The neutronics analysis of blankets for the hybrid fusion neutron source.

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Introduction

The design of a hybrid FNS "fusion-fission" system for manufacture of artificial nuclear fuel and nuclide transmutation is actual [1-4]. At a stage of outline designing the hybrid source of thermal neutrons with neutron multipliers on the basis of Be and Pb was considered as one of the FNS basic variants. The additional neutron generation was from fission reactions of 14 MeV neutrons with nuclei of ^{232}Th or ^{238}U and thermal neutrons with nuclei of ^{233}U or ^{239}Pu.

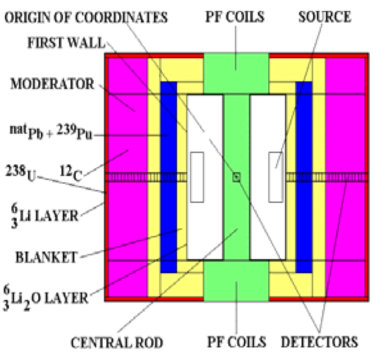
Objectives and Methods

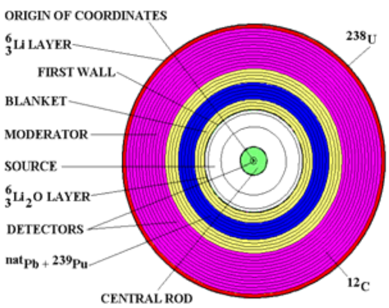
To estimate the FNS opportunities in ^{233}U and ^{239}Pu production and ^{239}Pu reprocessing the neutronics analysis was performed for the blankets with use of heavy water solutions of salts and oxides of uranium and thorium, the solid-state and molten salt blanket. The tritium generation and criticality analysis for electricity production were taken into account. FNS nuclear synthesis power was from 1 to 10 MW.

The MCNP-4 code with FENDL-2 and ENDF/B-6 point cross-section libraries was used for calculations.

Results

Vertical, horizontal and first wall section view of the basic FNS model.





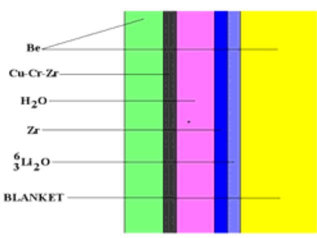


Table. Reaction rates and nuclear fuel production in FNS blankets. Fusion power is 1 n/s for reaction rates and 1.775·10^{18} n/s (5 MW) for fuel production.

The solid-state blankets with ^{238}U

Variant	1	2	3	4	5	6
Moderator material	^{238}U	^{238}U	^{238}UN	^{238}UN+ 1%(vol.) ^{239}Pu	^{238}UN+ 2%(vol.) ^{239}Pu	^{238}UN+ 3%(vol.) ^{239}Pu
Blanket material	Be	^{238}U	^{238}UN	^{238}UN+ 1%(vol.) ^{239}Pu	^{238}UN+ 2%(vol.) ^{239}Pu	^{238}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	1.78·10^{-1}	7.55·10^{-1}	5.59·10^{-1}	1.252	1.984	3.140
(n,2n)+(n,3n)^{238}U	9.58·10^{-2}	4.37·10^{-1}	3.57·10^{-1}	3.62·10^{-1}	3.69·10^{-1}	3.80·10^{-1}
^{239}Pu, kg/year	40.41	76.94	55.73	80.99	106.6	146.3

Blankets with uranium salt and thorium dioxide solutions in water and heavy water

Variant	1	2	3	4	5	6
Moderator material	^{238}UO$_2$SO$_4$·3H$_2$O +H$_2$O	^{238}UO$_2$SO$_4$·3D$_2$O +D$_2$O	^{238}UO$_2$SO$_4$·3D$_2$O +D$_2$O	^{238}UO$_2$SO$_4$·3D$_2$O +D$_2$O	^{dep}UO$_2$SO$_4$·3D$_2$O +D$_2$O	^{232}ThO$_2$
Blanket material	^{238}UO$_2$SO$_4$·3H$_2$O +H$_2$O	^{238}UO$_2$SO$_4$·3D$_2$O +D$_2$O	Be 15 cm	Be 15 cm ^{238}U 3 cm	Be 15 cm	Be 15 cm
(n,γ)-^{238}U	7.812·10^{-2}	3.892·10^{-1}	4.190·10^{-1}	3.762·10^{-1}	5.260·10^{-1}	6.264·10^{-1}
(n,f)- ^{238}U	1.440·10^{-2}	1.723·10^{-2}	3.867·10^{-3}	2.169·10^{-3}	2.078·10^{-1}	1.220·10^{-3}
(n,2n)+(n,3n)- ^{238}U	1.384·10^{-2}	1.762·10^{-1}	3.312·10^{-2}	1.823·10^{-2}	3.476·10^{-2}	3.675·10^{-2}
(n,t)-^{6}Li$_2$O	–	–	9.408·10^{-1}	7.921·10^{-1}	9.684·10^{-1}	9.245·10^{-1}
(n,t)-^{6}Li	–	–	5.067·10^{-1}	4.536·10^{-1}	5.686·10^{-1}	3.359·10^{-1}
(n,t)-total	–	–	1.448	1.246	1.537	1.260
^{3}H, kg/year	–	–	4.062·10^{-1}	3.495·10^{-1}	4.311·10^{-1}	3.534·10^{-1}
^{239}Pu, kg/year	1.730	8.617	9.277	8.329	11.64	13.58 (^{233}U)

The solid-state blankets with ^{232}Th

Variant	1	2	3	4	5	6
Pipe material, vol. %	100% ^{232}Th	FLiNaK	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$_2$O	FLiNaK	D$_2$O	^{nat}Pb	96% ^{232}Th 4% ^{233}U	CO$_2$
^{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$_2$O	D$_2$O	D$_2$O	^{nat}Pb	^{nat}Pb	CO$_2$
Blanket material	Be	Be	Be	^{nat}Pb	^{nat}Pb	Be
(n, γ)—^{232}Th	6.110·10^{-1}	8.792·10^{-2}	1.621	3.260·10^{-1}	6.953·10^{-1}	3.749·10^{-1}
(n, t)—^{6}Li$_2$O	8.841·10^{-1}	8.550·10^{-1}	1.905	6.510·10^{-1}	6.130·10^{-1}	8.712·10^{-1}
(n, t)—^{6}Li	4.326·10^{-1}	3.750·10^{-1}	2.718	1.873·10^{-1}	1.659·10^{-1}	3.827·10^{-1}
(n, t)—total	1.317	1.230	4.623	8.383·10^{-1}	7.789·10^{-1}	1.254
(n, f) + (n, xn)	2.0·10^{-2}	6.0·10^{-3}	3.339	2.227·10^{-1}	4.798·10^{-1}	2.2·10^{-2}
Spent ^{233}U, kg/year	—	—	72.38	4.828	9.771	—
Spent ^{233}U with respect to its initial mass, %	—	—	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	3.693·10^{-1}	3.450·10^{-1}	1.296	2.351·10^{-1}	2.185·10^{-1}	3.518·10^{-1}

Blankets with natural lead ^{nat}Pb and molten solt blankets

Variant	1	2	3	4
^{239}Pu mass in blanket, kg	939.5	939.5	939.5	704.6
Moderator material	^{nat}Pb	^{12}C	^{12}C	^{12}C
Blanket material	^{nat}Pb 15cm ^{nat}Pb+ 2% (mas.) ^{239}Pu 25cm ^{nat}Pb 20 cm	^{nat}Pb 15cm. ^{nat}Pb+2% (mas.) ^{239}Pu 25cm ^{nat}Pb 20 cm	FLiBe 15 cm FLiBe+11.4%(mas.) ^{239}Pu 25 cm FLiBe 20 cm	^{nat}Pb 15 cm F^{7}LiBe+8.55%(mas.) ^{239}Pu 25 cm ^{nat}Pb 20 cm
(n,γ)-^{239}Pu	1.387	1.774	2.372·10^{-2}	7.06
(n,f)- ^{239}Pu	3.244	4.284	4.661·10^{-2}	14.8
(n,2n)+(n,3n)- ^{239}Pu	9.269·10^{-2}	9.744·10^{-2}	2.770·10^{-2}	2.81·10^{-1}
(n,t)-^{6}Li$_2$O	1.861	2.020	1.340·10^{-1}	6.26
(n,t)-^{6}Li	3.952·10^{-1}	4.825·10^{-1}	6.664·10^{-2}	1.94
(n,t)-FLiBe	—	—	6.174·10^{-1}	4.20·10^{-4}
(n,t)-total	2.257	2.503	8.170·10^{-1}	8.20
(n,γ)-^{238}U	1.213	1.744	2.192·10^{-1}	7.17
^{3}H production, kg/year	6.331·10^{-1}	7.021·10^{-1}	2.292·10^{-1}	2.300
^{239}Pu production, kg/year	26.86	38.61	4.853	158.7
^{239}Pu reprocessing, kg/year	104.6	136.3	2.171	490.2

Conclusions

The best result in the nuclear fuel production and spent fuel reprocessing is obtained for the molten salt blankets using natural Pb and FLiBe. In these blankets it is possible to install an operating mode with keff ~ 0, 95 if the thermal neutron spectrum is prevailed. But molten Pb can’t move inside pipes at presence of an electromagnetic field. So the blanket design is extremely problematic.

Use of the fast neutron spectrum for the production or reprocessing of fissile isotopes in all blankets does not take an advantage over use of the thermal neutron spectrum because the fission reaction rate is essentially decreased. The key problem for the solid-state blankets is extraction of a considerable quantity of fission products, and for the blankets using heavy water solutions of salts or oxides of uranium and thorium is radiolysis and sedimentation of radio toxic elements on walls of vessels.

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