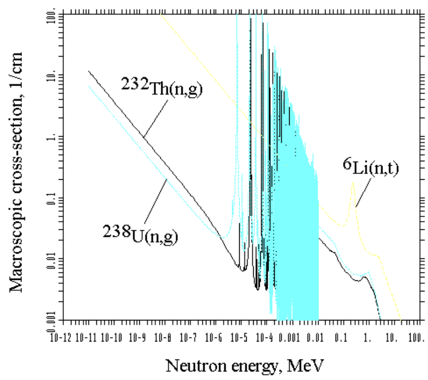


Figure 1. $^{232}Th\ (n,\gamma)$, $^{238}U\ (n,\gamma)$ and $^{6}Li\ (n,t)$ cross-sections.

3.1. Solid-state blankets

The FNS calculation model with a thorium-containing blanket is shown in Figure 2 and 3. The parameters of the model components are presented in Table 1.

Table 1. The components and parameters of the FNS model.
R is a major radius, r is a minor radius, h is a height

Component	Parameters
FNS itself	Overall sizes: R = 200.8 cm, h = 301.6 cm
Source	14.1 MeV D-T neutrons, isotropic; R = 70 cm, r = 50 cm, h = 60 cm
Central rod	Be, R = 20 cm, h = 201.6 cm
Poloidal fieid coils (PFC)	Be, R = 50 cm, h = 50 cm
Vacuum chamber	R = 75 cm, r = 20.8 cm, h = 200 cm
First wall	3 mm W, 3 mm H_2O, 2 mm V92-Ti4-Cr4
Blanket	Be, R = 90.9 cm, r = 75.9 cm, h = 215.9 cm
Moderator	D_2O
Pipes	Cylindrical shell: R = 2 cm, r = 1.7 cm, h = 231.6 cm.
$^6Li_2O$ layer	R – r = 1 mm
^6Li layer	R – r = 1 cm
Detectors	25 detectors: R – r = 5 cm, h = 10 cm for every one. Central detector: R = 5 cm, h = 10 cm

The model includes two layers containing ^6Li for tritium generation. Behind the first wall there is the $^6Li_2O$ layer. Another one is consisted of pure ^6Li. This layer is an external cover of the model.

There are one or two pipe rows in the model. Each row consists of 120 pipes, which distributed with regular intervals on a circle. The pipes are composed of metal ^{232}Th or material containing ^{232}Th. They can be placed inside the blanket or the moderator. The distances from the pipe rows to the first wall are changeable. The pipes can be filled by different materials. The total weight of 120 pipes from pure ^{232}Th is 1136.2 kg.

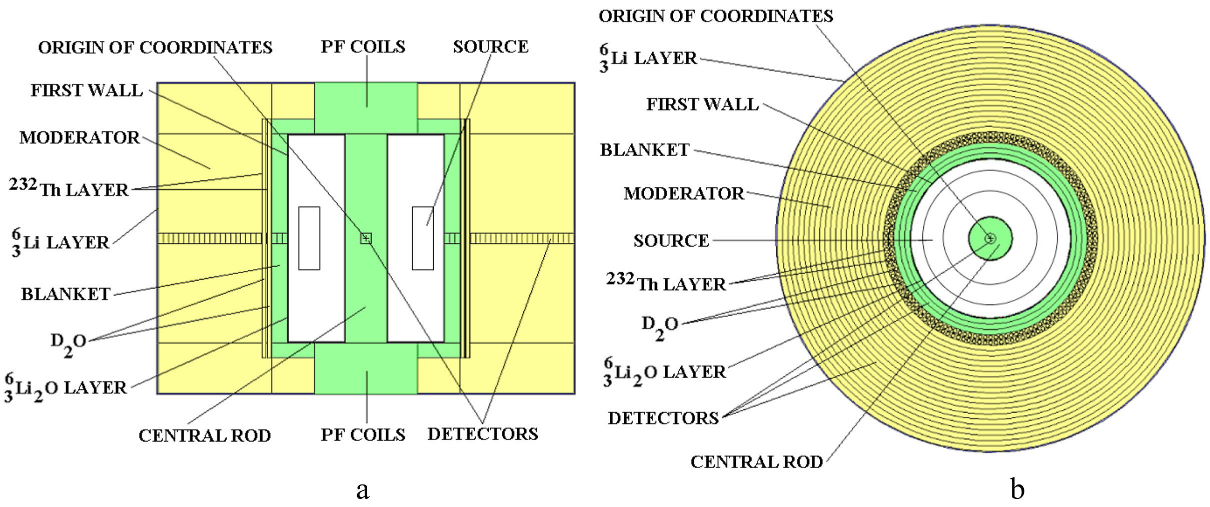
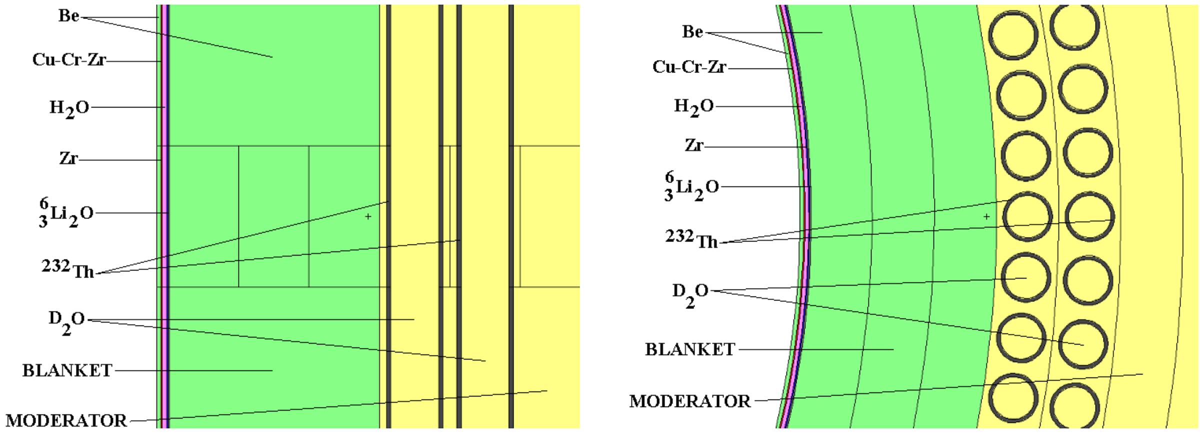


Figure 2. Vertical (a) and horizontal (b) section view of the FNS model.



a b

Figure 3. Vertical (a) and horizontal (b) section view of the FNS ^{232}Th pipes in a D_2O moderator.

The calculation results of this model with different blankets and moderators are presented in Tables 2, 3, 4 and 5.

Table 2. Reaction rates and nuclear fuel production in the FNS model with ^{232}Th pipe rows. The mass of 120 thorium pipes of each row is 1136.2 kg. h is a Be blanket thickness. After "Row N" a rod surrounding material and a distance from the rod row to the first wall are presented. Fusion power is 1n/s for the reaction rates and $1.775\cdot10^{18}$ n/s (5 MW) for the fuel production.

Variant	1	2	3	4	5
h Be, cm	15	15	15	10	15
Row 1	Be 5.5 cm	Be 10.5 cm	Be 10.5 cm	D_2O 10.5 cm	D_2O 15.5 cm
Row 2	Absent	Absent	D_2O 15.5 cm	D_2O 15.5 cm	D_2O 20.5 cm
(n, γ)—^{232}Th	$4.610\cdot10^{-1}$	$4.729\cdot10^{-1}$	$6.358\cdot10^{-1}$	$5.835\cdot10^{-1}$	$6.110\cdot10^{-1}$
(n, t)—$^6Li_2O$	$7.073\cdot10^{-1}$	$7.970\cdot10^{-1}$	$7.700\cdot10^{-1}$	$6.690\cdot10^{-1}$	$8.841\cdot10^{-1}$
(n, t)—^{6}Li	$6.659\cdot10^{-1}$	$5.816\cdot10^{-1}$	$4.759\cdot10^{-1}$	$5.075\cdot10^{-1}$	$4.326\cdot10^{-1}$
(n, t)— total	1.373	1.379	1.246	1.176	1.317
(n, f) + (n, xn)	$3.8\cdot10^{-2}$	$2.0\cdot10^{-2}$	$3.3\cdot10^{-2}$	$3.6\cdot10^{-2}$	$2.0\cdot10^{-2}$
^{233}U, kg/year	9.994	10.25	13.78	12.65	13.25
^{3}H, kg/year	$3.852\cdot10^{-1}$	$3.867\cdot10^{-1}$	$3.495\cdot10^{-1}$	$3.300\cdot10^{-1}$	$3.693\cdot10^{-1}$

Table 3. Reaction rates and nuclear fuel production in the FNS model with two rows of pipes consisted of different materials. The pipes are placed in different moderators. The distance from the first wall to the 1-st pipe row is 15.5 cm and to the 2-nd is 20.5 cm. The blanket thickness is 15 cm. Fusion power is 1n/s for the reaction rates and $1.775\cdot10^{18}$ n/s (5 MW) for the fuel production.

Variant	1	2	3	4	5	6
Pipe material, vol. %	100% ^{232}Th	ThF_{4+} FLiNaK[1])	96% ^{232}Th 4% ^{233}U	96% ^{232}Th 4% ^{233}U	96% ^{232}Th 4% ^{233}U	100% ^{232}Th
Material in pipes, vol. %	D_2O	ThF_{4+} FLiNaK[1])	D_2O	^{nat}Pb	96% ^{232}Th 4% ^{233}U	CO_2[2])

^{232}Th total mass, kg	2272.4	1857.4	2181.5	2181.5	7861.5	2272.4
^{233}U total mass, kg	0.0000	0.0000	145.75	145.75	525.26	0.0000
Moderator material	D_2O	D_2O	D_2O	^{nat}Pb	^{nat}Pb	CO_2
Blanket material	Be	Be	Be	^{nat}Pb	^{nat}Pb	Be
(n, γ)—^{232}Th	$6.110 \cdot 10^{-1}$	$8.792 \cdot 10^{-2}$	1.621	$3.260 \cdot 10^{-1}$	$6.953 \cdot 10^{-1}$	$3.749 \cdot 10^{-1}$
(n, t)—$^{6}Li_2O$	$8.841 \cdot 10^{-1}$	$8.550 \cdot 10^{-1}$	1.905	$6.510 \cdot 10^{-1}$	$6.130 \cdot 10^{-1}$	$8.712 \cdot 10^{-1}$
(n, t)—^{6}Li	$4.326 \cdot 10^{-1}$	$3.750 \cdot 10^{-1}$	2.718	$1.873 \cdot 10^{-1}$	$1.659 \cdot 10^{-1}$	$3.827 \cdot 10^{-1}$
(n, t)—total	1.317	1.230	4.623	$8.383 \cdot 10^{-1}$	$7.789 \cdot 10^{-1}$	1.254
(n, f) + (n, xn)	$2.0 \cdot 10^{-2}$	$6.0 \cdot 10^{-3}$	3.339	$2.227 \cdot 10^{-1}$	$4.798 \cdot 10^{-1}$	$2.2 \cdot 10^{-2}$
Spent ^{233}U, kg/year [3])	–	–	72.38	4.828	9.771	–
Spent ^{233}U with respect to its initial mass, % [3])	–	–	50	3	2	–
^{233}U without spent part, kg/year	13.25	1.906	35.14	7.068	15.07	8.128
^{233}U with spent part, kg/year	–	–	17.57	6.856	14.77	–
^{3}H, kg/year	$3.693 \cdot 10^{-1}$	$3.450 \cdot 10^{-1}$	1.296	$2.351 \cdot 10^{-1}$	$2.185 \cdot 10^{-1}$	$3.518 \cdot 10^{-1}$

[1]) 30 mol. % of $^{232}ThF_4$ and 70 mol. % of FLiNaK.

[2]) Liquid CO_2 density is 1.0299 g/cm^3 at temperature of –20^o C and pressure of 2.033 MP.

[3]) Without ^{233}U generated from ^{232}Th.

Table 4. Reaction rates and nuclear fuel production in the FNS model with two rows of pipes consisted from ^{232}Th enriched of ^{233}U. The pipes are placed in a D_2O moderator. The distance from the first wall to the 1-st pipe row is 15.5 cm and to the 2-nd is 20.5 cm. The blanket thickness is 15 cm. Fusion power is1n/s for the reaction rates and $1.775 \cdot 10^{18}$ n/s (5 MW) for the fuel production.

Variant	1	2	3	4	5
Pipe material-, vol. %	96% ^{232}Th 4% ^{233}U	97% ^{232}Th 3% ^{233}U	98% ^{232}Th 2% ^{233}U	99% ^{232}Th 1% ^{233}U	100 % ^{232}Th
Material in pipes, vol. %	D_2O	D_2O	D_2O	D_2O	D_2O
^{232}Th total mass, kg	2181.5	2204.2	2227.0	2249.7	2272.4
^{233}U total mass, kg	145.75	109.32	72.878	36.439	0.0000
Moderator material	D_2O	D_2O	D_2O	D_2O	D_2O
Blanket material	Be	Be	Be	Be	Be
(n, γ)—^{232}Th	1.621	1.440	1.062	$7.992 \cdot 10^{-1}$	$6.110 \cdot 10^{-1}$
(n, t)—$^{6}Li_2O$	1.905	1.500	1.229	1.038	$8.841 \cdot 10^{-1}$
(n, t)—^{6}Li	2.718	1.797	1.197	$7.641 \cdot 10^{-1}$	$4.326 \cdot 10^{-1}$
(n, t)—total	4.623	3.293	2.426	1.802	1.317
(n, f) + (n, xn)	3.339	1.992	1.116	$4.940 \cdot 10^{-1}$	$2.0 \cdot 10^{-2}$
Spent ^{233}U, kg/year[1])	72.38	43.18	24.19	10.71	–
Spent ^{233}U with respect to its initial mass, % [1])	50	40	33	29	–
^{233}U without spent part, kg/year	35.14	31.22	23.02	17.33	13.25
^{233}U with spent part, kg/year	17.57	18.73	15.42	12.30	–
^{3}H, kg/year	1.296	$9.237 \cdot 10^{-1}$	$6.805 \cdot 10^{-1}$	$5.055 \cdot 10^{-1}$	$3.693 \cdot 10^{-1}$

[1])Without ^{233}U generated from ^{232}Th.

The analysis of the data in Tables 2 and 3 allows concluding that the best results for ^{233}U and ^{3}H generation were obtained for the FNS model with 15 cm Be blanket, a D_2O moderator, and two rows of 100% ^{232}Th pipes located in the moderator. An important feature of this model is a relatively low rate of the reactions (n, f) and (n, xn), which are responsible for generating radioactive toxic substances. The use of the FLiNaK salt instead of metal ^{232}Th decreases the ^{233}Ugeneration by a factor of 7, the use of CO_2 instead of D_2O as a moderator decreases this value by a factor of 1.6.

The generation of ^{233}U can be increased by moving the pipes from a D_2O moderator to a Be blanket at the position where there is the maximum flux density of thermal neutrons. However, it is appeared problems of loss of the material which generates neutrons, and complexity of blanket manufacture.

Table 4 shows the accumulation of ^{233}U, which was evaluated taking into account its burnup Exposure time of a material containing thorium was 1 year. The burnup of ^{233}Pa was not considered due to the lack of information on the appropriate cross sections in the ENDF/B-6 files. The data of Table 4 indicate that the burnup of ^{233}U accumulating in the material containing 100 % (vol.) of ^{232}Th is not more than 30%. In the material containing 99% (vol.) of ^{232}Th, and 1% (vol.) of ^{233}U, the accumulation of ^{233}U is dominated. When the initial content of ^{233}U is increased to 2% (vol.) and more the burnup dominates. The ^{233}U accumulation is achieved approximately 3 mass % in relation to the total fertile and fissile material mass. That is acceptable for a nuclear fuel used in reactors.

The absorption of neutrons in the useful reactions ^{232}Th(n, γ) and ^{6}Li(n, t) reaches 80% of the total neutron loss in the system.

The calculation results of FNS models with uranium-containing solid-state blankets are presented in Table 5. There are no pipes composed of fertile materials in these models. Fertile isotopes were part of the blanket and moderator materials.

Table 5. Reaction rates and nuclear fuel production in FNS solid-state blankets with ^{238}U. The blanket thickness is 15 cm. Fusion power is 1n/s for the reaction rates and $1.775 \cdot 10^{18}$ n/s (5 MW) for the fuel production.

Variant	1	2	3	4	5	6
Moderator material	^{238}U	^{238}U	^{238}UN	^{238}UN+1%(vol.) ^{239}Pu	^{38}UN+2%(vol.) ^{239}Pu	^{38}UN+3%(vol.) ^{239}Pu
Blanket material	Be	^{238}U	^{238}UN	^{38}UN+1%(vol.) ^{239}Pu	^{38}UN+2%(vol.) ^{239}Pu	^{38}UN+3%(vol.) ^{239}Pu
(n,γ)-^{238}U	1.825	3.475	2.517	3.658	4.813	6.610
(n,f)- ^{238}U	$1.785 \cdot 10^{-1}$	$7.555 \cdot 10^{-1}$	$5.591 \cdot 10^{-1}$	1.252	1.984	3.140
(n,2n)+(n,3n)-^{238}U	$9.581 \cdot 10^{-2}$	$4.371 \cdot 10^{-1}$	$3.572 \cdot 10^{-1}$	$3.621 \cdot 10^{-1}$	$3.690 \cdot 10^{-1}$	$3.798 \cdot 10^{-1}$
^{239}Pu, kg/year	40.41	76.94	55.73	80.99	106.6	146.3

The ^{239}Pu production is significantly increased at presense of the fissile material. The models with uranium nitride show the most rates of the reactions producing ^{239}Pu, which generation is from 56 to 150 kg/year for 5 MW ($1.775 \cdot 10^{18}$ n/s) fusion power, depending on fissile isotope percentage. It is satisfactory performance of the nuclear fuel for a low-power facility. But the burnup and accumulating of fission products are increased from 2 to 6 times also.

3.2. Aqueous homogeneous blankets

The calculation model of blankets with water, or heavy water solutions of salts or oxides, which includes fertile isotopes is shown in Figure 4. The calculation results are presented in Tables 6, 7 and 8.

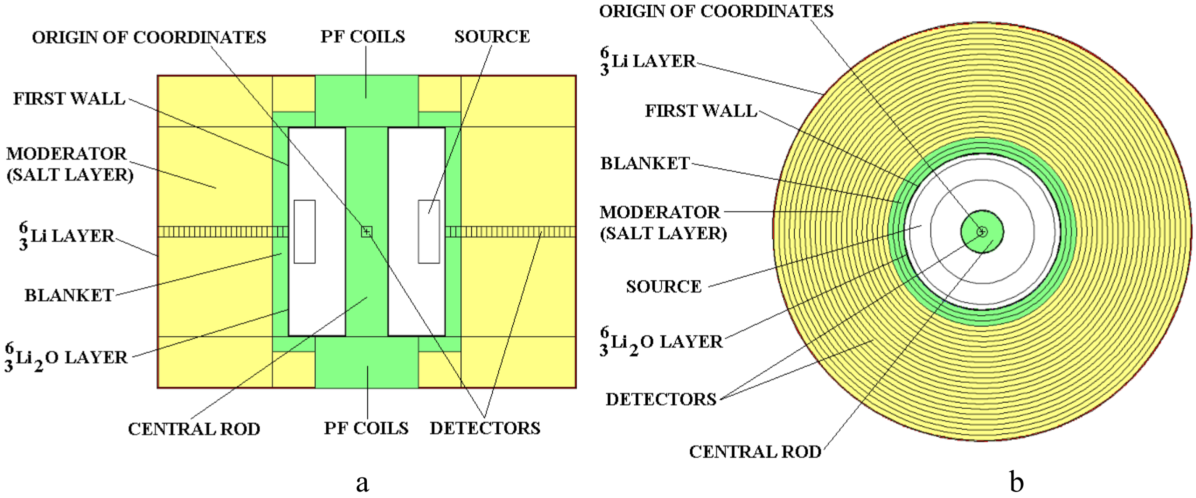


Figure 4. Vertical (a) and horizontal (b) section view of the FNS model.

Table 6. Reaction rates and nuclear fuel production in the FNS blankets with uranium salt solutions in water and heavy water. The concentration is 61.18 g of uranium salt to 100 g of H_2O or D_2O. The blanket thickness is 15 cm. Fusion power is 1 n/s for reaction rates and $1.775\cdot10^{18}$ n/s (5 MW) for fuel production.

Variant	1	2
Moderator material	$^{238}UO_2SO_4\cdot3H_2O+H_2O$	$^{238}UO_2SO_4\cdot3D_2O+D_2O$
Blanket material	$^{238}UO_2SO_4\cdot3H_2O+H_2O$	$^{238}UO_2SO_4\cdot3D_2O+D_2O$
(n,γ)-^{238}U	$7.812\cdot10^{-2}$	$3.892\cdot10^{-1}$
(n,f)- ^{238}U	$1.440\cdot10^{-2}$	$1.723\cdot10^{-2}$
(n,2n)+(n,3n)-^{238}U	$1.384\cdot10^{-2}$	$1.762\cdot10^{-1}$
^{233}Pu, kg/year	1.730	8.617

The data in Table 6 show that the use of D_2O instead of H_2O in the uranium salt solutions increases the ^{239}Pu production by a factor of 5. The ^{238}U(n,f) reaction rate isn't increased.

Table 7. Reaction rates and nuclear fuel production in the FNS blankets with uranium salt solutions in heavy water. The concentration is 61.18 g of uranium salt to 100 g of D_2O. The blanket thickness is 15 cm. Fusion power is 1 n/s for reaction rates and $1.775\cdot10^{18}$ n/s (5 MW) for fuel production.

Variant	1	2	3
Moderator material	$^{238}UO_2SO_4\cdot3D_2O+D_2O$	$^{238}UO_2SO_4\cdot3D_2O+D_2O$	$^{dep}UO_2SO_4\cdot3D_2O+D_2O$ $^{1)}$
Blanket material	Be 15 cm	Be 15 cm ^{238}U 3 cm	Be 15 cm
(n,γ)-^{238}U	$4.190\cdot10^{-1}$	$3.762\cdot10^{-1}$	$5.260\cdot10^{-1}$
(n,f)- ^{238}U	$3.867\cdot10^{-3}$	$2.169\cdot10^{-3}$	$2.078\cdot10^{-1}$
(n,2n)+(n,3n)-^{238}U	$3.312\cdot10^{-2}$	$1.823\cdot10^{-2}$	$3.476\cdot10^{-2}$

(n,t)-$^6Li_2O$	$9.408 \cdot 10^{-1}$	$7.921 \cdot 10^{-1}$	$9.684 \cdot 10^{-1}$
(n,t)-^6Li	$5.067 \cdot 10^{-1}$	$4.536 \cdot 10^{-1}$	$5.686 \cdot 10^{-1}$
(n,t)-total	1.448	1.246	1.537
^3H, kg/year	$4.062 \cdot 10^{-1}$	$3.495 \cdot 10^{-1}$	$4.311 \cdot 10^{-1}$
^{239}Pu, kg/year	9.277	8.329	11.64

[1]) ^{dep}U is depleted uranium

Table 8. Reaction rates and nuclear fuel production in the FNS blanket with thorium dioxide solutions in heavy water. The concentration is 35.95 g of $^{232}ThO_2$ to 100 g of D_2O. The blanket thickness is 15 cm. Fusion power is1 n/s for reaction rates and $1.775 \cdot 10^{18}$ n/s (5 MW) for fuel production.

Moderator material	$^{232}ThO_2$+D_2O
Blanket material	Be
(n,γ)-^{232}Th	$6.264 \cdot 10^{-1}$
(n,f)- ^{232}Th	$1.220 \cdot 10^{-3}$
(n,2n)+(n,3n)-^{232}Th	$3.675 \cdot 10^{-2}$
(n,t)-$^6Li_2O$	$9.245 \cdot 10^{-1}$
(n,t)-^6Li	$3.359 \cdot 10^{-1}$
(n,t)-total	1.260
^3H, kg/year	$3.534 \cdot 10^{-1}$
^{233}U, kg/year	13.58

According to the data of Table 7 the use of the ^{238}U shell around the Be blanket decreases the ^{239}Pu production by 10%, and the ^3H production by 14%. The use of depleted uranium ^{dep}U with 0.4% (mas.) of ^{235}U instead of ^{238}U increases the ^{239}Pu production by 25%, and the ^3H production by 6%.

The comparison of the calculated results in Tables 3 and 8 shows that the rates of the reactions (n, γ), which generate the fuel isotopes, are not much different from each other in the solid-state blanket, and the blanket with the solution of thorium dioxide in heavy water.

The rate of the reaction $^{232}Th(n, \gamma)^{233}Th$ is equal to $6.110 \cdot 10^{-1}$ for 1 D-T neutron in the solid-state blanket, and $6.264 \cdot 10^{-1}$ in the blanket with the solution of thorium dioxide in heavy water. The difference is not more than 3%.

The rate of the reaction ^6Li (n, ^4He)^3H is equal to 1.317 for 1 D-T neutron in the solid-state blanket, and 1.260 in the blanket with the solution of thorium dioxide in heavy water. The difference is not more than 4-5%.

The rate of the reactions (n, f) + (n, 2n) +(n, 3n) is 2 times more in the blanket with the thorium dioxide solution.

3.3. Molten salt and molten lead blankets

The calculation model of a molten salt blanket for ^{233}U and ^3H production is shown in Figure 4. The moderator is composed of the molten salt $^{232}ThF_4$ (30% mol.)–FLiNaK (70% mol.). The atomic weight of the salt is 121.313 g/mol and the density is 4.634 g/cm^3. Its composition is presented in Table 9.

Table 9. The chemical composition of molten salt substances. ρ is the density of the substance, η is the molar fraction of chemical elements in the substance.

ThF_4+FLiNaK, $\rho = 4.634$ g/cm^3		Natural lithium ^{nat}Li, $\rho = 5.35 \cdot 10^{-1}$ g/cm^3	
Elements	η, %	Elements	η, %
ThF_4	30.00	^7Li	92.5
^{nat}LiF	32.55	^6Li	7.5
KF	29.40	–	–
NaF	8.05	–	–

The calculation results are shown in Table 10.

Table 10. Reaction rates and nuclear fuel production in the FNS molten salt blanket consisted of $^{232}ThF_4$ (30% mol.) and FLiNaK (70% mol.). The blanket thickness is 15 cm. Fusion power is1n/s for reaction rates and $1.775 \cdot 10^{18}$ n/s (5 MW) for fuel production.

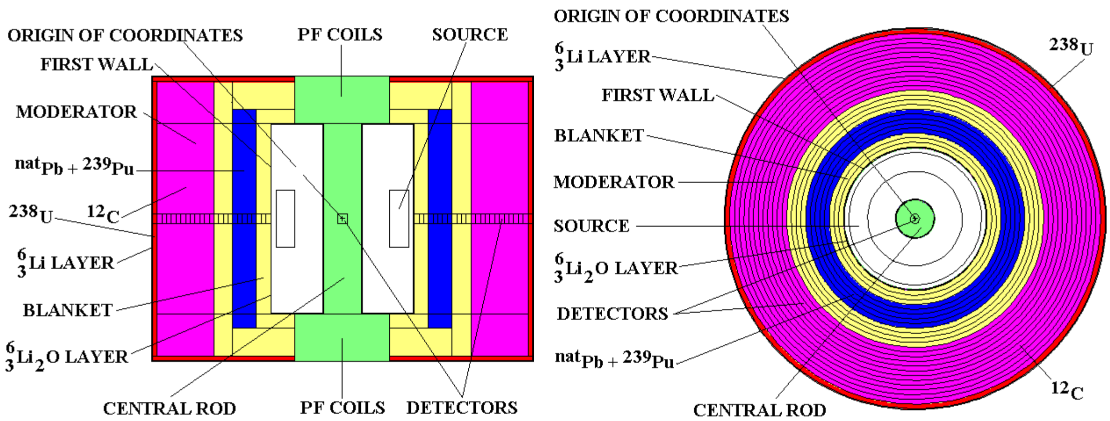
Moderator material	$^{232}ThF_4$+FLiNaK
Blanket material	Be
(n,γ)-^{232}Th	$3.973 \cdot 10^{-1}$
(n,f)- ^{232}Th	$1.034 \cdot 10^{-2}$
(n,2n)+(n,3n)-^{232}Th	$3.593 \cdot 10^{-2}$
(n,t)-$^6Li_2O$	$8.516 \cdot 10^{-1}$
(n,t)-^{6}Li	$2.977 \cdot 10^{-2}$
(n,t)-total	$8.814 \cdot 10^{-1}$
^{3}H, kg/year	$2.472 \cdot 10^{-1}$
^{233}U, kg/year	8.613

In the molten salt blanket ThF_4+FLiNaK, the rate of the reaction $^{232}Th(n, \gamma)^{233}Th$ is equal to $3.973 \cdot 10^{-1}$for 1 D-T neutron. The rate of the reaction ^6Li (n, $^4He)^3H$ is equal to $8.814 \cdot 10^{-11}$for 1 D-T neutron. The rate of the both reactions is by 30% less than in the solid-state blanket, and the blanket with the solution of thorium dioxide in heavy water.

So the molten salt blanket ThF_4+FLiNaK shows the 30% decline in the nuclear fuel production in comparison with the solid-state blanket, and the blanket with the solution of thorium dioxide in heavy water.

Also two other kindes of blankets were considered. There were a molten lead and a molten FLiBe salt blanket. These blankets were used for reprocessing of ^{239}Pu. A fuel production problem together with a capability to produce electricity and heat were considered also.

The calculation model of a molten lead blanket with ^{239}Pu dissolved in natural lead ^{nat}Pb is shown in Figure 5. The model includes 5 cm ^{238}U layer for ^{239}Pu generation, using the leakage neutrons. The calculation results of this model are presented in Table 11.



a b

Figure 5. Vertical (a) and horizontal (b) section view of the FNS model.

Table 11. Reaction rates and nuclear fuel production in the FNS blankets with ^{239}Pu dissolved in natural lead ^{nat}Pb. The blankets are consisted of two or three layers from different materials. Fusion power is1 n/s for reaction rates and $1.775 \cdot 10^{18}$ n/s (5 MW) for fuel production.

Variant	1	2	3	4
^{239}Pu mass in blanket, kg	300.6	939.5	939.5	939.5
Moderator material	^{nat}Pb	^{nat}Pb	^{nat}Pb	^{12}C
Blanket material	^{nat}Pb - 15 cm. FLiBe + 4%(mas.) ^{239}Pu - 25 cm	^{nat}Pb - 15 cm. ^{nat}Pb+2%(mas.) ^{239}Pu - 25 cm	^{nat}Pb - 15 cm. ^{nat}Pb+2%(mas.) ^{239}Pu - 25 cm	^{nat}Pb - 15 cm. ^{nat}Pb+2%(mas.) ^{239}Pu - 25 cm. ^{nat}Pb - 20 cm.
(n,γ)-^{239}Pu	$3.92 \cdot 10^{-2}$	1.350	1.387	1.774
(n,f)- ^{239}Pu	$5.80 \cdot 10^{-2}$	3.167	3.244	4.284
(n,2n)+(n,3n)-^{239}Pu	$6.33 \cdot 10^{-3}$	$9.304 \cdot 10^{-2}$	$9.269 \cdot 10^{-2}$	$9.744 \cdot 10^{-2}$
(n,t)-$^{6}Li_2O$	$5.43 \cdot 10^{-1}$	1.822	1.861	2.020
(n,t)-^{6}Li	$1.92 \cdot 10^{-1}$	$7.701 \cdot 10^{-1}$	$3.952 \cdot 10^{-1}$	$4.825 \cdot 10^{-1}$
(n,t)-FLiBe	$4.08 \cdot 10^{-1}$	—	—	—
(n,t)-total	1.14	2.592	2.257	2.503
(n,γ)-^{238}U		No ^{238}U	1.213	1.744
^{3}H production, kg/year	$3.20 \cdot 10^{-1}$	$7.271 \cdot 10^{-1}$	$6.331 \cdot 10^{-1}$	$7.021 \cdot 10^{-1}$
^{239}Pu production, kg/year	—	—	26.86	38.61
^{239}Pu reprocessing, kg/year	1.42	102.2	104.6	136.3

The FLiBe molten salt blanket with the natural lead moderator has no advantages with respect to the molten lead one with the same moderator. For the molten lead blanket the use of the graphite moderator gives the better results than the use of the natural lead one. The increase is 30% for the ^{239}Pu reprocessing, 44% for the ^{239}Pu production, and 11% for the ^{3}H production. It means that in the lead blanket the fast neutron spectrum does not take an advantage over the thermal one for reprocessing of spent fuel and production of fissile elements, because the fission reaction rate is essentially decreased. The capture-to-fission ratio for ^{239}Pu almost is not varied, and is equal to 0.428 for the thermal spectrum, and 0.407 for the fast one.

For the molten lead blanket with the graphite moderator k_{eff} reached the value of 0.95.

A high rate of spent nuclear fuel reprocessing, and nuclear fuel production can be obtained in a molten blanket, using natural lead in combination with a graphite moderator. This blanket can be used as a subcritical nuclear reactor for electricity and heat generation. But molten lead can't move inside pipes in the presence of an electromagnetic field, so a molten lead blanket design is extremely problematic.

The calculation model of a molten salt blanket with ^{239}Pu dissolved in FLiBe is the same as shown in Figure 5. The calculation results are presented in Table 12.

Table 12. Reaction rates and nuclear fuel production in the FNS blankets with ^{239}Pu dissolved in FLiBe. The blankets are consisted of two or three layers from different materials. Fusion power is1 n/s for reaction rates and $1.775 \cdot 10^{18}$ n/s (5 MW) for fuel production.

Variant	1	2	3
^{239}Pu mass in blanket, kg	939.5	939.5	704.6
Moderator material	^{12}C	^{12}C	^{12}C
Blanket material	FLiBe - 15 cm. FLiBe+11.4%(mas.) ^{239}Pu - 25 cm. FLiBe - 20 cm.	Void - 15 cm. F^7LiBe+11.4%(mas.) ^{239}Pu - 25 cm. F^7LiBe - 20 cm.	^{nat}Pb - 15 cm. F^7LiBe+8.55%(mas.) ^{239}Pu - 25 cm.
(n,γ)-^{239}Pu	$2.372 \cdot 10^{-2}$	$3.774 \cdot 10^{-1}$	7.06
(n,f)- ^{239}Pu	$4.661 \cdot 10^{-2}$	$8.047 \cdot 10^{-1}$	14.8
(n,2n)+(n,3n)-^{239}Pu	$2.770 \cdot 10^{-2}$	$4.603 \cdot 10^{-2}$	$2.81 \cdot 10^{-1}$
(n,t)-$^{6}Li_2O$	$1.340 \cdot 10^{-1}$	$4.784 \cdot 10^{-1}$	6.26
(n,t)-^{6}Li	$6.664 \cdot 10^{-2}$	$2.561 \cdot 10^{-1}$	1.94
(n,t)-FLiBe	$6.174 \cdot 10^{-1}$	$4.298 \cdot 10^{-3}$	$4.20 \cdot 10^{-4}$
(n,t)-total	$8.170 \cdot 10^{-1}$	$7.388 \cdot 10^{-1}$	8.20
(n,γ)-^{238}U	$2.192 \cdot 10^{-1}$	$8.225 \cdot 10^{-1}$	7.17
^{3}H production, kg/year	$2.292 \cdot 10^{-1}$	$2.072 \cdot 10^{-1}$	2.300
^{239}Pu production, kg/year	4.853	18.21	158.7
^{239}Pu reprocessing, kg/year	2.171	18.83	490.2

If FLiBe includes natural lithium a high rate of the ^{239}Pu reprocessing and fuel production can not be achieved due to the presence of ^{6}Li in the natural lithium mix (~ 7.5 mol. %). The isotope ^{6}Li is a strong absorber of thermal neutrons. The positive result is obtained only if the ^{6}Li isotope is extracted. If ^{7}Li is used instead of natural lithium the fission reaction rate has a considerable increase (by a factor of more than 10). The presence of 15 cm blanket layer from natural lead increases the rate of ^{239}Pu reprocessing by a factor of 26, the ^{239}Pu production by a factor of 9, the ^{3}H production by a factor of 11.

In the ^{nat}Pb- F^{7}LiBe blanket with the graphite moderator, k_{eff} is acquired a value more than 0.95.

In this blanket the ^{239}Pu reprocessing rate is 980.4 kg/year, the ^{239}Pu producing rate is 317.4 kg/year, and the ^{3}H producing rate is 4.600 kg/year for fusion power of 10 MW ($3.550 \cdot 10^{18}$ n/s). These rates of nuclear fuel production and reprocessing reach the values corresponding to the requirements for commercial use of nuclear reactors.

Summary

The neutronics analysis shows that all kinds of the considered blankets have sufficient abilities as nuclear fuel producers.

But extracting generated ^{233}U or ^{239}Pu from a solid blanket creates significant challenges and is a little attractive. An accumulation of ^{239}Pu and ^{233}U in a solid-state blanket is accompanied with their burnup. The burnup of nuclear fuel causes a builup of radiotoxic fission nuclides.

Production of plutonium creates a problem of nuclear weapon material proliferation.

The application of the thorium nuclear fuel cycle has significant challenges also [15].

^{233}U is generated by β-decay of ^{233}Pa. ^{233}Pa resulting from β-decay of a ^{233}Th isotope has great half-life (26.967 days), which causes its accumulation in a blanket material. ^{233}Pa possesses a high neutron capture cross section. In the result of neutron capture, ^{235}U is generated, creating a competition to β-decay reactions with ^{233}U generation. The ^{235}U generation requires absorption of more than 2 neutrons, which increases their consumption. The interactions of ^{235}U with neutrons result in generation of transuranium elements.

In addition, the burnup of ^{233}U leads to generation of ^{232}U ($T_{1/2}$=68.9 years), in which decay chain there are isotopes producing high energy γ-radiation such as ^{210}Bi (E_γ=1.6 MeV), ^{212}Po (E_γ=1.6 MeV), and especially problematic ^{208}Tl (E_γ=2.6 MeV). This requires development of fuel remote reprocessing and refabrication technologies. The emergence of short-lived isotopes of thorium increases radiotoxicity, which may complicate thorium recycling. Also the strong high-energy radiation of thorium fuel damages electronics. It was one of the factors that have hampered the use of ^{233}U for manufacture of nuclear weapons. Besides this, the isotope ^{232}U cannot be extracted from the spent nuclear fuel by chemical methods.

Among the fast neutron reactions in thorium the most intense is the reaction (n, 2n). It leads to a significant accumulation of an isotope ^{231}Pa in the secondary products of the reactions

^{232}Th(n, 2n) ^{231}Th(β$^-$, $T_{1/2}$ = 25.6 h) ^{231}Pa.

^{231}Pa is a α-emitter with $T_{1/2}$ = 32 480 ± 260 years, and is extremely toxic. The products of the (n, 3n) and fission reactions give an additional contribution to radiotoxic pollutions from thorium.

Interactions of ^{231}Pa with neutrons lead to generation of ^{233}Pa and ^{232}U.

To prevent the undesirable reactions, permanent removal of accumulated protactinium ^{233}Pa from an area of neutron irradiation is required. This can be done in blankets with solutions of salts or oxides of thorium in heavy water, or molten-salt blankets, using fluoride of thorium.

The comparison of the calculated results shows that the rates of the reactions generating the nuclear fuel are similar to each other in the solid-state blanket and the blanket with the solution of thorium oxide in heavy water. The molten-salt blanket ThF_4+FLiNaK is inferior to these two blankets from this viewpoint. However, it has significant advantages from the viewpoint of creating a save hybrid tokamak.

The interactions of neutrons with heavy water lead to radiolysis, which poses a serious threat of explosions and fires. The solutions circulating in the contours cause corrosion and erosion of reactor materials [16].

So today fluoride molten-salt blankets are the most suitable for nuclear fuel production. It is distinguished by fire safety, low accident probability, high radiation and thermal stability, chemical inertness with respect to water and air, easy fluidity in magnetic fields, a high boiling point and a low pressure at operating temperatures, the ability to protect surfaces of reactor components and systems from the negative impact of the molten substance and to ensure the self-healing of cracks, and so on [1, 17].

The molten salt blankets with the use of natural lead and FLiBe obtain a high rate of spent nuclear fuel reprocessing and nuclear fuel production. In these blankets it is possible to install an operating mode with $k_{eff} \sim 0, 95$ if a thermal neutron spectrum is prevailed. It means that a hybrid tokamak can work like a reactor producing electricity and heat. However, due to high sensitivity of lead to influence of an electromagnetic field, probably, FLiBe blankets can be used only.

Natural lithium which is a part of FLiBe must be cleared from a ^{6}Li isotope, which is a strong absorber of thermal neutrons.

For all kinds of blankets it is necessary to add fissile materials in fertile one to increase a yield of fissile isotopes to more than 1 nuclide per 1 D-T neutron.

The key problem of implementation of molten-salt systems is the lack of their operation experience that may result in unexpected intractable problems.

For solid-state blankets the key problem is an extraction of a considerable quantity of fission products. For blankets with heavy water solutions of salts, or oxides of uranium and thorium, the one is radiolysis, material corrosion, and sedimentation of radiotoxic elements on walls of vessels. The radiation safety problem, and vessel material resistance to corrosion are important for molten salt blankets also.

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References

[1] E. P. Velikhov, E. A. Azizov, P.N. Alekseev, M.I. Gurevich, S.A. Subbotin, A.L. Shimkevich, “Concept of «green» nuclear power engineering”, VANT, Ser. Fusion, 2013, Vol. 1, p. 5-16.

[2] E. P. Velikhov, V. A. Glukhikh, V. V. Guriev, et al., “Hybrid thermonuclear tokamak reactor for the production of fissile fuel and electricity”, Atomic Energy, Vol. 45, No. 1, 3–9 (1978).

[3] E. A. Azizov, G. G. Gladush, A. V. Lopatkin, and I. B. Lukasevich, “Tokamak based hybrid systems for fuel production and recovery from spent nuclear fuel”, Atomic Energy, Vol. 110, No. 2, June, 2011 (Russian Original Vol. 110, No. 2, February, 2011).

[4] E. P. Velikhov, E. A. Azizov (presented by B. V. Kuteev), “On Russion Strategy for Controlled Fusion”, MFERW, Princeton, NJ, USA, September 7-10, 2011.

[5] B. V. Kuteev, “Fusion for Neutrons as Necessary Step to Commercial Fusion”, MFERW, Princeton, NJ, USA, September 7-10, 2011.
[6] B. V. Kuteev, P. R. Goncharov, V. Yu. Sergeev, and V. I. Khripunov, “Intense Fusion Neutron Sources”, ISSN 1063 780X, Plasma Physics Reports, 2010, Vol. 36, No. 4, pp. 281–317.
[7] B.V. Kuteev, A.A. Borisov, A.A. Golikov et al., “Tokamak-based MW-Range Fusion Neutron Sources”, DOE Workshop on Fusion-Fission Hybrid Reactors, Gaithersburg, Maryland, USA, September 30 – October 2, 2009.
[8] B.V. Kuteev, P.R. Goncharov, V.Yu. Sergeev, and V.I. Khripunov, “Intense Fusion Neutron Sources, Plasma Physics Reports”, 2010, Vol.36, No 4, pp.281-317.
[9] H.R. Wilson, G.M. Voss, R.J. Akers et al., “The Spherical Tokamak as a Components Test Facility”, FT/3-1Ra, 20th IAEA Fusion Energy Conf., Vilamoura, Portugal, November 1 - 6, 2004.
[10] Y.-K.M. Peng, C.A. Neumeyer, P.J. Fogarty et al., “Physics and Engineering Assessments of Spherical Torus Component Test Facility”, FT/3-1Rb, 20th IAEA Fusion Energy Conf., Vilamoura, Portugal, November 1 - 6, 2004.
[11] D. Steiner, E. Cheng, R. Miller, et al., “The ARIES Fusion Neutron Source Study”, UCSD-ENG-0083 (University of California, San Diego, CA, 2000); http://aries.ucsd.edu/LIB/REPORT/ARIES1-MISC/FNS1-final.pdf.
[12] MCNP-4B: Monte Carlo N-Particle Transport Code System. RSIC Computer Code Collection, CCC-200. Oak Ridge National Laboratory, 1997.
[13] FENDL-2.1. Evaluated nuclear data library for fusion applications, INDC (NDS)-467, D.L. Aldama and Trkov, 2004.
[14] BNL-NCS-44945 (ENDF-102). Revision 10/91. ENDF-102 Data Formats and Procedures for the Evaluated Nuclear Data File ENDF-6, October 1991, Edited by R.F. Rose and C.L. Dunford.
[15] Thorium Fuel Cycle – potential benefits and challenges. – IAEA-TECDOC-1450, IAEA, Vienna, 2005.
[16] B. M. Adamson, S.E. Beall, W.E. Browning, et al, “Aqueous homogeneous reactors”. Part I. Edited by J.A. Lane, moltensalt.org›references/static/downloads/pdf/….
[17] Le Brun, C.; L. Mathieu, D. Heuer and A. Nuttin. "Impact of the MSBR concept technology on long-lived radio-toxicity and proliferation resistance". Technical Meeting on Fissile Material Management Strategies for Sustainable Nuclear Energy, Vienna 2005. Retrieved 2010-06-20.