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FISSILE FUEL BREEDING IN THE ARIES-ST FUSION REACTOR BY USING MOLTEN SALT WITH UF_4

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ABSTRACT

Fissile fuel breeding in the ARIES-ST of the 1000 MW_{el} power plant is investigated by using molten salt containing UF_4. Calculations are done with the aid of one-dimensional code of SCALE4.3. In this hybrid model, a substantial amount of fissile fuel can be produced with a fissile fuel breeding ratio of $^{239}Pu = 0.115$ per incident neutron at start-up conditions, that corresponds to 3558 kg ^{239}Pu/year by a full fusion power of 2740 MW. Tritium breeding ratio is found as 1.14 so that tritium self-sufficiency is maintained for DT fusion driver.

Key Words: ARIES-ST fusion reactor, molten salt, fissile fuel breeding.

UF_4 İÇEREN ERİYİK TUZ KULLANARAK ARIES-ST FÜZYON REAKTÖRÜNDE FİSİL YAKIT ÜRETİMİ

ÖZET

UF_4 içeren eriyik tuz kullanarak 1000 MW_{el} gücündeki ARIES-ST reaktöründe fisil yakıt üretimi incelenmiştir. Hesaplamalar tek boyutlu SCALE4.3 kodu yardımıyla yapılmıştır. Bu hibrid modelde, önemli miktarda fisil yakıt üretilebilecektir. Başlangıç şartlarındaki nötron başına fisil yakıt üretim oranı $^{239}Pu = 0.115$'tir ve bu da 2740 MW'lık tam füzyon gücü altında 3558 kg ^{239}Pu/yıl demektir. Trityum üretim oranı 1.14 olarak bulunmuştur ve bundan dolayı DT füzyon kaynağı için trityumun kendi kendine yeterli olması sağlanmaktadır.

Anahtar Kelimeler: ARIES-ST füzyon reaktörü, eriyik tuz, fisil yakıt üretimi.

1. INTRODUCTION

ARIES-ST is a 1000 MW_{el} fusion power plant design based on the spherical tokamak concept which has many attractive features, including high beta and power density, low magnetic field, high self-driven current fraction and a compact power core(1). Many studies have been done to improve the performance of the ARIES-ST fusion reactor(1-5). However the penetration of a competitive pure fusion reactor into the energy market is not expected before the year ~ 2050.

The fusion-fission hybrid is a combination of the fusion and fission processes, having features, which are complementary. The idea is to surround the fusion plasma with a blanket made of the fertile materials (^{238}U or ^{232}Th) to convert them into fissile materials (^{239}Pu or ^{233}U) by transmutation through the capture of the high yield fusion neutrons. Under the irradiation of the high energetic 14 MeV- (D,T) neutrons, the fertile materials may also undergo a substantial amount of fission. Also, the bred fissile material can be partly burnt in the hybrid blanket "in situ" and partly it can be extracted as an additional fuel supply for the existing great number of light water reactors (LWRs). The multiplication of the total plant energy together with fissile fuel production can lead to a commercial hybrid plant driven by a low performance fusion reactor.

Recently, the hybrid models of different fusion reactor design concepts, namely the hybrid PROMETHEUS reactor(6) and hybrid-ARIES-RS(7) have been presented. This study presents the fissile fuel breeding in the hybrid model of the ARIES-ST design using heavy molten salt containing UF_4.

2. HYBRID MODELING OF THE ARIES-ST

The ARIES-ST consists of high temperature shield following first wall in the inner blanket where breeding blanket does not exist. The outer blanket has an advanced 'dual cooled' breeding blanket with flowing $Li_{17}Pb_{83}$ and He-cooled ferritic steel structures. Tritium breeding zone following immediately first wall is situated with a thickness of 100 cm in the outer blanket. He manifolds and low temperature shield follow thereafter.

LiPb is a good tritium breeding material which is compatible to a water-cooled component. It has also good nuclear heating removal capability. However, LiPb has a limited surface heat removal capability due to its low thermal conductivity. Therefore, another coolant, helium is used for first wall cooling. SiC is used to keep hot LiPb away from the ferritic steel(3).

In ARIES-ST reactor, low activation ferritic steel(with 9% Cr, 2% W, 0.25% V, 0.07% Ta, 0.1% C) is chosen as main structure material. Borated stainless steel is also used as structural material in the low temperature shield. After the decommissioning of the fusion power plant, these structural materials end up with class C nuclear waste according to the 10 CFR 61 regulations (8) that means suitable for shallow burial purposes. Table 1 gives the material compositions and densities of the ARIES-ST.

The breeding zone has four quadratic channels each of which has a thickness of 25 cm. In hybrid modeling, for simple change from the basic ARIES-ST design, the outer three breeding zone channels are replaced with molten salt containing flibe and UF_4 to breed fissile fuel, without making any change elsewhere in the pure fusion ARIES-ST reactor. Figure 1 shows the one-dimensional hybrid modeling of the ARIES-ST.

Using two different fluids in the quadratic channels may lead to the requirement of more complicated design of ARIES-ST. However, a hybrid reactor due to its fissile fuel breeding capability and more higher fusion energy multiplication is more advantageous than a fusion reactor with a low performance.

3. CALCULATIONAL PROCEDURE

Numerical calculations are performed with the aid of the SCALE4.3 SYSTEM using the 238 groups library, derived from ENDF/B-V (9). The neutron transport calculations are performed by solving the Boltzmann transport equation with transport code XSDRNPM (10) in S_8-P_3 approximation by using Gaussian quadratures (11). The numerical output of XSDRNPM is evaluated with XSCALC (12) to determine the main reactor parameters.

The resonance calculations in the fissionable fuel element cell are performed with

* BONAMI (13) for unresolved resonances and

* NITAWL-II (14) for resolved resonances.

In order to produce the resonance self-shielded weighted cross-sections for XSDRNPM, CSAS control module (15) is used.

4. NUMERICAL RESULTS

TBR (Tritium Breeding Ratio) is one of the most important design parameters in the hybrid reactor that should be taken into account. It should be higher than 1.05 for tritium self-sufficiency of DT fusion driver. Although the molten salt zone with a thickness of 75 cm replaced in the outer blanket, TBR is obtained as 1.14 for the investigated hybrid reactor. Therefore tritium sef-sufficiency is maintained for the DT fusion driver.

M (Energy Multiplication Factor) is defined as the ratio of the total energy release in the blanket to the incident fusion neutron energy. Total energy release in blanket can be calculated as

$$\text{Total energy release in blanket} = 200\Sigma_f + 4.784\, T_6 - 2.467\, T_7 \quad [1]$$

where, Σ_f is total fission rate, T_6 and T_7 are tritiums produced by $^6Li(n,t)$ and $^7Li(n,n't)$ reactions, respectively. Fission rate in the molten salt zone is 7.08 10^{-3} that is very low. Therefore the contribution to energy amplification in the molten salt region is low due to its very low fission rate. M values are 1.41 and 1.48 for ARIES-ST pure fusion reactor and hybrid model of ARIES-ST, respectively. Neutron leakage out of the blanket is $7.67.10^{-7}$ that is very low.

5. FISSILE FUEL BREEDING

The ^{239}Pu fissile fuel can be produced by $^{238}U(n,\gamma)$ reaction. Figure 2 illustrates the fissile fuel (^{239}Pu) breeding rate per incident neutron per cm^3 that decreases towards to outer end of the molten salt due to softening of neutron flux by deeper penetration (see Figure 3).

One-dimensional SCALE calculations yield a fissile fuel breeding rate of $^{238}U(n,\gamma)^{239}Pu$ = 0.115 per incident fusion neutron at start-up conditions, which corresponds to 3558 kg ^{239}Pu/year. However, previous studies on long-term plant operation of hybrid blankets show that this high fissile fuel production rate would decrease rapidly, because of the burn-up of the new fissile fuel in situ (16-18). Hence, the reduction in fissile fuel breeding can be more than by a factor of 2 after a plant operation of 1 year (19). The fissile fuel breeding rate is much more higher than the fission rate. Therefore, the main aim of the blanket is fissile fuel breeding rather than energy production.

In previous studies, fissile fuel breeding was investigated by using solid compounds of thorium and uranium in hybrid versions of pure fusion reactors, namely PROMETHEUS (6) and ARIES-RS (7). The tritium breeding zone of PROMETHEUS-H IFE reactor having thickness of 50 cm, was divided into three parts as 15 cm, 12 cm and 27 cm. Then, in order to breed fissile fuel from fertile fuel, fissile fuel breeding zone containing fuel spheres filled with fertile fuel and cladded with SiC, were located instead of the second part of the tritium breeding zone having a thickness of 12 cm.

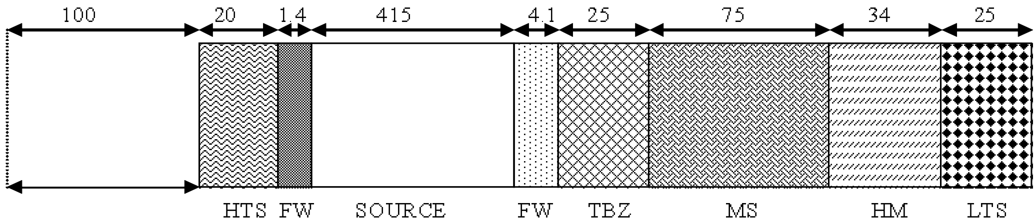

Figure 1. One-dimensional hybrid modeling of the ARIES-ST reactor (HTS = High Temperature Shield, FW=First Wall, TBZ = Tritium Breeding Zone, MS = Molten Salt, HM = Helium Manifolds, LTS= Low Temperature Shield,). Dimensions are given in cm, not in scale.

Table 1. Composition and dimension of materials in the zones of the blanket

Zone	Material	Species	Atomic densities ($10^{24}cm^{-3}$)
High temperature shield	80 % Ferritic steel	Cr	$5.980 \cdot 10^{-3}$
	20% Helium	W	$1.329 \cdot 10^{-3}$
		V	$1.661 \cdot 10^{-4}$
		Ta	$4.651 \cdot 10^{-5}$
		C	$6.645 \cdot 10^{-6}$
		Fe	$5.886 \cdot 10^{-2}$
		He	$2.167 \cdot 10^{-4}$
First wall (inner)	70 % Helium	Cr	$2.243 \cdot 10^{-3}$
	30 % Ferritic steel	W	$4.984 \cdot 10^{-4}$
		V	$6.229 \cdot 10^{-5}$
		Ta	$1.744 \cdot 10^{-5}$
		C	$2.492 \cdot 10^{-6}$
		Fe	$2.207 \cdot 10^{-2}$
		He	$7.586 \cdot 10^{-4}$
First wall (outer)	60 % Helium	Cr	$2.990 \cdot 10^{-3}$
	40 % Ferritic steel	W	$6.645 \cdot 10^{-4}$
		V	$8.306 \cdot 10^{-5}$
		Ta	$2.326 \cdot 10^{-5}$
		C	$3.322 \cdot 10^{-6}$
		Fe	$2.943 \cdot 10^{-2}$
		He	$6.502 \cdot 10^{-4}$
Helium manifold	10 % Ferritic steel	Cr	$7.475 \cdot 10^{-4}$
	90 % Helium	W	$1.661 \cdot 10^{-4}$
		V	$2.076 \cdot 10^{-5}$
		Ta	$5.814 \cdot 10^{-6}$
		C	$8.306 \cdot 10^{-7}$
		Fe	$7.357 \cdot 10^{-3}$
		He	$9.753 \cdot 10^{-4}$

Table 1. continued

Tritium breeding zone	76 % $Li_{17}Pb_{83}$ (containing 60 % enriched Li^6)	^6Li	$2.538 \cdot 10^{-3}$
	12 % SiC (75 % dense)	^7Li	$1.692 \cdot 10^{-3}$
	6 % Helium	Si	$4.336 \cdot 10^{-3}$
	6 % Ferritic steel	C	$4.337 \cdot 10^{-3}$
		He	$6.502 \cdot 10^{-5}$
		Cr	$4.485 \cdot 10^{-4}$
		W	$9.967 \cdot 10^{-5}$
		V	$1.246 \cdot 10^{-5}$
		Ta	$3.488 \cdot 10^{-6}$
		Fe	$4.414 \cdot 10^{-3}$
Molten salt	76 % molten salt $(Li_2BeF_4)_{0.88} \cdot (UF_4)_{0.12}$	^{235}U	$1.047 \cdot 10^{-5}$
	12 % SiC (75 % dense)	^{238}U	$1.486 \cdot 10^{-3}$
	6 % Helium	^6Li	$1.646 \cdot 10^{-3}$
	6 % Ferritic steel	^7Li	$2.030 \cdot 10^{-2}$
		Be	$1.097 \cdot 10^{-2}$
		F	$4.988 \cdot 10^{-2}$
		Si	$4.336 \cdot 10^{-3}$
		C	$4.337 \cdot 10^{-3}$
		He	$6.502 \cdot 10^{-5}$
		Cr	$4.485 \cdot 10^{-4}$
		W	$9.967 \cdot 10^{-5}$
		V	$1.246 \cdot 10^{-5}$
		Ta	$3.488 \cdot 10^{-6}$
		Fe	$4.414 \cdot 10^{-3}$
Low temperature shield	60 % H_2O	H	$4.015 \cdot 10^{-2}$
	25 % Borated stainless steel	O	$2.008 \cdot 10^{-2}$
	15 % Ferritic steel	Mn	$4.397 \cdot 10^{-4}$
		Si	$2.199 \cdot 10^{-4}$
		Cr	$5.298 \cdot 10^{-3}$
		Ni	$2.198 \cdot 10^{-3}$
		B	$2.198 \cdot 10^{-4}$
		Fe	$2.577 \cdot 10^{-2}$
		W	$2.492 \cdot 10^{-4}$
		V	$3.115 \cdot 10^{-5}$
		Ta	$8.721 \cdot 10^{-6}$
		C	$1.246 \cdot 10^{-6}$

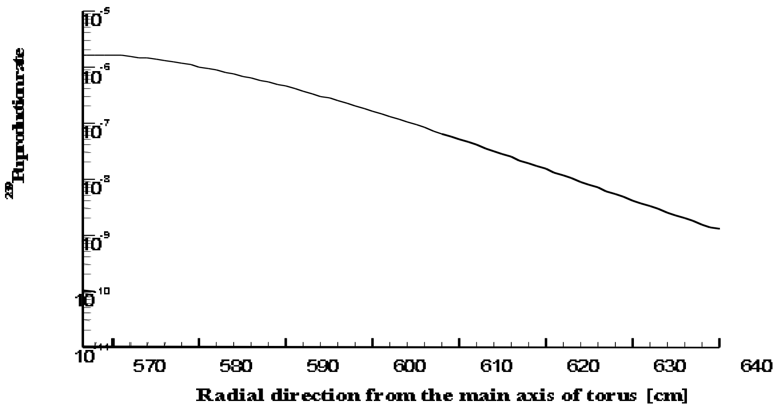

Figure 2. ^{239}Pu production rate per incident neutron per cm^3 in the molten salt zone

^{239}Pu production ratio changed from 0.118 to 0.168 according to the uranium fuel types much and the blanket fueled with UN had the highest ^{239}Pu breeding value while UO_2 had the lowest one in the hybrid version of the PROMETHEUS reactor (6).

In the hybrid version of the ARIES-RS reactor, fission zone with a thickness of 10 cm at the inner blanket conducted to a blanket multiplication of M = 3.03 with UC fuel and increased the fusion power from 2170 MW to 6500 MW. Despite a partial replacement of the lithium zone by the fissile zone, tritium breeding remained still > 1.05, which was required for a self-sustaining fusion driver. In addition to fusion power amplification, substantial fissile material was produced at start-up conditions with a fission breeding rate of ^{239}Pu = 0.263 (with UC) per incident fusion neutron, which corresponded to 6500 kg ^{239}Pu/year by a full fusion power of 2170 MW (7).

Figure 3 depicts the neutron spectrums at selected locations in the outer blanket. Along with the neutron penetration in the blanket, a continuous degradation of the fusion neutron peak and a spectrum softening occurs. Especially in the highly absorbing molten salt zone, the fast neutron fluxes decrease in the radial direction while the lower energy group fluxes increase. Generally, one can say that the neutron flux curves show a variation towards the outer boundary from the harder neutron spectrum shapes to the softer ones.

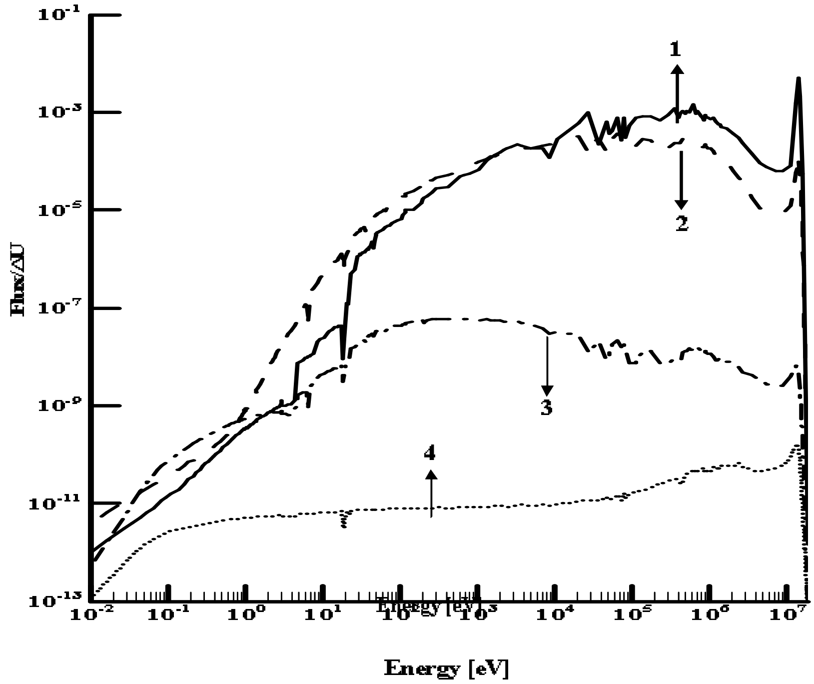

Figure 3. Neutron spectrums at selected locations of the outer blanket 1) first wall, 2) at the outer end of the tritium breeding zone, 3) at the outer end of the molten salt zone, 4) at the outer end of the low temperature shield

6. HEAT GENERATION

As it can be seen from equation(1), heat generation mainly depends on fission ratio and tritium production reactions of Li isotopes. Figure 4 shows the heat generation profile of the external driven blanket. Heat generation decreases exponentially from inner beginning to outer end of the tritium breeding zone and then peaks (~ 6 W/cm^3) at the inner beginning of the molten salt zone. Finally it sharply decreases towards to outer end of the molten salt.

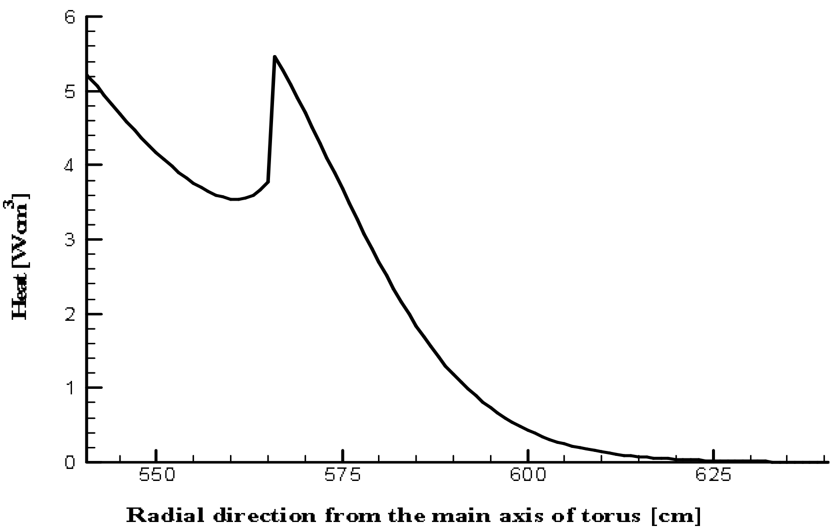

Figure 4. Heat generation profile for neutron load of 4.2 MW/m^2 at the outer first wall

7.CONCLUSIONS and RECOMMENDATIONS

The main conclusions can be outlined as follows:

* The use of molten salt containing UF_4 leads to a significant amount of ^{239}Pu, fissile fuel, production in ARIES-ST reactor.

* Tritium self-sufficiency is maintained for the DT driver in this hybrid model.

* Due to very low fission ratio, the increase in the energy production is very low and the replacement of heavy molten salt zone with a thickness of 75cm following the tritium breeding zone in the outer blanket raises the energy multiplication of the reactor slightly.

As a result, substantial amount of fissile fuel production is possible by using heavy molten salt in the ARIES-ST fusion reactor with maintaining sufficient TBR value for DT driver. For further studies, radiation damage to structural materials, especially inner and outer first walls, can be investigated.

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