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COMPARISON OF SPALLATION AND FUSION NEUTRON SOURCES IN A FUSION-FISSION SYSTEM (FFS)

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ABSTRACT

This study analyses the effect of the external neutron source in a Fusion-Fission System (FFS), which utilizes a central fusion chamber surrounded by multiple layers (blankets) of neutron multiplier material and moderating media, which include coolant plenums, beryllium (Be) multiplier layer, a fertile fission blanket, and a graphite-pebble reflector. The aim is to investigate the neutronic behavior of the systems under different neutron sources and analyze the fuel evolution using reprocessed fuel for 10 years. For that, two neutron sources have been analyzed, both were homogeneously distributed over the entire central sphere. One of the neutrons sources simulated was the neutrons with energy of 14.1 MeV produced by the D–T fusion reactions and the other one was a spallation neutron source. Furthermore, two different reprocessed fuels by GANEX technique were used in the system for both external neutron sources, one spiked with about 79% of depleted uranium, and the other one spiked with about 75% of thorium. The spent fuels used in the simulations have the composition equivalent to spent fuel discharged from the Brazilian PWR ANGRA-I, with initial enrichment of 3.1%. The fuel burn simulation was performed in ORIGEN 2.1 code for three cycles, with the burnup of approximately 11.000 MWd/tHM in each cycle, following the ANGRA-I power history of real cycles. After that, all the simulations were performed using the SERPENT Monte Carlo code version 2.1.31. The fusion-fission system results show that the decrease in the criticality was considerably lower during the burnup when fusion source was used. Therefore, the fusion source is the best option to provide a burnup extension. However, the spallation source yielded more U-233 than the fusion source, achieving the best results for fuel regeneration.

1. INTRODUCTION

Over the last decades, Fusion-Fission System (FFS) have been proposed as an alternative to the traditional reactors [1-5]. The FFS is a fusion reactor with a blanket region containing nuclear fuel. The basic idea is to use high-energy neutrons from fusion reactions to induce fission reactions in the nuclear fuel. The FFS simulated was based on the concept [6] that combines current laser inertial confinement fusion technology with a fission reactor technology. This system utilizes a central fusion chamber surrounded by multiple layers (blankets) of neutron multiplier material and moderating media. These

layers include coolant plenums, beryllium (Be) multiplier layer, a fertile fission blanket, and a graphite-pebble reflector. The FFS design has been used in several other studies [7-9]. The aim of the present study is to compare the performance of the fusion source and the spallation source. In other words, the FFS was simulated with two neutron sources: the fusion source and the spallation source.

For both sources were used reprocessed fuel spiked with thorium and reprocessed fuel spiked with depleted uranium. The reprocessing technique used is the GANEX (Group ActiNide EXtraction) process. In this reprocessing method, the actinides are extracted as a group, reducing proliferation risks.

2. METHODOLOGY

The simulations were performed using the SERPENT Monte Carlo code version 2.1.31 [10]. A useful feature of the SERPENT code is the capability to allow an external neutron source, which is very convenient to accurately model a subcritical system [11]. The external source simulation is physically consistent, in that no artificial modifications are needed to complete the calculation. The results are unbiased in energy, space, and time. The average total number of histories generated from N_0 source neutrons are given by [12]:

$$N = N_0(1 + k + k^2 + \cdots) = \frac{N_0}{1 - k}$$

where k is the neutron multiplication factor of the subcritical system.

The neutron spectra generated in the central sphere normalized to 1 have been printed in Fig. 1 for both fusion and spallation sources. As can be seen, the fusion source has the greatest number of neutrons with about 14 MeV of energy, while the spallation source spectrum lies between 10E-08 and 10E+01.

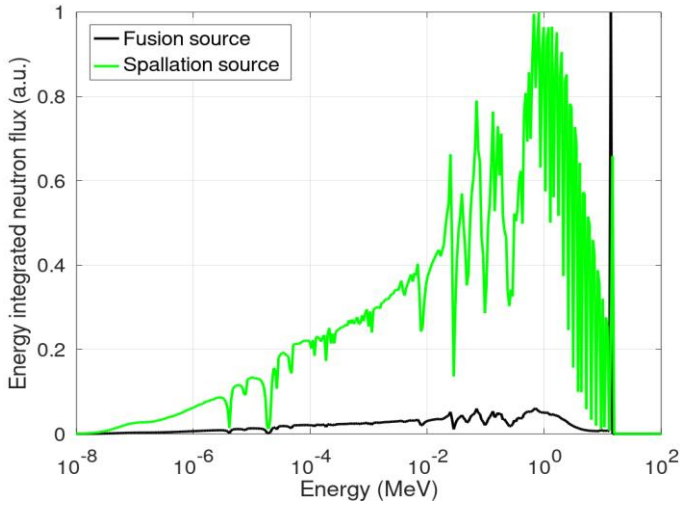

Fig. 1. Neutron flux in the cavity for both external neutron sources.

2.1.Geometry and Materials

Fig. 2 presents the geometry of the FFS that was based on concentrically spheres with dimensions as purposed by [6]. This type of system operates in the subcritical range, therefore an external neutron source is needed because fission chain reactions are not enough to keep the thermal power. For this reason, it is considered inherently safe for fuel breeding and waste burnup capabilities.

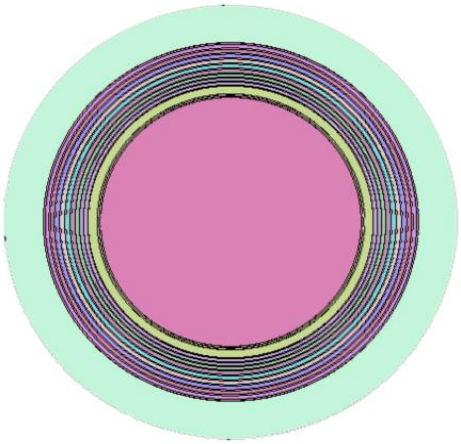

Fig. 2. Fusion-fission system cross-section.

Both external sources simulated were homogeneously distributed over the entire central sphere, one of them was neutrons of 14.1 MeV produced by the Deuterium–Tritium fusion reactions and the other one was a spallation neutron source. More details about the spallation source can be found in [13]. The material used in the system are: a tungsten alloy for the first wall, a mixture of blankets with $Li_{17}Pb_{83}$ as a coolant, and $2LiF + BeF_2$ and graphite as a reflector. Finally, the structure is made based on ODS ferritic steel (ODS-FS), as shown in Tab. 1, where is possible to see all the dimensions of the system components and their respective materials.

The fuels used in the simulations have the composition equivalent to spent fuel discharged from the Brazilian PWR ANGRA-I, with initial enrichment of 3.1%. This fuel was burned in ORIGEN 2.1 code for three cycles, with the burnup of approximately 11.000 MWd/tHM in each cycle, following the ANGRA-I power history of real cycles 1, 2, and 3. More details of this spent fuel can be found in [14]. After cooling for five years, the spent fuel was reprocessed by GANEX technique.

The GANEX process developed by *Commissariat `a l´ énergie atomique et aux energies alternatives* (CEA) for the reprocessing of Generation IV spent nuclear fuels is composed of two extraction cycles following the dissolution of the spent fuel. Once the uranium is selectively extracted from the dissolution solution by a monoamide solvent, the transuranic elements (Np, Pu, Am, and Cm) are separated from the fission products in a second cycle prior to the co-conversion step [15].

Tab. 1. Geometric parameters of the Fusion-fission system according to [9].

Zone name	Radius (cm) Inner - Outer	Material	Zone name	Radius (cm) Inner - Outer	Material
Cavity	000.000 - 250.000	Air	Region 8	297.200 - 306.700	Fuel
1ª wall	250.000 - 250.025	W1.1TiC	Region 9	306.700 - 309.200	$Li_{17}Pb_{83}$
Wall	250.025 - 250.300	ODS-FS	Region 10	309.200 - 316.700	Fuel
Coolant	250.300 - 253.300	LiPb	Region 11	316.700 - 319.200	$Li_{17}Pb_{83}$
2ª wall	253.300 - 253.600	ODS-FS	Region 12	319.200 - 326.700	Fuel
Coolant	253.600 - 256.600	Flibe	Region 13	326.700 - 329.200	$Li_{17}Pb_{83}$
3ª wall	256.600 - 256.900	ODS-FS	Region 14	329.200 - 336.700	Fuel
Coolant	256.900 - 272.900	Be Flibe	Region 15	336.700 - 339.200	$Li_{17}Pb_{83}$
4ª wall	272.900 - 273.200	ODS-FS	Region 16	339.200 - 346.200	Fuel
Region 1	273.200 - 275.200	$Li_{17}Pb_{83}$	Region 17	346.200 - 349.200	$Li_{17}Pb_{83}$
Region 2	275.200 - 281.200	Fuel	Region 18	349.200 - 356.218	Fuel
Region 3	281.200 - 283.200	$Li_{17}Pb_{83}$	Region 19	356.218 - 359.460	$Li_{17}Pb_{83}$
Region 4	283.200 - 289.200	Fuel	Back wall	359.460 - 359.960	ODS-FS
Region 5	289.200 - 291.200	$Li_{17}Pb_{83}$	Reflector	359.960 - 434.960	Graphite
Region 6	291.200 - 297.200	Fuel	Final wall	434.960 - 435.460	ODS-FS

In terms of GANEX results, according to [16], neptunium, plutonium, americium, and curium are recovered altogether in one liquid flow and the losses are estimated at a value lower than 0.5% (neptunium essentially), corresponding to a recovery yield of actinides higher than 99.5%. The decontamination factors versus some lanthanides (especially Nd, Sm, and Eu) are much lower than expected and the mass of lanthanides in the actinide product is around 5% at the end. The amount of uranium after the reprocessing is 0.01% of the total amount of uranium in the spent fuel. The isotopic composition after the GANEX reprocessing is shown in Tab. 2.

Tab. 2. Fuel composition (normalized) after GANEX reprocessing.

Nuclide	**Weight fraction**	**Nuclide**	**Weight fraction**
^{234}U	1.3812E-06	^{241}Am	7.4082E-03
^{235}U	7.1884E-05	^{242}Am	1.3634E-05
^{236}U	3.6746E-05	^{242}Cm	2.3068E-03
^{238}U	8.7385E-03	^{244}Cm	2.6445E-03
^{233}U	1.8485E-11	^{245}Cm	9.2022E-05
^{237}U	5.2354E-08	^{237}Np	4.2305E-02
^{238}Pu	1.6537E-02	^{238}Np	6.9311E-05
^{239}Pu	4.3304E-01	^{239}Np	4.3763E-03
^{240}Pu	1.4804E-01	Nd	1.0993E-02
^{241}Pu	1.3919E-01	Sm	2.2045E-03
^{242}Pu	5.2604E-02	Eu	4.6878E-04
^{243}Am	1.0060E-02	O	1.1879E-01

In this work, one of the fuels was GANEX reprocessed fuel spiked with 75% of thorium. The other fuel was spiked with 79% of depleted uranium. The compositions are shown in

Tab. 3 and 4 respectively. The percentages of thorium and depleted uranium were chosen in order to achieve the same initial k_{eff} for each neutron source simulated with both fuels.

Tab. 3. Composition of the reprocessed fuel spiked with thorium.

Isotope	**Mass Fraction**	**Isotope**	**Mass Fraction**	**Isotope**	**Mass Fraction**
^{232}Th	6.6264E-01	^{238}Np	1.7051E-05	^{243}Am	2.4747E-03
^{233}U	4.5473E-12	^{239}Np	1.0766E-03	^{242}Cm	5.6748E-04
^{234}U	3.3978E-07	^{238}Pu	4.0682E-03	^{244}Cm	6.5055E-04
^{235}U	1.7684E-05	^{239}Pu	1.0653E-01	^{245}Cm	2.2637E-05
^{236}U	9.0396E-06	^{240}Pu	3.6418E-02	^{143}Nd	2.7043E-03
^{237}U	1.2879E-08	^{241}Pu	3.4242E-02	^{147}Sm	5.4231E-04
^{238}U	2.1497E-03	^{242}Pu	1.2941E-02	^{153}Eu	1.1532E-04
^{237}Np	1.0407E-02	^{241}Am	1.8224E-03	O	1.2058E-01

Tab. 4. Composition of the reprocessed fuel spiked with uranium.

Isotope	**Mass Fraction**	**Isotope**	**Mass Fraction**	**Isotope**	**Mass Fraction**
^{232}Th	0	^{238}Np	1.4213E-05	^{243}Am	2.0629E-03
^{233}U	3.7905E-12	^{239}Np	8.9741E-04	^{242}Cm	4.7303E-04
^{234}U	2.8323E-07	^{238}Pu	3.3911E-03	^{244}Cm	5.4229E-04
^{235}U	1.4163E-03	^{239}Pu	8.8799E-02	^{245}Cm	1.8870E-05
^{236}U	7.5352E-06	^{240}Pu	3.0357E-02	^{143}Nd	2.2542E-03
^{237}U	1.0736E-08	^{241}Pu	2.8543E-02	^{147}Sm	4.5206E-04
^{238}U	7.0116E-01	^{242}Pu	1.0787E-02	^{153}Eu	9.6128E-05
^{237}Np	8.6751E-03	^{241}Am	1.5191E-03	O	1.1853E-01

3. RESULTS

The evolution of the multiplication factors during the 10 years of burnup is presented in Fig. 3. The FFS using the fusion source has an initial $k_{eff} \approx 0.95$ for both fuels meanwhile if the spallation source is used the initial k_{eff} is around $k_{eff} \approx 0.87$. The higher criticality values are achieved when the fusion source was used. That behavior can be explained by the fact that neutrons emitted by fusion have higher energy than the neutrons produced by spallation reactions. Therefore, it is expected a larger number of fission reactions when fusion source is used. The decrease in the criticality is considerably lower during the burnup when fusion source was used. This indicates that the usage of the fusion source could potentially enable additional burnup extension.

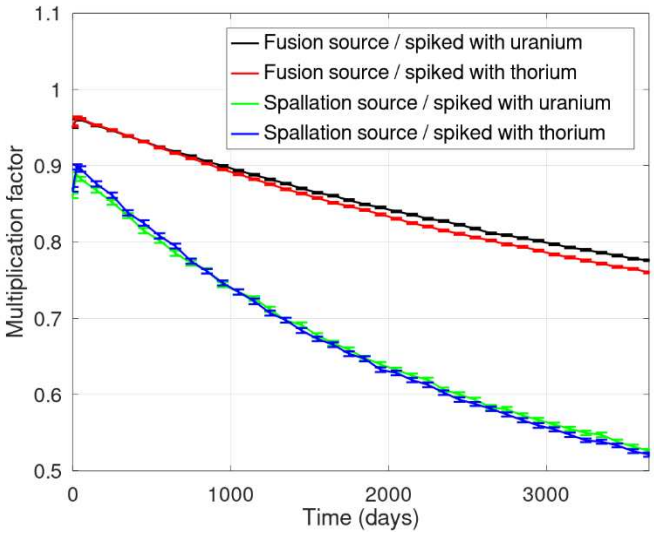

Fig. 3. Multiplication factor for the fusion-fission system during 10 years.

Fig. 4 shows the neutron flux in the reprocessed fuel zone (region 1 to 19) throughout the burnup for all the simulations. It can be seen that most of the neutron flux lies between 1E-02 and 1E+01 MeV in all fuels during the burnup, which means that the system presented a hard neutron spectrum in all simulations. From the flux results, it can be observed that the spectrum is significantly affected by the increase in burnup. It can be noted that the neutron spectra became slightly harder after ten years of burnup than at the beginning of life. This indicates that the plutonium quantities increase as it shall be demonstrated.

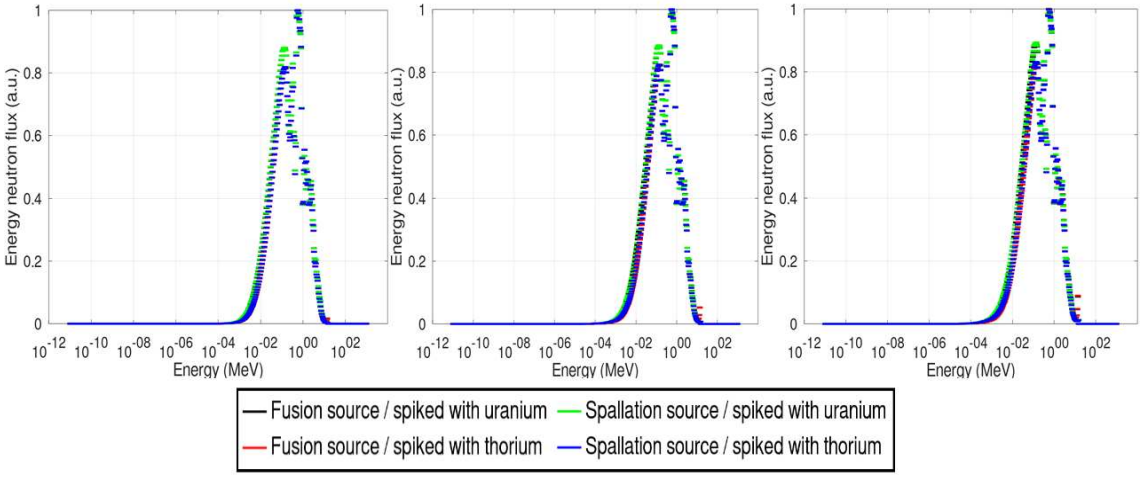

Fig. 4. Neutron flux in the transmutation zone (a) at the beginning of life (b) after 5 years (c) after 10 years of burnup.

Fig. 5 (a) and (b) present the mass variation of some nuclides, that are important to evaluate the fuel breeding and transmutation, after ten years of burnup. It can be verified that about 800 kg of ^{232}Th were consumed in the systems with reprocessed fuels spiked with thorium. This is the result of the high capture cross-section of that isotope, which

allows the ^{233}U production (~770 kg). When the fuels were spiked with uranium, the consumption of ^{238}U (~ 880 kg) allows ^{239}Pu production (~600 kg). ^{239}Pu is formed when ^{238}U captures a neutron, and it soon undergoes two beta decays. Therefore, the use of reprocessed fuel spiked with thorium and reprocessed fuel spiked with uranium are efficient for nuclear fuel breeding.

Comparing the neutron sources, both are similar in fissile isotopes production. The spallation source is slightly more efficient with 770 kg in the ^{233}U production, while fusion source achieves 761 kg. The same behavior is observed for ^{239}Pu. Fusion source yields $\approx$ 600 kg and spallation source achieves $\approx$ 607 kg. Thus, the exchange of the neutron source does not have considerable differences in the production of the fissile isotopes, both achieve solid results for fuel regeneration.

From Fig. 5 (b) it is also possible to observe some high radiotoxicity nuclides are transmuted (^{241}Pu, ^{242}Cm, and ^{238}Np). The decrease in the amount of ^{238}Np, is mainly due to ^{238}Pu production by β decay, which justifies the high ^{238}Pu production when reprocessed fuel spiked with thorium is used. ^{235}U is produced for all cases.

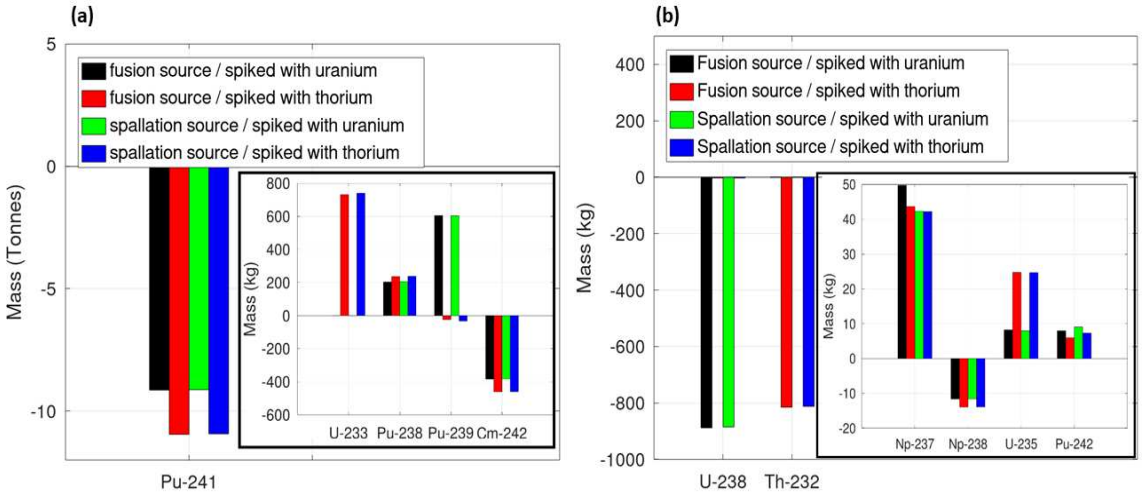

Fig. 5. Nuclides transmutation (-) or production (+).

Fig. 6 shows the total inhalation radiotoxicities evolutions during the burnup for the four cases simulated in the fusion-fission system. The inhalation radiotoxicity was reduced during the burnup mainly due to the ^{241}Pu transmutation, but the reprocessed fuel spiked with uranium presents lower values than the reprocessed fuel spiked with thorium. It was possible because the ^{238}Pu production reaches larger amounts in the fuel spiked with thorium than in the fuel spiked with uranium, as well as in ^{241}Am production. Both of them have a high level of radiotoxicity and are the reasons for the fuel spiked with thorium higher inhalation radiotoxicity.

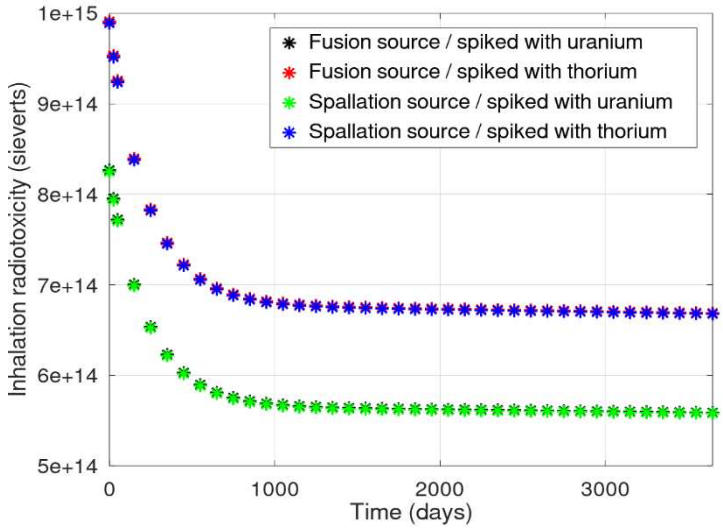

Fig. 6. Total inhalation radiotoxicity in Sieverts.

4. CONCLUSIONS

In this study, a Fusion-Fission hybrid system based on inertial confinement has been simulated. Simulations with SERPENT Monte Carlo code was performed for both fusion neutron and spallation sources using two different fuels reprocessed by GANEX technique for each source. FFS results show that the decrease in the criticality is considerably lower during the burnup when fusion source was used. Therefore, fusion source is the best option to provide a burnup extension.

However, when the spallation source was used together with reprocessed fuel spiked with thorium, it presented a slightly higher production of ^{233}U achieving almost 770 kilograms, while using fusion source the production was about 761 kilograms. The same behavior is observed in ^{239}Pu production. Thus, the exchange of the neutron source does not have considerable differences in the fissile isotopes production, both have achieved solid results for fuel regeneration.

The inhalation radiotoxicity is drastically reduced due to the decrease of high radiotoxicity inventory in all simulations. Although, the accented decrease k_{eff} values during the burnup when spallation source was used shows that the best choice for the FFS indeed is the fusion source. Another negative factor about using the spallation source in the fusion-fission system is the highest ^{238}Pu production.

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