Fusion Engineering and Design 87 (2012) 1385–1389

Contents lists available at SciVerse ScienceDirect

Fusion Engineering and Design

journal homepage: www.elsevier.com/locate/fusengdes

Study of thorium–uranium based molten salt blanket in a fusion–fission hybrid reactor

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ARTICLE INFO

Article history:
Available online 5 April 2012

Keywords:
Fusion–fission hybrid reactor
Molten salt
Blanket
Thorium

ABSTRACT

Not only solid fuels, but also liquid fuels can be used for the fusion–fission symbiotic reactor blanket. The operational record of the molten salt reactor with F–Li–Be was very successful, so the F–Li–Be blanket was chosen for research. The molten salt has several features which are suited for the fusion–fission applications.

The fuel material uranium and thorium were dissolved in the F–Li–Be molten salt. A combined program, COUPLE, was used for neutronics analysis of the molten salt blanket. Several cases have been calculated and compared. Not only the influence of the different fuels have been studied, but also the thickness of the molten salt, and the concentration of the ^{6}Li in the molten salt.

Preliminary studies indicate that when thorium–uranium–plutonium fuels were added into a F–Li–Be molten salt blanket and with a component of 71% LiF–2% BeF_2–13.5% ThF_4–8.5% UF_4–5% PuF_3, and also with the molten salt thickness of 40 cm and natural concentration of ^{6}Li, the appropriate blanket energy multiplication factor and TBR can be obtained.

The result shows that thorium–uranium molten salt can be used in the blanket of a fusion–fission symbiotic reactor. The research on the molten salt blanket must be valuable for the design of fusion–fission symbiotic reactor.

1. Introduction

The ITER project (International Thermonuclear Experimental Reactor) was first initiated by the former Soviet Union, the USA, the EU and Japan in 1985, and was joined by China and Korea in 2003, and by India in 2005 [1]. Although the aim of an entirely commercial energy from fusion may not be easy to achieve in a recent time, a fusion–fission hybrid reactor (FFHR) must be a practicable way to capture the fusion energy.

In a FFHR, the fissionable materials, breeding materials, coolant and other structural materials were fixed outside the reactor core to produce enough tritium to maintain the fusion reaction, to breed the fissionable materials, and to release the electric power at the same time. FFHR has a better ability of fuel breeding than the fast reactor. FFHR is a subcritical system that it has an inherent safety. Because the blanket has a higher energy multiplication factor, the FFHR system has lower requirements on the fusion technology, that means, the fusion conditions for a FFHR were more easily to achieve compared with a fusion reactor. Study of the FFHR can promote the development of the fusion technology.

Generally, in a fusion–fission hybrid reactor, the fusion power may be 200–300 MW for the request of a driven source, and depleted uranium or natural uranium is considered to be used in a fission blanket in the hybrid reactor. A solid blanket has been designed and analyzed in the previous work [2]. As we know, both the solid and liquid blanket can be used in the hybrid reactor. The liquid is easy to be shaped so the molten salt fuel may be more suitable for the fission blanket which has a complex structure. The molten salt reactor also has several advantages as follows. Fission reaction heat is generated within the molten salt and transferred by the liquid, so the heat removal process is simplified. Because of its flowing, the fission products can be removed with the online post-processing of the spent fuels, and new fuels can also be added online without a shutdown refueling. The molten salt reactor also has an inherent safety because it can be solidified in the freezing tank in an emergent situation.

Thorium and uranium mixed fuels in the fusion–fission symbiotic reactor has been studied. F–Li–Be was chosen as the molten salt, and the uranium and thorium mixed fuels were dissolved in the F–Li–Be molten salt. Several kinds of molten fuels have been considered and compared. There are different molar ratio of thorium in each of the molten fuels. This study presents the neutron physics analysis of fusion–fission hybrid reactor, using a molten salt of F–Li–Be, with ThF_4, UF_4 and PuF_3 dissolved in it.

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doi:10.1016/j.fusengdes.2012.03.020

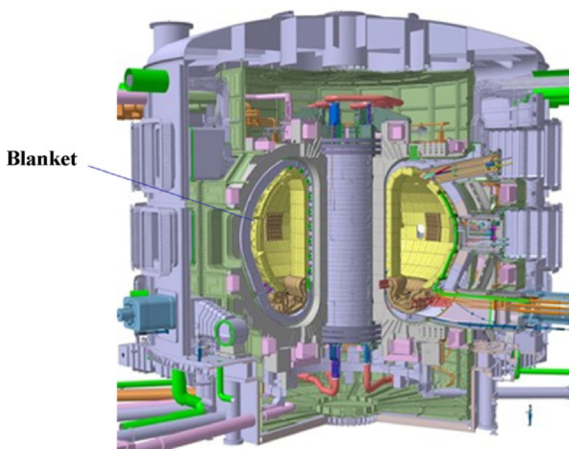

Fig. 1. Design model of ITER and the blanket.

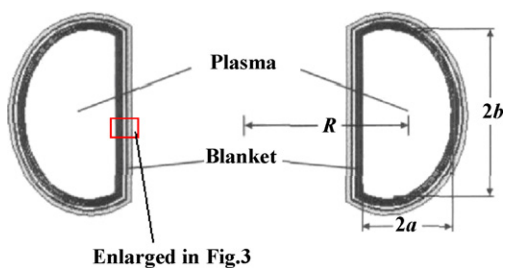

Fig. 2. The cross-section of the blanket D-model.

2. Blanket model design

A general D-model for a fusion–fission hybrid reactor in ITER was chosen for study. The D-model means the whole blanket is a circular ring, with the shape of the cross-section like a character "D", not a circle. The ITER model is shown in Fig. 1 and the blanket is marked in the figure. The cross-section of D-model is shown in Fig. 2. Sizes of the D-model and other parameters of the hybrid reactor are given in Table 1.

The shape of the whole structure is like a tyre, homogenized in the whole loop. The material of D-model is assumed to be same in the circumferential direction. The inner of the blanket can be described by a one-dimensional model, as shown in Fig. 3, which is also a more detailed description of the marked region in Fig. 2.

As shown in Fig. 3, the blanket is constituted of several main zones: the first wall, moderator, molten salt, reflector and the shield. The liquid fuels were dissolved in the F–Li–Be molten salts in the blanket. The molar fractions of the molten salt components are 71% LiF–2% BeF_2, and 27% fluoride fuel. Two different components of the liquid fuels have been considered, 13.5% ThF_4–13.5% UF_4, and 13.5% ThF_4–8.5% UF_4–5% PuF_3. Different molten salt thicknesses of 20 cm and 40 cm were also studied. The materials and the thicknesses of each zone are shown in Table 2.

Table 1
FFHR parameters.

Greater radius of plasma (*R*), cm	510
Smaller radius (*a*), cm	154.5
Branch point (*b*), cm	286
Fusion power, MW	200
Blanket thermal power, MW	3000

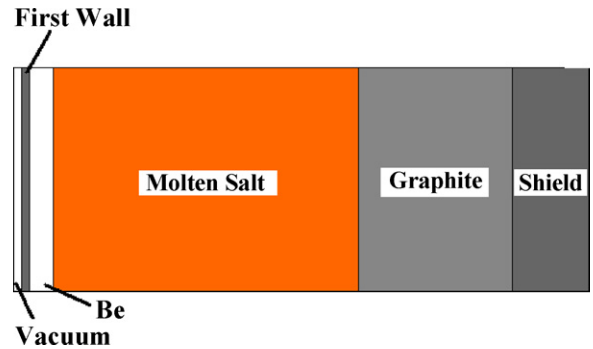

Fig. 3. One-dimensional model of the blanket.

3. Calculation methods

A combined program, named COUPLE [3], was used for neutronics analysis of the molten salt blanket, in which the neutron and photon transport calculations were performed by the Monte Carlo transport code MCNP, and the burnup calculations were performed using ORIGEN2.

4. Numerical results

4.1. Tritium breeding ratio

One of the most important prerequisite for the DT fusion is that a DT fusion or hybrid power plant must produce its own tritium to make itself continue working. Tritium can be produced by the breeding reactions of ^{6}Li and ^{7}Li isotopes in the blanket. Tritium breeding ratio (TBR) per neutron is one of the main parameters in a DT driven fusion–fission hybrid reactor. It should be greater than 1 to satisfy the tritium self-sufficiency of the DT driver of the fusion reactor.

The main tritium breeding reaction is $^6\mathrm{Li}(n, \alpha)T$ (the thermal neutron reaction with ^{6}Li), which is an exothermic reaction and has a large cross section. The fast neutron reaction with ^{7}Li, $^7\mathrm{Li}(n, n\alpha)T$, is an endothermic reaction. The neutron reactions with lithium gain energy in the blanket, and at the same time, produce enough tritium to maintain the fusion reaction. TBR can be given as follows [4]:

$$\mathrm{TBR} = T_6 + T_7 \tag{1}$$

where $T_6 = \int\int \phi \Sigma_{(n,\alpha)T} dEdV$ on ^{6}Li, that is the nuclear reaction rate of the $^6\mathrm{Li}(n, \alpha)T$ reaction. It also means how many tritium nucleus can be produced per fusion neutron in the reaction of neutron with ^{6}Li. And $T_7 = \int\int \phi \Sigma_{(n,n'\alpha)T} dEdV$ on ^{7}Li, is the nuclear reaction rate of the $^7\mathrm{Li}(n, n\alpha)T$ reaction, and as the same, it means how many tritium nucleus can be produced per fusion neutron in the reaction of neutron with ^{7}Li.

The enrichment ratio of ^{6}Li has a great influence on the tritium production rate. The natural lithium has an isotopic composition of 7.5% ^{6}Li and 92.5% ^{7}Li. The concentration of ^{6}Li will increase the TBR, but on the other hand, fewer neutrons will be involved in

Table 2
Parameters of the blanket.

Zone	Material	Thickness/cm
Vacuum	Vacuum	1
First wall	Ferritic steel	1
Moderator	Be	3
Molten salt	LiF–BeF_2–ThF_4–UF_4	20/40
	LiF–BeF_2–ThF_4–UF_4–PuF_3	
Reflector	Graphite	20
Shield	Ferritic steel	10