Study on Fission Blanket Fuel Cycling of a Fusion-Fission Hybrid Energy Generation System

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1. Introduction

Currently all nuclear power is generated by fission reactors [1]. However, most of existing nuclear power plants employ thermal neutron reactors and can only use less than 1% fission fuel because ^{238}U cannot be effectively used in these reactors. In fact, if we employ only thermal fission reactors in which the main part of nuclear fuel is ^{235}U, the available resources on the earth will be exhausted soon. To utilize the precious uranium resource efficiently, the fusion-fission hybrid reactor (FFHR) would be appropriate.

A FFHR utilizes fusion reaction T(D, n)^4He in the plasma confined in its Tokamak to generate high energy neutrons as the external source to drive nuclear fission in the subcritical blanket surrounding the plasma. The blanket is filled with both fissile (^{235}U, ^{239}Pu) and fertile (^{238}U or ^{232}Th) nuclides with natural enrichment of ^{235}U or equivalent enrichment of ^{239}Pu. When the fusion neutron moves into the blanket, it either generates energy by fission of ^{235}U/^{239}Pu or converts the ^{238}U to ^{239}Pu and ^{241}Pu by absorbing neutrons in the fission blanket. FFHR not only can consume ^{235}U to generate energy, but also can produce more fissile fuel. Because the fusion neutrons' energy is higher than the average value (2MeV) from thermal fission reactors, FFHR can use natural or depleted uranium as fission fuel. The lithium isotopes arranged in the blanket can be used to breed tritium for supplying fuel to burn plasma. Another advantage of FFHR is that its plasma condition is much lower than that in a fusion-only power reactor.

International Thermonuclear Experimental Reactor (ITER)[2] was designed to generate fusion power about 500MW, and could satisfy the requirement of FFHR. This paper studies the neutrons' performance in light water cooled/moderated fission blankets of a FFHR. The code system COUPLE2.0 [3], which couples the codes MCNPX and ORIGEN2, has been developed by Institute of Nuclear and New Energy Technology, Tsinghua University and is used to simulate the depletion of nucleus for FFHR blanket. The COUPLE2.0 code chart is depicted in Fig. 1. The code has been tested against a number of benchmark problems and the results have shown well agreement with the benchmark data. Thus, the code calculation results are reliable in sense of conceptual feasibility study of the blanket of a typical fusion-fission hybrid

* Acknowledgement: this research is supported by Chinese ITER special program with grant number 2010GB111006

reactor with ITER-scale Tokamak as the fusion neutron source.

2. Code Simulation Model

We have omitted many details of the real FFHR and focused on the neutronic performance of the blanket. We have chosen the "D-shape" model for simulating the cross-sectional area of the plasma zone of a typical Tokamak, which is the general practice in the state-of the-art Tokamak study as depicted in Fig.2. The major radius of the plasma R is 510cm, and the minor radius a is 154.5cm and b equals 286cm. the aspect ratio R/a is 3.30 and elongation b/a is 1.85, and the thickness of the fission blanket is about 0.5～0.8m. The blanket's enlarged drawing is shown in Fig. 3. It is divided into two parts: the front fuel zone with the volumetric ratio of coolant water to fission fuel approximately 2.0 and the back tritium breeding zone (divided by the first pink slab from left). The yellow colored slabs indicate the water coolant/moderator. The 1st plate of metal alloy nuclear fuel (blue slabs) is 1cm and the other fuel plates are 2cm in thickness. The tritium breeder layer thickness with Li_4SiO_4 is 3cm. The moderator layer between the two tritium breeder layers is 10cm in thickness. The Fe reflector layer is 15cm in thickness. In all calculations, the total fission power of the hybrid reactor is set to 3000MW.

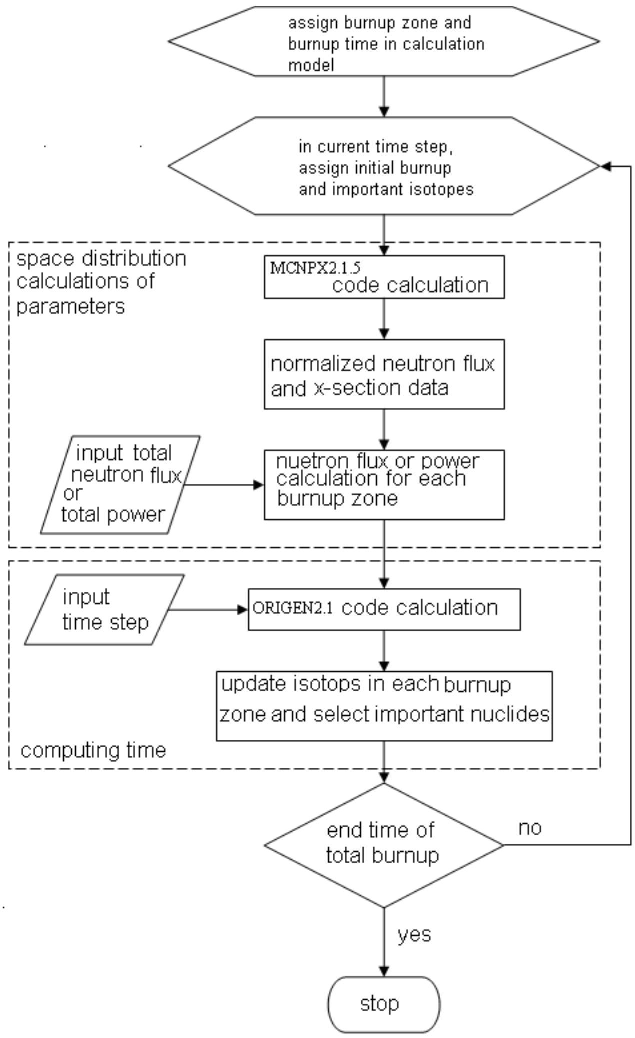

Fig 1: the flow chart of the COUPLE2.0 computer code

The advantage of using light water is its good performances as both coolant and moderator, which may yield the blankets more compact and with better cooling capability.

According to general neutron transport theory, a general neutron transport equation for a steady state problem related to fission blankets of a FFHR can be written as [4],

$$\mathbf{L}\phi = \mathbf{F}\phi + S \tag{1}$$

where S is the external source term and

$$\mathbf{L}\phi = \Omega \cdot \nabla\phi + \Sigma_t\phi - \int_0^\infty dE' \int \Sigma_s(r;E',\Omega' \longrightarrow E,\Omega)\cdot\phi(r,E',\Omega')\,d\Omega' ;$$

$$\mathrm{F}\phi = [\chi(E)/4\pi]\int_0^\infty dE' \int_{\Omega'} \nu\Sigma_f(E')\,\phi(r,E',\Omega')\,d\Omega' .$$

The definition of neutron multiplication factor with external neutron source can be written as

$$k_s = \langle F\phi\rangle / \langle \mathbf{L}\phi\rangle = \langle F\phi\rangle / (\langle F\phi\rangle + \langle S\rangle) \tag{2}$$

which yields,

$$1/k_s - 1 = \langle S\rangle / \langle F\phi\rangle \tag{3}$$

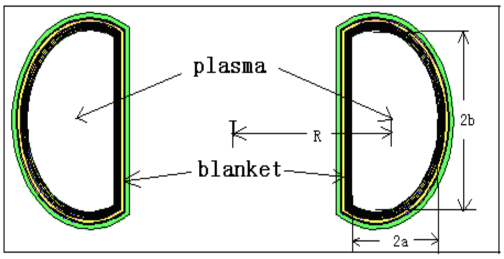

Figure 2: The computational model of FFHR

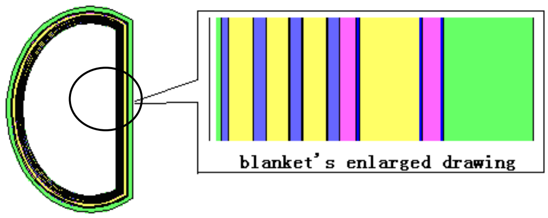

Figure 3 The blanket's enlarged drawing

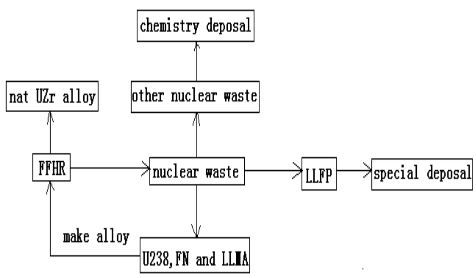

Figure 4 the fuel cycling of FFHR

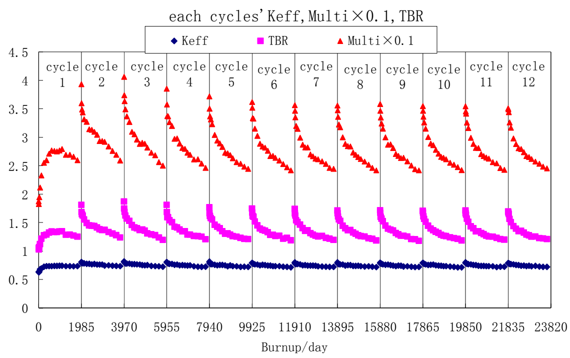

Figure 5 main calculation results

Similarly, the eigenvalue problem without external neutron source can be expressed as,

$$\mathbf{L}\phi = F\phi / k_{eff} \tag{4}$$

From (1) and (4), one obtains,

$$\langle \mathbf{L}\phi, \phi^*\rangle = \langle F\phi, \phi^*\rangle + \langle S, \phi^*\rangle \text{ and } \langle \mathbf{L}\phi, \phi^*\rangle = \langle F\phi, \phi^*\rangle / k_{eff}$$

which will result in,

$$1/k_{eff} - 1 = \langle S, \phi^*\rangle / \langle F\phi, \phi^*\rangle = \langle S\rangle \cdot \langle \phi_S^*\rangle / (\langle F\phi\rangle \cdot \langle \phi_f^*\rangle)$$

$$=(1/k_s-1)\ \langle\phi_S^*\rangle/\langle\phi_f^*\rangle \qquad (5)$$

In turn, one obtains the definition of the efficiency of the external neutron source by

$$\varphi^* \equiv\langle\phi_S^*\rangle/\langle\phi_f^*\rangle=(1/k_{eff}-1)/(1/k_s-1) \qquad (6)$$

Equation (6) indicates that the average value of the external neutron source will be higher than the fission neutron generated in the fission blanket if the k_s is larger than k_{eff}. The equation implies that one external fusion neutron may induce significantly more fissions than one fission neutron generated in the fission blanket, and in turn may amplify the energy gain in the fission blankets per fusion neutron of the proposed fusion-fission hybrid reactor.

3. Results and Discussions

In this study, 1805 days (approximately 5 years) were selected as the fuel cycle length for refueling and 12 cycles were calculated, totally 60 years (the same as the current PWR lifetime, but the FFHR can burn ^{238}U much more effectively and only a little fresh depleted or natural uranium needs to be added). The process of fuel cycling is shown in Figure 4. The fissile nuclide is ^{235}U in initial fuel of natural uranium and ^{239}Pu, ^{241}Pu are generated during operation.

The code system COUPLE2.0 developed at the Institute of Nuclear and New Energy Technology (INET) of Tsinghua University by coupling MCNPX with ORIGEN2 was used to simulate the burn-up process for each fuel cycle in the FFHR blanket. FFHR blanket has been divided into 4 fission fuel depletion zones and each fuel plate was set to be a depletion zone.

To make the calculation more accurate, we employed small depletion calculation time step lengths (approximately from 5 days to 20 days) at the beginning of each fuel cycle. After 4 months, we selected bigger depletion calculation time step length about 120-180 days. Each fuel cycle length is 1805 days (approximately 5 years). The results show that soft neutron spectrum with optimized fuel to moderator ratio can yield an energy amplifying factor of M>20 while maintaining the TBR>1.1 and the CR > 1 (the conversion ratio of fissile materials) in a reasonably long refueling cycle. Using an in-site fuel reprocessing plant, it will be an attractive way to realize the goal of burning ^{238}U with such a new type of fusion-fission hybrid reactor system to generate electric power.

The advantageous of applying light water as coolant consists in the fact that the mature technology employed in current light water reactors, including various passive safety concepts, are of potential to be adapted into the future fusion-fission hybrid nuclear power plant design. It will reduce the risk of R&D related to the functional design of nuclear safety significantly. It is clear that light water system will make the system more compact and will ensure cheaper in investment cost.

4. Conclusions and Remarks

The major results of this study are summarized as:

(1) Using metal or metal-alloy nuclear fuel with natural uranium in fission blanket of a fusion-fission subcritical hybrid reactor with light water as moderator is easily to amplify the fusion energy carried by fusion neutron with a factor of greater than

20.

(2) No additional neutron multiplication material is required if the metal or metal-alloy fuel with natural uranium or even depleted uranium is employed.

(3) Light water is superior as coolant and moderator with respect to investment cost and technology risk [5]. The mature cooling and energy conversion technologies of the-state-of the-art commercial light water nuclear power plant are easily attainable for the hybrid system discussed in this paper, although some R&D efforts to demonstrate its feasibility are still necessary.

(4) Because the deep subcritical in neutron chain reaction and relatively low volumetric power density in the fission blanket of such a fusion-fission hybrid power generation system, the inherent safety of the hybrid system may be achievable.

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