Progress in Conceptual Research on Fusion Fission Hybrid Reactor for Energy

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Abstract. Fusion Fission Hybrid Reactor for Energy (FFHR-E) is fueled by natural uranium and cooled by light water. A three dimensional code MCORGS, which is coupled by MCNP and ORIGENS, has been adopted in neutronics analysis of the three dimensional blanket model. A simplified pyro-reprocessing scheme is suggested. It is expected that the spent fuel can be heated up to 2100K by its decay heat, then the fission product elements whose boiling points below this temperature will be evaporated. The displacement per atom (dpa) of the blanket materials in one year was evaluated. A reload period around ten years is suggested, the spent fuel can be reused multiple times after reprocessing. The average Tritium Breeding Ratio (TBR) is about 1.15 and the blanket energy multiplication is about 12 in the first 60 years. While in the 2nd to 9th 60 years, the average TBR and M are 1.35 and 18 separately.

1. Introduction

Fusion science and technology has made great progress in recent decades, however the development of pure fusion energy is still facing big challenges, such as high fusion gain, tritium self sufficiency, irradiation of the first wall, etc. Fusion Fission Hybrid Reactor (FFHR) is a possible option to accelerate the early application of fusion energy[1,2,3,4,5]. Under the support of the National Magnetic Confinement Fusion Energy Project in China, a conceptual design of a Fusion Fission Hybrid Reactor for Energy (FFHR-E) is proposed[6,7]. The fusion core parameters are similar to ITER (i.e. fusion power ~300 MW and Q~5). An alloy of natural uranium and zirconium (U–10Zr) is adopted in the fission blanket, which is cooled by light water. Nuclear fissile materials are bred and burned in the blanket and the spent fuel is reprocessed every ten years by simplifying the reprocessing process and can be reused in later cycles so as to make fuller use of uranium resources. In this paper, the computational code MCORGS[8] is first introduced and then the progress in the conceptual design of FFHR-E is given.

2. Blanket Neutronics and MCORGS CODE

Fusion plasma can be treated as a fast neutron source in FFHR. Under irradiation by 14.1 MeV neutrons, the composition of the blanket changes with time. Assuming that both the one-group neutron flux density (Φ) and different types of effective transition cross sections remain the same during one burn step (t) at a fixed point (r), the burnup equations for species i (N_i) can be treated as a group of ordinary differential equations[9]:

$$\begin{aligned} dN_i(r,t)/dt = & \sum_{k\neq i} N_k(r,t)\sigma_{eff}^{k\to i}(r)\Phi(r) - N_i(r,t)\sigma_{a,\,eff}^{i}(r)\Phi(r) \\ & + \sum_{j\neq i} f_{j\to i}\lambda_j N_j(r,t) - \lambda_i N_i(r,t) \qquad \text{(i=1,2...)} \end{aligned} \quad . \tag{1}$$

The left side of Eq. (1) is the rate of change of the number density of species i. The first two terms on the right side are the generation and consumption rates by neutron irradiation, and the last two terms are the corresponding rates due to decay. $\sigma_{eff}^{k\to i}$ is the effective transition cross section from species k to $\boldsymbol{i}$. It may correspond to (n,f), (n,γ), $(n,2n)$, or another reaction channel. $\sigma_{a,eff}^{i}$ is the total absorption cross section. $f_{j\to i}$ is the decay branch ratio from species j to i. λ_i is the decay constant of species j. The definition of one-group flux and effective cross section is given below in Eqs. (2)– (4):

$$\Phi(r)=\int\Phi(r,E)dE \quad . \tag{2}$$

$$\sigma_{eff}^{k\to i}(r)=\frac{\int\sigma^{k\to i}(r,E)\Phi(r,E)dE}{\Phi(r)}. \tag{3}$$

$$\sigma_{a,eff}^{i}(r)=\frac{\int\sigma_{a}^{i}(r,E)\Phi(r,E)dE}{\Phi(r)}. \tag{4}$$

The neutron flux $\Phi(r,E)$ is obtained from the continuous energy Monte Carlo code MCNP[10] with point wise nuclear data from ENDF/B-VII.1. Nearly 423 nuclei and nine different types of transition cross sections are considered in the transport calculation. All of the $\Phi(r)$, $\sigma_{eff}^{k\to i}$ and $\sigma_{a,eff}^{i}$ are obtained from MCNP by rigorous problem-specific calculation. These quantities are input as coefficients to Eq. (1), which is solved by the burnup code ORIGEN-S. After one burnup step, the nuclide densities are updated and the procedures above repeated until the end of the problem. MCORGS, a linkage code that couples MCNP and ORIGEN-S, is developed to model reactor burnup problems.

For FFHR, several important parameters, such as the energy multiplication M, the tritium breeding ratio TBR, and the fuel breeding ratio F/B (also called the conversion ratio), are defined in Eqs. (5)– (7) below:

$$M=\frac{\mathit{energy}\ \text{deposited in the blanket by one fusion source}}{\text{energy released by one fusion reaction (17.6Mev)}} \quad . \tag{5}$$

$$TBR=\int(\Sigma_{(n,T)}^{Li^6}+\Sigma_{(n,n')}^{Li^7})\Phi(r,E)drdE \quad . \tag{6}$$

$$\frac{F}{B}=\frac{\mathit{fissile}\ \text{material generate rate}}{\text{fissile material consume rate}}, \tag{7}$$

Where $\Sigma_{(n,T)}^{Li^6}$ and $\Sigma_{(n,n')}^{Li^7}$ are macroscopic tritium generation cross sections of ${}^6Li$ and ${}^7Li$.

MCORGS is validated through several typical benchmarks, such as the fuel rod benchmark from OECD[11], the VVER MOX assembly benchmark [12], and the ADS benchmark[13]. MCORGS has also been used to calculate the ultra-deep burnup hybrid model of Laser Inertial Confinement Fusion Fission Energy (LIFE) [14,15] and the fluid transmuter model of In-Zineraters[16,17]. It shows good agreement with related results.

3. Progress in conceptual design of FFHR-E

On the basis of former neutronics research on one dimensional models, a three dimensional blanket model of FFHR-E has been established. The fuel recycling strategy has also been given.

3.1. Design Guidelines

The design guidelines of FFHR-E are listed below:

(1) Tritium self-sufficiency. The TBR should be greater than 1.05 at the start of the core, and the average TBR in the long run should be greater than 1.15.

(2) Thermal power of blanket maintains 3000MW. Since ITER's fusion power is around 300 to 500 MW, M should be between 6 to10.

(3) F/B greater than 1, more fissile material generated than consumed. F/B >1 can compensates for the buildup of fission products and improves the blanket neutronics performance over the long term. It also simplifies the reprocessing by removing only part of the fission product and not having to separate transuranics from spent fuel.

3.2 The blanket model of FFHR-E

The three dimensional blanket model of FFHR-E is described below. The D-shaped plasma torus of ITER is stretched to a D-shaped semi-cylinder for simplification. The inner borderlines remain the same as in ITER except that the divertor has been simplified. The blanket is constructed around the inner borderlines. The cross section of the blanket perpendicular to the "axes" of the cylinder is shown in Fig. 1. Along the poloidal direction, the blanket is cut into 17 fission fuel modules (1-17) and three divertor modules (D1-D3). The cross section of one fuel module is shown in Fig. 2. Along the radial direction, the blanket is divided into the fission fuel zone, the tritium breeding zone, and the shield zone; the total depth of the blanket is around 1 m, the total natural uranium used in the blanket is 750 tons.

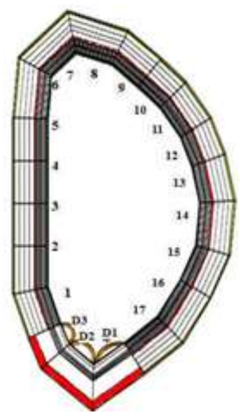

Fig. 1 Poloidal cross section of the blanket

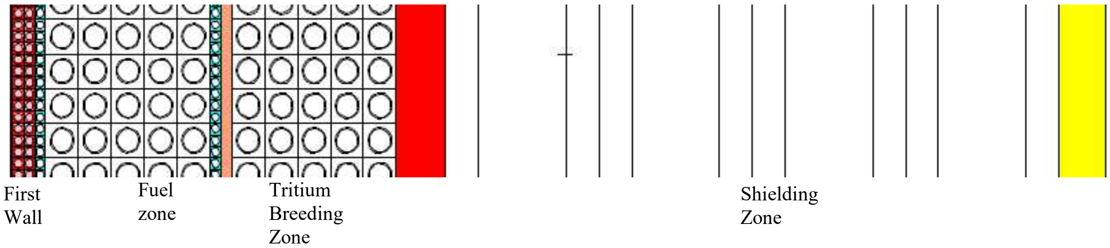

Fig. 2 Radial cross section of the fuel module

The plasma-facing material is a 0.3 cm Be layer. Next to it is a 2 cm thick first wall (FW) with two rows of hollow pipes in it, which are designed to remove heat by pumping water in case of a emergency. The FW is made of HT-9; the pipes' volume ratio (VR) to the HT-9 volume in the FW is 1:1. Outside the FW is the fission fuel zone. There are two layers of 1 cm Zr–2.5Nb to separate each fuel zone layer from its neighboring layer and remove heat in the same way as in the FW. The fission fuel zone is 15 cm thick and divided into five layers. The fission fuel is U–10Zr; its density is 13.6 g/cm3, which is 85% of the theoretical density. The fission product gasses can stay in the 15% gap volume. Light water at 15.5 MPa is used to remove heat from the fission zone. The pipe's inner radius is 0.98 cm, and its thickness is 0.1 cm. The VR of solid parts (fuel plus pipe) to water is 2:1. The water flows into module 1 and out from module 17.

The tritium breeding zone is 15 cm thick. Li_4SiO_4 is the tritium breeder material, and the packing ratio is 0.6. The lithium is enriched to 90% ^{6}Li. Since the (n,T) cross section follows the $1/v$ law, light water is used to moderate the neutrons so as to improve the tritium generation efficiency and reduce the amount of Li_4SiO_4 in the blanket. The VR of lithium to water is 1:1.

The shielding zone is 68 cm thick. Fe and light water are arranged alternately. Light water can moderate the neutrons dramatically so they can be absorbed in Fe. The leakage rate of source neutrons from the blanket is less than 10^{-4}.

The geometry of diver modules are simplified and are covered by a tritium breeding zone. The plasma facing material is a 0.4cm thick W, follows by 7cm Nb-5V-1.25Zr, 10cm hollow zone, 15cm tritium breeding zone, 12cm hollow zone and 20cm Fe.

This model is not trivial. The geometry part of the input file for transportation is about 1300 lines. A base module is first prepared, the geometry and material information of the other 19 modules are generated automatically.

At the beginning of the core (BOC), 400,000 source particles are used, the statistical error of flux, energy deposition and tritium generation are all below 1%. At BOC, the calculated neutronics parameters are as follows: TBR=1.06, M=9.14, F/B=2.27, F-B=0.78. In the long term burnup calculations, 80,000 source particles are used every time step to save computational time, the statistical error is roughly less than 3%.

3.3 Reprocessing strategy

Since hundreds tons of natural uranium are loaded in the blanket, it will improve the economics of FFHR-E by simplifying the reprocessing procedure and decreasing the reprocessing frequency. Two kinds of reprocessing scenarios are proposed as follows.

(1) Simplified pyro-reprocessing. Heat up the spent fuel to a high reprocessing temperature by decay heat, and fission product elements whose boiling points are below the temperature will evaporate. Since FFHR-E uses the U–10Zr alloy, it may be possible to heat the spent fuel by the decay heat and let some fission products evaporate because the melting point of the alloy is lower than traditional UO_2 fuel.

(2) Simplified aqueous-reprocessing or advanced pyro-reprocessing. All the fission products and no transuranic elements are removed.

Scenario 1 will be most frequently used; scenario 2 will be used only when the neutronics performance of FFHR-E degrades and needs to be improved.

3.4 The selection of pyro-reprocessing temperature

Three reprocessing temperatures, 1700 K, 2100 K, and 2500 K, are compared. At 1700 K, the following fission product elements will be removed: Kr, Xe, Br, I, As, Cs, Se, Rb, Cd, Te, Yb, and Sr. At 2100 K, another three elements, Sb, Eu, and Sm will be removed. Among them, Eu and Sm are strong neutron absorbers. At 2500 K, another four elements Tm, Ag, In, and Ga will be removed.

In order to compare the influence of pyro-reprocessing temperature on TBR and M, the spent fuel of the blanket is reprocessed every five years and reused. It can be seen from Figs. 4 and 5 that both M and TBR for 2100 K reprocessing are better than those for 1700 K. However, if the temperature is raised from 2100 K to 2500 K, there is little improvement in TBR and M. It is highly suggested that a reprocessing temperature of 2100 K be adopted.

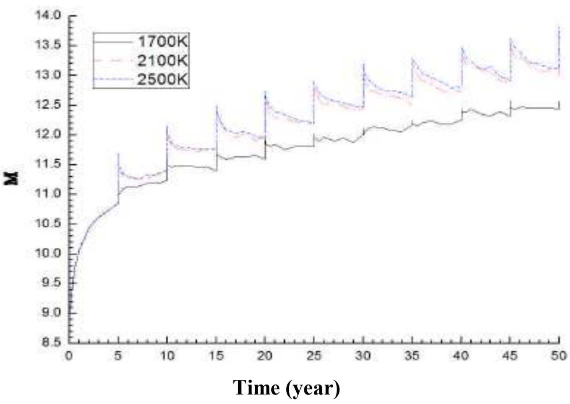

Fig. 3 M vs. time under three pyro-reprocessing temperature

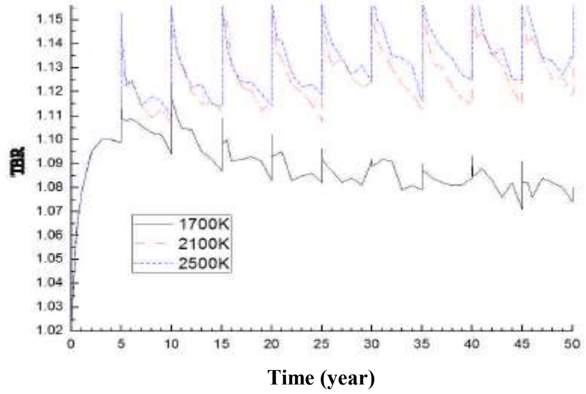

Fig. 4 TBR vs. Time under three pyro-reprocessing temperature

3.5 Neutron irradiation damage to blankets' material

The refueling period is limited by the materials ability to withstand neutron irradiation damage. The displacement per atom (dpa) is a crucial parameter to evaluate this effect. Dpa for materials can be computed by the reaction rate tally (F4). Assuming the neutron spectrum remains unchanged, it can easily get dpa values per year for different materials in the blanket.

In fig 5, we can see the dpa's distribution for plasma facing material (Be), first wall (Fe), emergency pipes before the fuel zone (Zr_front) and after it (Zr_back). In the radial direction, the most severe irradiation appears in the first wall, then in the emergency pipes before the fuel zone, plasma facing material, emergency pipes after the fuel zone respectively. In the poloidal direction, the most severe irradiation appears in module 3 and 15. The dpa's distribution for the cooling pipe in the fuel zone and tritium breeding zone are also calculated but not showed in the figure. The max dpa value for the first wall, the emergency pipes before the fuel zone, the cooling pipe in the first row of fuel are 11dpa/year, 9.94dpa/year, and 8.97dpa/year respectively. It is reported that stainless steel can withstand around up to 50dpa's irradiation, however 200dpa is needed for pure fusion reactors. If first wall material can withstand 100 dpa's irradiation, the refueling period for FFHR-E would be around 10 years.

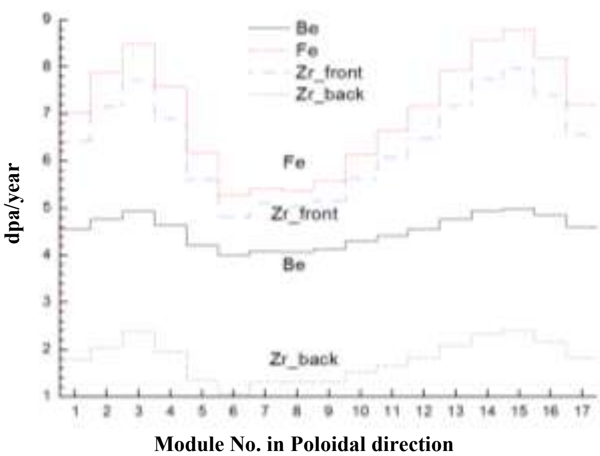

Fig. 5 First wall dpa distribution around poloidal direction

3.6 The combination of two reprocessing strategies

In the following research, simplified pyro-reprocessing is used every 10 years, and simplified aqueous-reprocessing is introduced every 60 years, which is the typical lifetime of a fission reactor. The burnup for 600 years is computed in order to analyze uranium resource utilization potential of FFHR-E. In the first 60 years, the average TBR and M are 1.15 and 12, respectively. From the second to tenth 60-year periods, the average TBR and M are about 1.35 and 18. FFHR-E is deep subcritical system, it is not necessary to start criticality calculation every time step. In order to know the range of keff value, criticality calculation are done at some characteristic time. Keff values at BOC and EOC of the first, the second, the sixth 60-year period are 0.5798,0.6126；0.6421，0.6538；0.7016，0.6898 respectively.

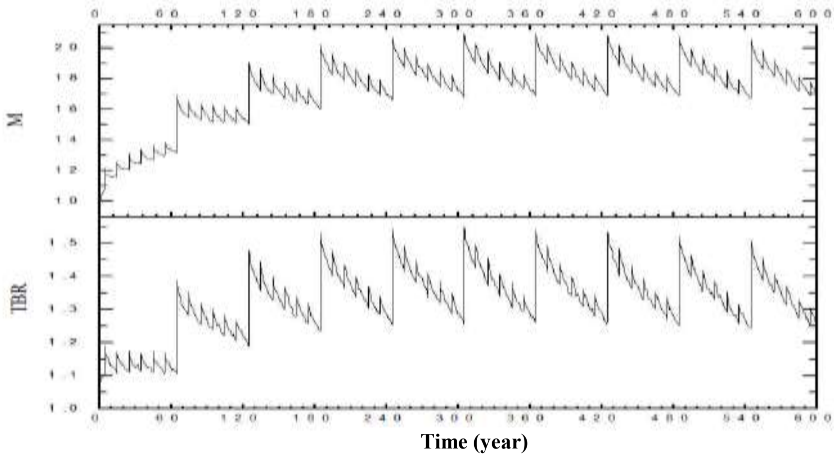

Fig.6 TBR and M vs. time in case of combination of scenarios 1 and 2.

4. Conclusions

Numerical model of MCORGS code and its application in FFHR-E is introduced in this paper. The recent progress in FFHR-E conceptual research is as follows.

(1)The temperature for simplified pyro-reprocessing is suggested to be 2100K；

(2)The refuelling period is around 10 years; the spent fuel can be reused multiple times so as to make fuller use of natural uranium.

5. References

[1] D. L. Jassby, D. H. Berwald, J. Garner.Fast-Fission Tokamak Breeder Reactors[J], Journal of Fusion Energy,1986, 5(3):171-180.

[2] J. D. Lee I and R. W. Moir. Fission-Suppressed Blankets for Fissile Fuel Breeding Fusion Reactors[J], Journal of Fusion Energy, 1981, 1(3):299-303.

[3] Ralph W. Moir. The Fusion Breeder[J].Journal of Fusion Energy.1982,2(4):351～367.

[4] Yican Wu,Progress in fusion-driven hybrid system studies in China[J], Fusion Engineering and Design 2002,63-64: 73-80.

[5] Adem Acir ,Mustafa U beyli, Burning of Reactor Grade Plutonium Mixed with Thorium in a Hybrid Reactor[J], J Fusion Energ (2007) 26:293–298.

[6] Xueming Shi. Conceptual research on blanket neutronics of fusion fision hybrid for energy [D],China Academy of Engineering Physics.2010. (in Chinese)
[7] Xueming Shi, Xianjue Peng. Preliminary conceptual design on blanket neutronics of a fusion fision hybrid for enengy[J] , Nuclear power and engineering (in Chinese),2010,31(4):5-8
[8] Xueming Shi, Ben'ai Zhang.Development of Transport-Burnup Code MCORGS[J], Nuclear power and engineering (in Chinese),2010,31(3):1-4
[9]Herman O W, Westfall R M. ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms[R]. Tennessee, Oak Ridge National Laboratory, 1998.
[10]Judith F. MCNP——A General Monte Carlo N-Particle Transport Code[R]. Los Alamos National Laboratory, 1997.
[11] M.D. Dehart. OECD/NEA burnup credit calculation criticality benchmark phase I-B Results[R]. US: Oak Ridge National Laboratory, 1996.
[12] Kalugin M,Shkarovsky D,Gehin J. A VVER-1000 LEU and MOX Assembly Computational Benchmarks[R]. NEA/NSC/DOC,2002.
[13] Slessarev I., Istiako A.. IAEA ADS Benchmark Results and Analysis. IAEA ADS Benchmark [C]. Madrid: TCM.1999:451-482.
[14] Edward I Moses and T Diaz do la Rubia.Powering the future with LIFE[R], California ,Lawrence Livermore National Laboratory ,2009.
[15] Junyun Yang, Xueming Shi, Yangjun Ying. Numerical simulation on blanket neutronics of laser inertial confinement fusion fission energy [J] . Atomic Energy Science and Engineering(in Chinese),2015,49(11):1961-1965
[16] Benjamin B Cipiti, Virginia D Cleary, Jason T Cook,et al.Fusion Transmutation of Waste: Design and Analysis of the In-Zinerator Concept[R]. California ,Sandia National Laboratory, 2006.
[17] Xueming Shi), Junyun Yang, Cheng'an, Liu.Transport and Burnup Numerical Simulation on theLiquid Blanket Burnup of In-Zinerater[J]. Nuclear Physics Review(in Chinese),2014,31(2):119-123.