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The neutronics studies of a fusion fission hybrid reactor using pressure tube blankets

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ABSTRACT

In this paper, a fusion fission hybrid reactor used for energy producing is proposed based on the situation of nuclear power in China. The pressurized light water is applied as the coolant. The fuel assemblies are loaded in the pressure tubes with a modular type structure. The neutronics analysis is performed to get the suitable design and prove the feasibility. The energy multiplication and tritium self-sustaining are evaluated. The neutron load is also cared. From different candidates, the PWR spent fuel is selected as the feed fuel. The results show that the hybrid reactor can meet the expected reactor core lifetime of 5 years with 1000 MWe power output. Two ways are discussed including burning the discharged PWR spent fuel and burning the reprocessed plutonium. The energy multiplication is big enough and the tritium can be self-sustaining for both of the two ways. The neutron wall load in the operating time is kept smaller than the one of ITER. The way to use the reprocessed plutonium brings low neutron wall load, but also brings additional difficulties in operating the hybrid reactor. The way to use the discharged spent fuel is proposed to be a better choice currently.

1. Introduction

The fusion fission hybrid reactor has been proposed for decades. The reactor employs the fission technology to multiply the fusion neutrons for various applications. Such an approach will significantly relax the restriction of fusion reactor. For developing a fusion fission hybrid reactor, one of the keys is to design suitable blankets for neutron multiplication. Recently, the R&D of the fission blankets has progressed a lot in US, Japan, China and Turkey. Most of these studies applied the ITER-like (some were even smaller) fusion source rather than the real fusion reactor. The neutron multiplied in the blanket was used to for energy producing, nuclear fuel breeding and nuclear waste transmutation.

In US, Stacey proposed a kind of fast blanket for burning transuranic in the hybrid reactor, in which a small D–T tokamak fusion neutron source is based on and the $Li_{17}Pb_{83}$ was used as the coolant. It demonstrated that the hybrid reactor was realizable based on the easily achieved fusion technology [1–3]. In Japan, Murata proposed a kind of thermal blanket based on the ITER fusion source. The light water was applied as the coolant. The reprocessed plutonium in the matrix of depleted uranium is used as the fuel. It is indicated that such a selection is suitable for energy producing [4]. Also the Th–U fuel cycle was evaluated in the further studies [5,6]. Other progresses have also been made in the theoretical researches on the fuel breeding and transmutation [7–10].

In China, the hybrid reactor has been widely recognized. A series of conceptual design have been proposed based on the EAST and ITER fusion source [11,12]. As a potential solution of the energy access in the future, the R&D of hybrid power reactor now attracts more attention and has been strongly pushed by the Chinese government. A project begun in 2010 is involved in the National Magnetic Confinement Fusion Science Program [13]. It aimed the R&D of hybrid reactor for energy producing. The main work is to propose a suitable blanket design for energy multiplication. The studies are based on the ITER-like fusion source. The light water is selected as the coolant using the current NPP technologies in China. A lot work on evaluation and design has begun [14,15]. The fuel selection and structural design are the current keys.

In this paper, we propose a modular type blankets using the pressure tubes to contain the pressurized light water. The fuel zone in the blanket is designed similar to the one of CANDU reactor. It makes the highly pressurized light water accessible in the square zone of blanket with only sheet structural support. The modular blanket is designed to fit the feed water system. The neutronics analysis is performed in this paper based on the pressure tube blanket. The energy multiplication factor (M) is one of the key parameters in the evaluation, together with the tritium breeding ratio (TBR) and the neutron wall load of the first wall (FW).

Based on the analysis, the modular type blanket design is proposed. The spent fuel discharged from PWRs is used as the

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Table 1
The main parameters of ITER.

Parameters	Value
Major radius/m	6.2
Minor radius/m	2.0
Plasma elongation	1.85
Fusion power/MW	500
FW load/MW/m^2	0.57

candidate fuel. The 15.5 MPa light water is used as the coolant, while the 4.8 MPa helium is used to cool other systems in the blanket. They are independently arranged in the modular type blankets. The Li_2O with 90% enriched ^6Li is used as the breeding material. The small moderator-to-fuel ratio is adopted in the fuel zone. The self-sustaining of tritium is ensured in the reactor core lifetime of 5 years. The reprocessed plutonium is also considered in the evaluation to increase the energy multiplication and decrease the neutron wall load (FW load).

In Section 2, the main parameters of fusion source and blanket are proposed. Then, the model based on the design is described in Section 3, together with the computational methods for evaluation. Section 4 presents the neutronics analysis for determining the blanket design and demonstrating the feasibility. Finally, Section 5 gives the conclusions to close the paper.

2. Fusion source and blanket configuration

2.1. Parameters of the fusion source

The main work in this study focuses on the fission blankets. The plasma physics itself and engineering technology of fusion reactor design are referred from the previous studies. The ITER fusion source is the most usual one in the world wide researches. In this paper, it is referred. Table 1 gives the parameters [16]. Based on its 500 MW power output, the energy multiplication factor should be not less than 6 to obtain the 1000 MWe (suppose the thermoelectric conversion is 1/3, i.e. 3000 MWt is needed).

However, due to the reactivity swing in the burn-up history, the required intensity of fusion source should be adjusted. The FW load will change correspondingly. In this paper, we aim to realize big energy multiplication in the reactor core lifetime. The FW load will be correspondingly small. It is expected to be smaller than the one of ITER.

2.2. Pressure tubes water cooling blanket

The blankets are designed based on the configuration of ITER fusion source. Two types of blankets are involved in the hybrid reactor, i.e. the inboard blanket and the outboard blanket (fission blanket). Around the ring of plasma, 20 pairs of inboard and outboard blankets are arranged as sketched in Fig. 1.

Three zones are divided in the inboard blanket. Forward the toroidal field coil, three zones are the FW using low activation ferritic–martensitic steel (RAFM) [17,18], tritium breeding zone (20 cm) and shielding layer (20 cm) using SS316, respectively. No fuels are loaded in these blankets to protect the toroidal field coil from the damage of neutron flux and gamma ray.

The energy multiplication is realized in the outboard blankets surrounding the plasma. Forward the plasma, five zones are designed, including the FW using RAFM, fuel zone, tritium breeding zone, graphite reflector and shielding layer using SS316.

To increase the percentage of coverage, the outboard blanket is designed with the trapezoidal cross section as in Fig. 2. Previous press analysis indicates [19] that in such a fuel zone, the FW using

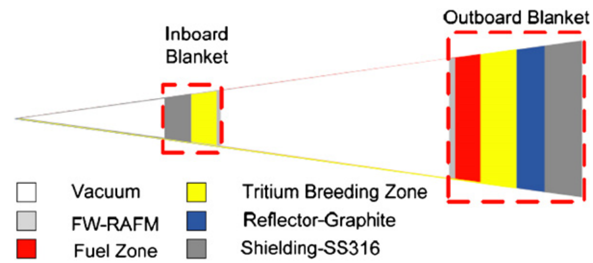

Fig. 1. The sketch of cross section of the hybrid reactor.

RAFM should be at least several centimeters thick to contain the highly pressurized coolant. But the neutronics analysis as in Section 3 shows that the thick structural wall is severely bad for the tritium self-sustaining. Another choice is to avoid the press acting on the straight structural wall. So the fuel zone using pressure tubes is proposed.

As in Fig. 2, the red tubes are the pressure tubes, which are loaded crossing the radial axle line. Two rows of tubes are loaded. Fig. 3 illustrate the inside configuration of the tube. The fuel pins are loaded along the tube and arranged ring by ring, similar to the one in CANDU reactor. The RAFM is used as the structural material to resist the neutron damage.

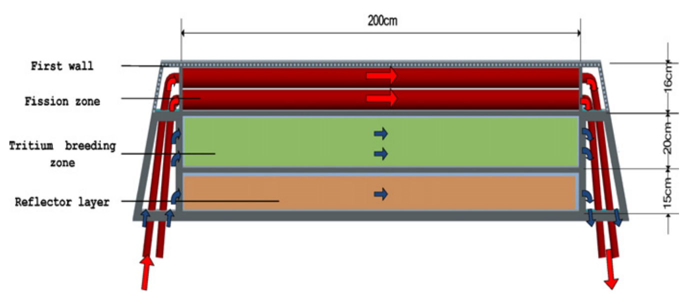

Fig. 2. The cross-section of outboard blanket.

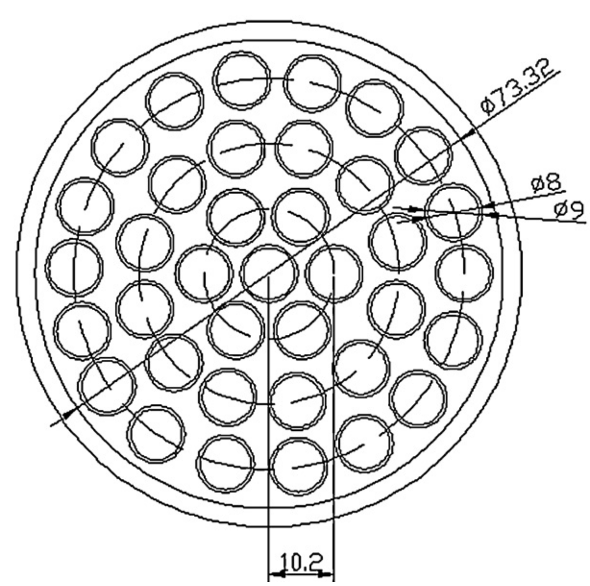

Fig. 3. The inside configuration of pressure tube.

Fig. 4. Simplified model of the hybrid reactor.

3. Modeling and computational methods

3.1. Simplified model for neutronics analysis

To investigate the neutronics properties, the simplified D-type model is proposed considering four zones: the inside void, inboard blankets, neutron source and outboard blankets. Fig. 4 illustrates the simplified model from front and top view.

Inside the fuel zone, the heterogeneity of fuel lattice is considered simply. A layered model keeping the mass of fuel, coolant and other materials equivalent is proposed as in Fig. 5.

3.2. Computational methods and codes

In the evaluation, three key parameters are calculated, i.e. the energy multiplication factor (M) and the tritium breeding ratio (TBR).

Eq. (1) gives the definition of M as:

$$M \doteq \frac{\kappa}{E_s \times \nu} \frac{K_s}{1-K_s} \tag{1}$$

where κ and ν are the energy and number of neutrons released from once fission, respectively. E_s is the energy of a fusion neutron. It is 14.1 MeV. K_s is the neutron multiplication factor of sub-critical system. The energy gotten from the tritium breeding is not considered here.

The TBR is defines as the tritium gain from once fusion as in Eq. (2). It benefits from both the breeding in the inboard and outboard blankets. It is evaluated as:

$$\mathrm{TBR} = \frac{\langle \phi \cdot \Sigma_{{}^6\mathrm{Li}(n,\alpha)} \rangle + \langle \phi \cdot \Sigma_{{}^7\mathrm{Li}(n,n'\alpha)} \rangle}{S} \tag{2}$$

where ϕ is the neutron flux, Σ_6Li(n,α) and Σ^7Li(n,n'α) are the macro cross section of ^{6}Li + T and ^{7}Li + T reactions, respectively. S is the intensity of fusion neutron.

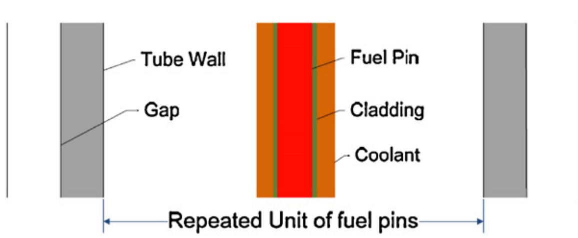

Fig. 5. Layered fuel zone model.

For a given power output, different M requires different intensity of fusion source. The FW load is therefore different. The FW load is evaluated as:

$$\text{FW load} = \frac{E_s S}{A} \tag{3}$$

where A is the surface area of FW, S is the intensity of fusion source according to the M.

In the evaluation, a Monte-Carlo code considering burn-up calculation is applied. Fig. 6 gives the computational flow. The source mode is adopted to get K_s.

4. Neutronics analysis

Aiming to find the suitable fuel and configuration, different factors effecting on the neutronics are analyzed. Mostly, only the

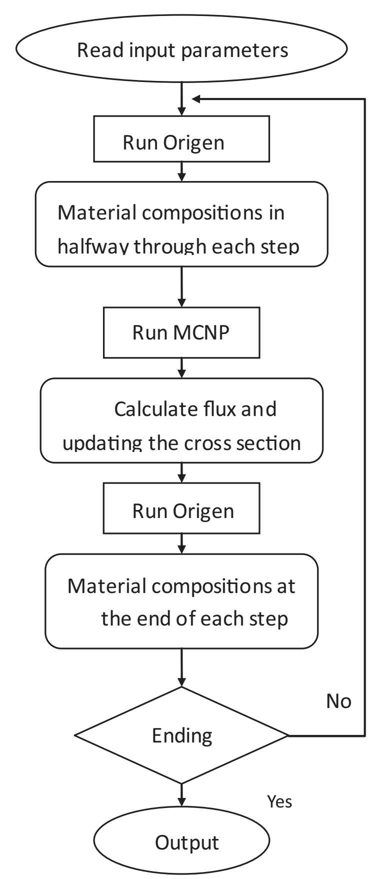

Fig. 6. Computational flow of the evaluation code.

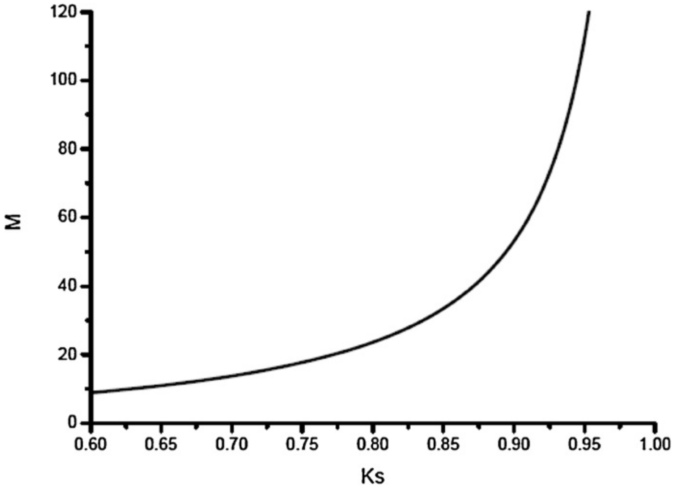

Fig. 7. The change of energy multiplication factor with K_s.

condition at the beginning of reactor core lifetime (BOL) is considered in discussing the sensitivity. The burn-up history will be considered finally to demonstrate the feasibility.

4.1. Energy multiplication and its requirements

The energy multiplication is the key important parameter for a hybrid power reactor. High M is grateful in the fission blankets, which makes the plasma parameters easy to achieve. Due to the reactivity swing in the burn-up history of fissionable fuels, the M changes with the varying neutron multiplication factor, i.e. K_s. For the 1000 MWe (3000 MWt) power output, the M should be ensured to be greater than 6 in the whole burn-up history. The changing M requires the adjustment of fusion source. It is very difficult to operate, so the reactivity swing is required to be as small as possible. Fig. 7 indicates the requirements of K_s for different M. For the required M, the K_s should be around 0.6. The greater K_s like 0.8 will bring much larger multiplication. But with the growth of K_s, the M becomes more and more sensitive to the varying K_s. It is not suitable, because even very small swing will need large adjustment of the fusion source. Therefore, the fuels whose K_s varies around 0.6–0.8 will be suitable.

Three kinds of fuel are tested, considering the usual moderator-to-fuel ratio in current PWRs. The change of K_s in the lifetime is illustrated in Fig. 8. The enriched uranium is the one usually used in current PWRs. The enrichment is 4.45%. The spent fuel is discharged form PWRs after 33 GWd/tU burn-up, partitioned the fission products by high temperature reprocessing. The heavy isotopes are not partitioned. The component is given in Table 2.

From the viewpoint of multiplication, the enriched fuel suffers larger reactivity swing. Its large K_s at BOL is not what we want, either. The spent fuel and natural uranium fuel are both the suitable candidates. But considering the situation of China, a great deal of natural uranium should be reserved for the large number of PWRs in the future.

4.2. Tritium breeding and its requirements

Another important parameter is the TBR. The tritium must be self-sustaining in the hybrid reactor. Considering the loss in operating, the TBR is required to be larger than 1.05 (5% loss). It will be contributed from the breeding in both of inboard and outboard blankets.

Table 2
The component of isotopes in the spent fuel.

Isotopes	Mass percent/%
^{235}U	8.87E−01
^{236}U	3.91E−01
^{238}U	9.75E+01
^{239}Pu	6.39E−01
^{240}Pu	2.44E−01
^{241}Pu	1.55E−01
^{242}Pu	5.20E−02
^{237}Np	4.35E−02
^{241}Am	3.98E−03
^{242}Am	7.00E−05
^{243}Am	9.97E−03
^{242}Cm	1.41E−03
^{244}Cm	3.08E−03

The solid tritium breeding material is used. Both the Li_2O and $LiSiO_4$ are tested. Fig. 9 gives the TBR. Different enrichment of ^6Li is considered. Obviously, the higher enrichment of ^6Li conduce the larger TBR. The Li_2O is better than $LiSiO_4$ for its higher content of Li in unit volume. The Li_2O with 90% enriched ^6Li is loaded in the inboard and outboard blankets. The helium is applied to collect the tritium and cooled the breeding zones.

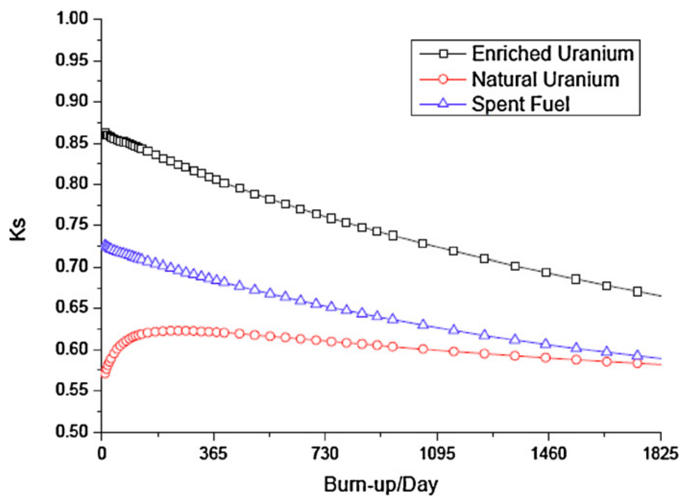

Fig. 8. The K_s varying of different fuels in the lifetime.

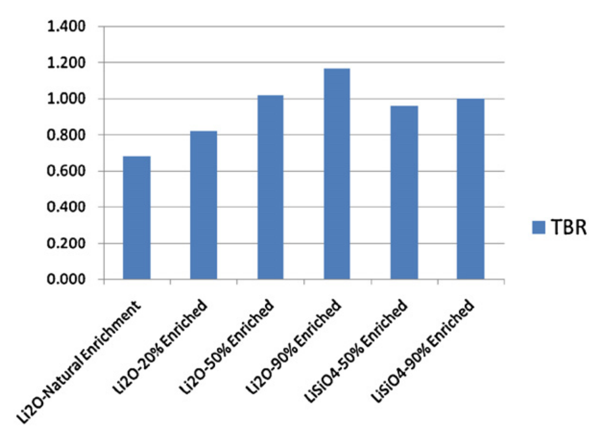

Fig. 9. The change of TBR with different material.

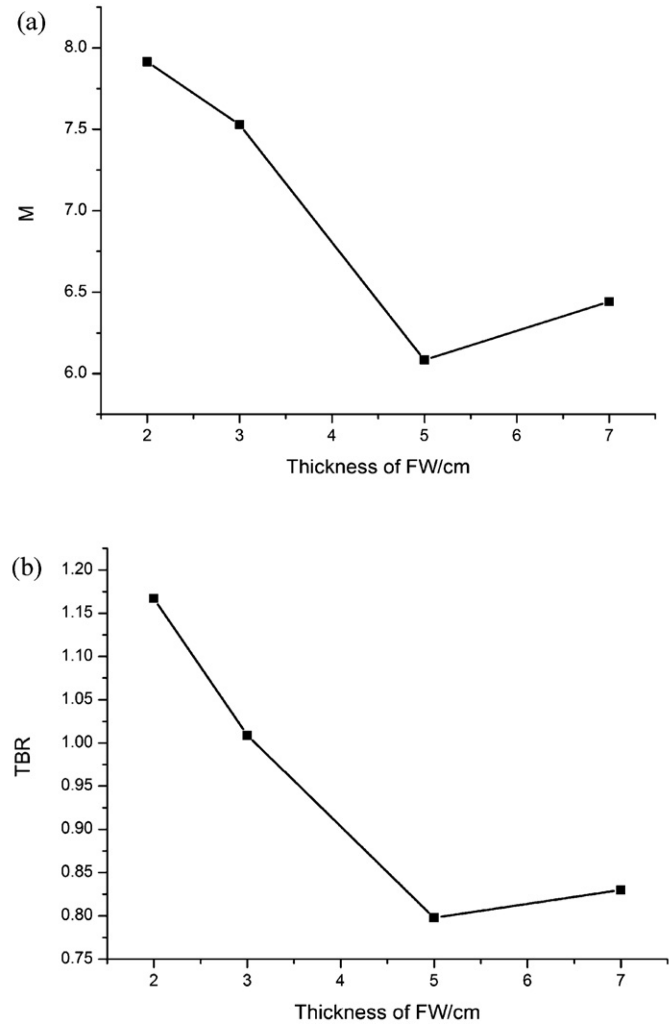

Fig. 10. The effect of the thickness of FW.

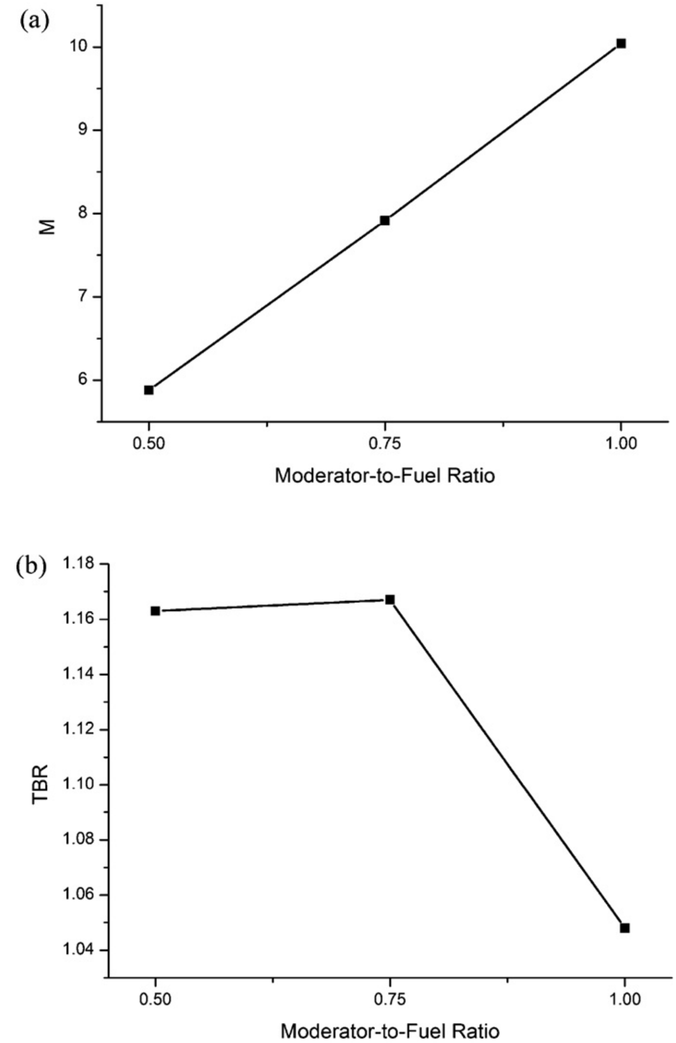

Fig. 11. The effect of the moderator-to-fuel ratio.

4.3. Effect of the thickness of FW

The fission in the outboard blanket is driven by the fusion neutrons. The RAFM is used as the candidate of FW material. For the neutron moderation and absorption of ferrum (Fe), the thickness of structural wall will affect the neutronics performance in the blanket. Fig. 10 gives the variation of *M* and TBR with the change of thickness.

Generally, the thinner the wall is, the better the neutron performs. But with the increase of the thickness, the regularity will change as seen in Fig. 10. It is due to thicker wall brings better moderation and reflection, where it performs like a reflector in traditional thermal reactor. However, such effect will not bring significant improvement of the neutron performance. The structural wall of 2 cm thick is more suitable for the energy multiplication and tritium breeding.

As mentioned before, such a sheet wall cannot contain the highly pressurized coolant securely. This is the reason why we propose a pressure tube water cooled blanket.

4.4. Effect of the moderator-to-fuel ratio

In a water cooled blanket, the moderator-to-fuel ratio affects the neutronics performance significantly. For the thermal neutron spectrum, the big ratio benefits the energy multiplication as illustrate in Fig. 11. But it will be bad for the tritium self-sustaining for the neutron loss in the coolant by absorption. The latter one is more important for the hybrid reactor. Therefore, the ratio is selected to be 0.75. Furthermore, the smaller ratio brings harder neutron spectrum. It is helpful to decrease the reactivity swing in the burn-up history.

The small ratio is realized by the design of pressure tube. The configuration has been given in Fig. 3. Instead of filling the gaps between tubes with water, the helium is applied. Then, the moderator-to-fuel ratio is adjusted to be 0.75.

4.5. Effect of different component in the fuel

The neutronics analysis above shows that the water cooled blanket using the PWR spent fuel for energy multiplication is feasible. However, it is still attractive to improve the performance by increase the neutron multiplication. For a thermal blanket, increasing the fraction of fissile isotopes in the fuel is the most useful way. It can be realized by using the reprocessed plutonium. In this way, the reprocessed plutonium will be mixed with the matrix of depleted uranium.

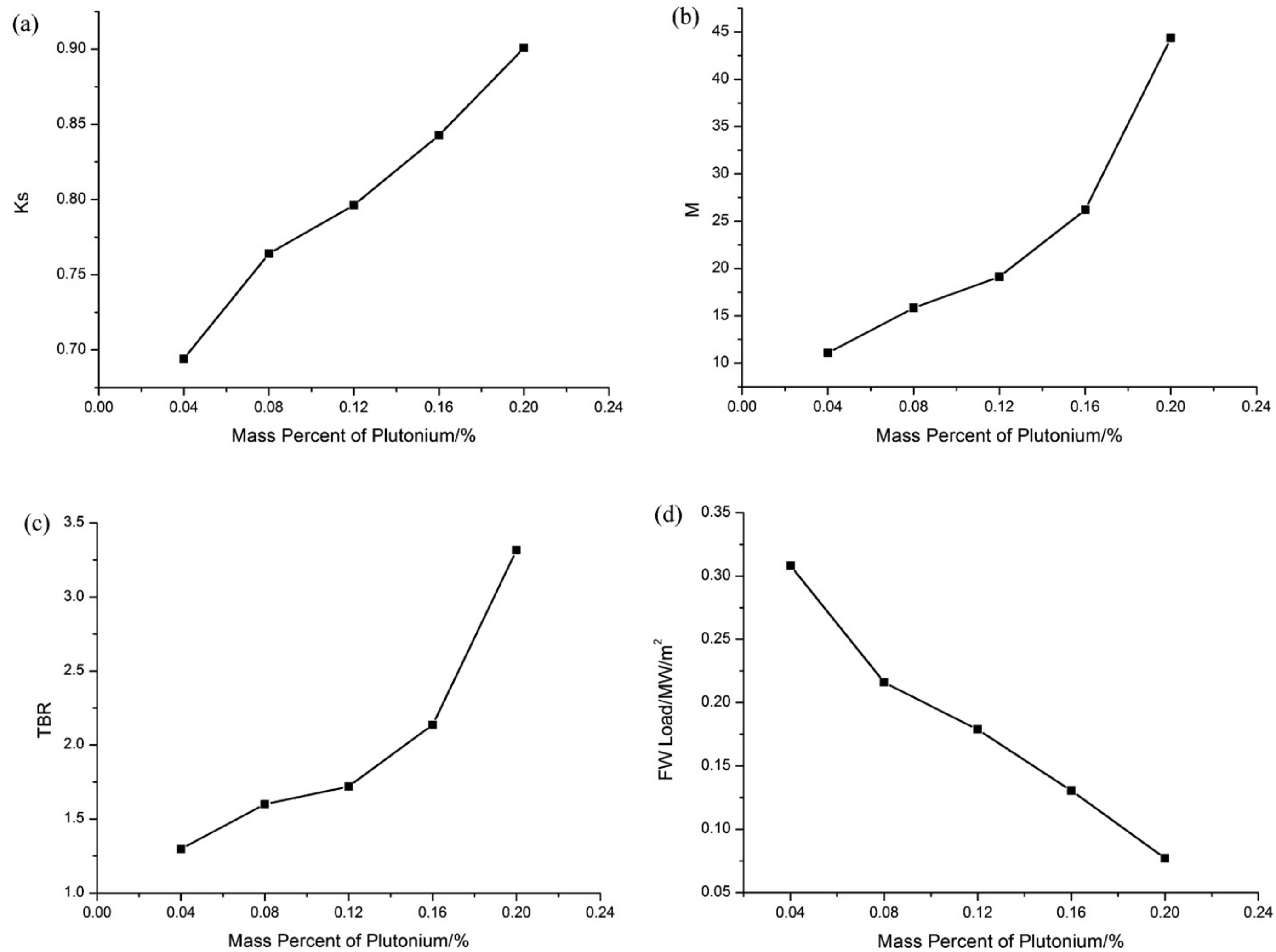

Fig. 12. The effect of burning reprocessed plutonium with different fraction.

Fig. 12 indicates the change of K_s, *M*, TBR and FW load with the increasing fraction of plutonium in the fuel. Both the *M* and TBR are improved with the increasing fraction of plutonium. The FW load decreased correspondingly.

However, for the high fraction, the K_s becomes too large and exceeds the confinement as given in Section 4.1. It will not be suitable for the mass percent to be bigger than 12%. In this study, the fuel with 8% reprocessed plutonium is chose for further analysis.

4.6. Performance in the burn-up history

A preliminary design is proposed based on the neutronics analysis. Table 3 gives the main parameters determined above. The PWR spent fuels are loaded in the pressure tubes. Two ways to apply the spent fuel are both evaluated, i.e. burning the discharged spent fuel and burning the reprocessed plutonium. The expecting reactor core lifetime is 5 years. The changes of *M*, TBR and FW load in the lifetime are shown in Fig. 13.

The results prove that such a blanket design meet the key requirements in the whole lifetime, no matter using the discharged PWR spent fuel or using the reprocessed plutonium. The energy multiplication is big enough to supply 1000 MWe power output based on the ITER fusion source. The tritium can be self-sustaining in the operating time. The FW load in the operating time will not be greater than 0.5 MW/m^2.

From the viewpoint of energy multiplication and tritium self-sustaining, the way to burn the reprocessed plutonium performs better. Also, the FW load will be significantly decreased. However, for a water cooled hybrid reactor, the high fraction of fissile isotopes induces large reactivity swing. It needs wide-range adjustment of the fusion power to supply suitable neutron source. To realize a stable power output, this problem will bring additional difficulties in operating a hybrid reactor.

For burning the discharged spent fuel, the key parameters have already acceptable. Furthermore, from the results in Fig. 13, the burn-up performance keeps almost stable in the whole lifetime. Such stability make it is a more suitable way.

Table 3
The summary of neutronics analysis.

	Determined parameter
Fuel	Discharged spent fuel Reprocessed plutonium (8%)
Tritium breeding material	Li_2O with 90% enriched ^6Li
FW	2 cm thick using RAFM
Moderator-to-fuel ratio	0.75

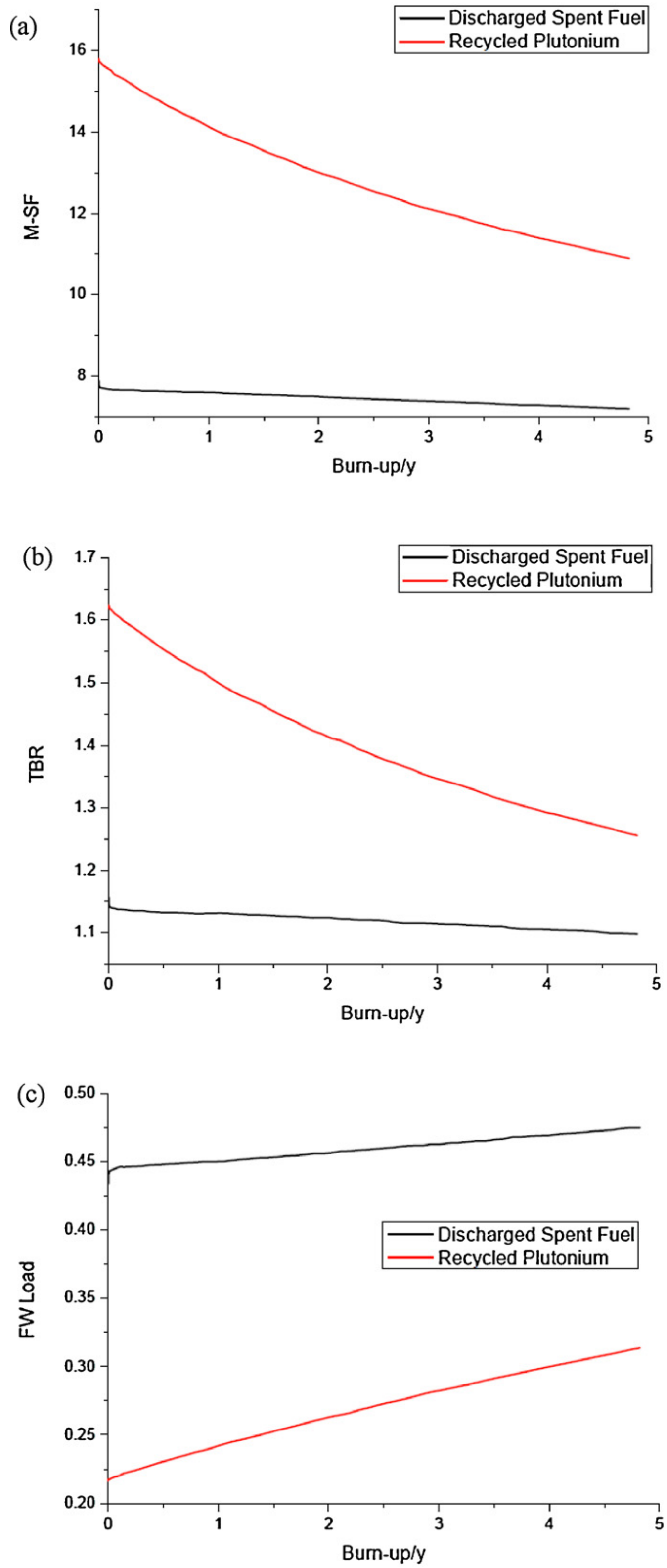

Fig. 13. The variation of key parameters in the burn-up history.

5. Conclusion

In this paper, the hybrid reactor with modular type pressure tube fission blanket is proposed. The pressurized water is selected as the coolant based on the most developed fission technologies in China. It is expected to be of 1000 MWe power output in 5 years. The pressure tube is used to contain the pressurized water and protect the sheet structural wall. The modular type blanket is used due to the arrangement of pressure tubes.

The neutronics effect is analyzed and the performance is evaluated. Theoretically, different fuels can be used in the subcritical reactor. But considering the neutronics properties and the situation in China, the PWR spent fuel is selected as the candidate. The Li_2O with 90% enriched ^6Li and the structural wall of 2 cm thick are taken to improve the tritium breeding. The structural material including the tube wall uses the RAFM to resist the neutron damage. The helium is used to cool the tritium breeding zones. Also it is used to adjust the neutron spectrum in the fuel zones.

The results of numerical evaluation demonstrate that the candidate fuel can satisfy the targets in the main parameters. The energy multiplication is big enough while using the discharged spent fuel as feed fuel for an ITER based hybrid reactor. The tritium can be self-sustaining in the operating time of 5 years. The FW load will be not more than 0.5 MW/m^2. If the reprocessed plutonium can be supplied, this parameter can be further decreased. But the flexible adjustment of fusion power should be realized previously.

This paper gives the fundamental evaluation of designing a water cooled hybrid power reactor. The results demonstrate the feasibility. Further investigation and design will be performed based on the conclusions in the future.

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