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A fusion–fission hybrid reactor with water-cooled pressure tube blanket for energy production

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ABSTRACT

A fusion–fission hybrid reactor is proposed to achieve the energy gain of 3000 MW thermal power with self-sustaining tritium. The hybrid reactor is designed based on the plasma conditions and configurations of ITER, as well as the well-developed pressurized light water cooling technologies. For the sake of safety, the pressure tube bundles are employed to protect the first wall from the high pressure of coolant. The spent nuclear fuel discharged from 33GWD/tU Light Water Reactors (LWRs) and natural uranium oxide are taken as driver fuel for energy multiplication. According to thermo-mechanics calculation results, the first wall of 20 mm is safe. The radiation damage analysis indicates that the first wall has a lifetime of more than five years. Neutronics calculations show that the proposed hybrid reactor has high energy multiplication factor, tritium breeding ratio and power density; the fuel cannot reach the level of plutonium required for a nuclear weapon. Thermal-hydraulic analysis indicates that the temperatures of the fuel zone are well below the limited values and a large safety margin is provided.

1. Introduction

World-wide fusion system studies have shown that a fusion energy system has some attractive advantages. Nuclear fusion has an enormous potential to provide a safe, clean and unlimited energy source. However, it is generally recognized that the commercial pure fusion reactors facing many obstacles will not be run in a short period. To demonstrate the plasma physics feasibility of fusion energy, International Thermonuclear Experimental Reactor (ITER) has been under construction for years. The DT fusion power is 500 MW and the average neutron wall loading is 0.57 MW/m^2 in ITER (Aymar et al., 2002), which are much lower than the levels required in a commercial fusion reactor.

At present, all nuclear power is generated by fission reactors. The fission technologies have been well developed in the past decade. However, the conventional fission reactors have some drawbacks. There are two main technical impediments to a sustainable expansion of fission power. The first is the shortage of the nuclear fuel. The other one is the spent nuclear fuel (SNF) which is produced continuously and difficult to be incinerated.

As an intermediate reactor at the transition stage to pure fusion reactors, the Fusion–Fission Hybrid Reactor (FFHR) consisting of the fusion and fission processes has many advantages. A FFHR uses high energy fusion neutrons to drive nuclear fission in the subcritical blanket surrounding the plasma. Because the nuclear fuel in the blanket can multiply the neutrons and energy, the FFHR can be constructed based on the current or expectable fusion technologies. On other hand, when a fusion neutron moves into the fission blanket, it introduces either fission reactions of fissile isotopes or the conversion from fertile isotopes to the fissile ones. For this reason, FFHR can make use of natural uranium as nuclear fuel. In SNF, there is a significant amount of reactor-grade plutonium, and the total atomic density of the fissile isotopes (fissionable plutonium and ^{235}U) is higher than that in natural uranium. These fissile isotopes can also undergo fission reactions for energy production in FFHR. FFHR makes an efficient way of exploiting the uranium resources.

It was proved that an ITER-type tokamak is feasible to be used as a neutron source for energy production (Murata et al., 2005, 2007; Zhou et al., 2011). The possibility that an ITER-type tokamak neutron source could be used for the transmutation of SNF was also investigated at Georgia Tech (Stacey, 2007, 2009).

Pressurized light water cooling technology has been well developed in the fission industry. Light water has high moderation property, so the light water cooled fission blanket has higher energy multiplication capability. The neutronics feasibility to apply the light water as coolant for an energy production blanket was also proven (Jiang et al., 2010; Murata et al., 2005, 2007; Zhou et al.,

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2011). In these works, the homogeneous layered nuclear fuel was considered for simplification. Because of the large scale of tokamak, huge mass of nuclear fuels was loaded in the blanket (for example, 520 tons (Zhou et al., 2011). This greatly increases the burden to reprocess the waste. Moreover, the metal-alloy nuclear fuel or recovered plutonium from spent nuclear fuel was employed to increase the atomic densities of fissile isotopes. A boiling water cooled concept (Unalan et al., 2003) was investigated for rejuvenation of LWRs spent fuel. The operation pressure was selected as 70 bar.

The motivation of this study is to propose a FFHR for energy production using current fusion technologies and mature fission technologies. The ITER-type tokamak is selected as the external neutron source, and the pressurized light water cooling technology is employed in the fission blanket. However, a problem will arise from the use of pressurized light water. The first wall is not suggested to be directly exported to the coolant; otherwise, the first wall must be thick enough to withstand the high coolant pressure. In this situation, the property of the high-energy fusion neutrons will be greatly degraded when the neutrons get through the thick first wall into the nuclear fuel. For this reason, the concept of pressure tube bundle is employed in this study. The fuel pins are filled in pressure tubes which serve as the pressure boundary. The pressurized water flows along the pressure tubes and the first wall is separated from the pressurized coolant, which makes the first wall much thinner and safer. Moreover, due to the external source, the power distribution in the blanket is very uneven in radial direction. It is easy to adjust the flow rate of coolant in the tubes to make sure the fuel pins are adequately cooled.

The pressure tube blanket is designed to achieve the following objectives in this paper. The first wall must be safe. The blanket provides an output of 3000 MW fission power with the temperatures of the fuel zone below the limited values. In order to exploit the uranium resource efficiently, natural uranium oxide and the spent nuclear fuel discharged from 33GWD/tU LWRs with initial uranium enrichment of 3.1% are used as the nuclear fuel. To avoid the physics parameters from exceeding those of ITER, the energy multiplication factor (M, defined as the ratio of fission power produced by the blanket to the fusion power) must be greater than 6 to keep the fusion power lower than 500 MW. Tritium breeding ratio (TBR) is expected to be greater than 1.05 to obtain tritium self-sufficiency. The refueling operation period is set to five years.

The conceptual design of the blanket is described in Section 2. The calculation methods and results are given in Section 3. In the final section, some conclusions are given.

2. Design of the blanket

2.1. Fusion neutron source

In this paper, an ITER-type tokamak is used as the fusion neutron source for the purpose of properly exploiting the current plasma physics and fusion technologies. The main parameters of the neutron source are given in Table 1. The utilization of the proven or expectable fusion technologies can indicate a practical path to early application of fusion energy.

Table 1
Main parameters of ITER.

Parameters	Values
Major radius (m)	6.2
Minor radius (m)	2.0
Plasma elongation	1.85
Fusion power P_{fus} (MW)	500
Neutron wall loading Γ_n (MW/m^2)	0.57

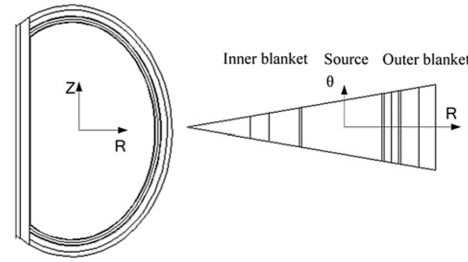

Fig. 1. Cross-sections of the blanket.

If the blanket provides an output of 3000 MW fission power, energy multiplication factor (M) must be greater than 6 to keep P_{fus} less than 500 MW and Γ_n less than 0.57 MW/m^2.

2.2. Description of the blanket

Because there are 20 toroidal coils in ITER, the torus is modeled by 20 equal sectors (Aymar et al., 2002; Murata et al., 2005) with a toroidal segment of 18° as shown in Fig. 1. The plasma chamber is surrounded by the inner blanket and outer blanket. The inner blanket consists of tritium breeding zone and shielding layer. In each sector, the outer is divided into 4 small modules along the vertical direction (along the Z direction in Fig. 1) as shown in Fig. 2. For each module, it is separately cooled by pressurized light water. It is convenient for installation and replacement of the blanket. The outer blanket is designed to achieve energy multiplication and tritium breeding. And it consists of the first wall, fission zone, tritium breeding zone and shielding layer in radial direction (along the R direction in Fig. 1) as depicted in Fig. 3.

The fission zone is designed based on the well-developed pressurized light water cooling technologies. The pressure of the coolant is 15.5 MPa. The coolant flows in the pressure tubes along the arrowhead direction as shown in Fig. 3. There are two rows of pressure tubes in the radial direction, and the pressure tubes are put along the poloidal direction, namely perpendicular to the radial direction and axial direction as displayed in Fig. 2. There are 37 pins in each tube. The structure and the dimensions of the bundle are depicted in Fig. 4. The coolant to fuel volume ratio is about 1.0.

In the blanket, the pressurized coolant is limited in the pressure tubes which serve as the pressure boundary. The thickness of pressure tube wall needs to be calculated for the sake of safety. It is calculated as followed:

$$t > \frac{pD}{2[\sigma] - p} \tag{1}$$

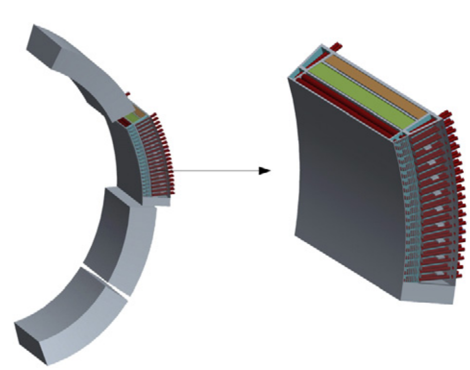

Fig. 2. Structure of the outer blanket.

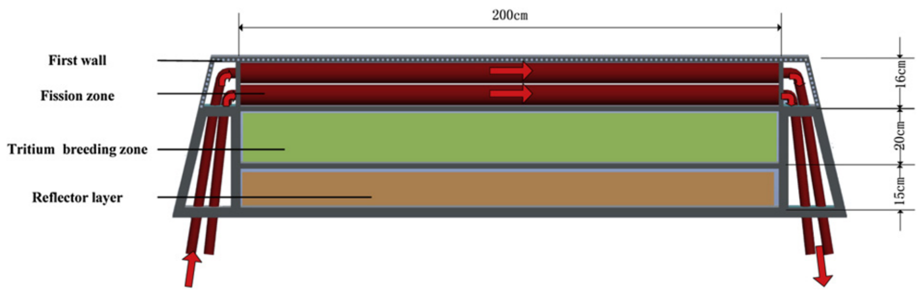

Fig. 3. Cross-section of outboard.

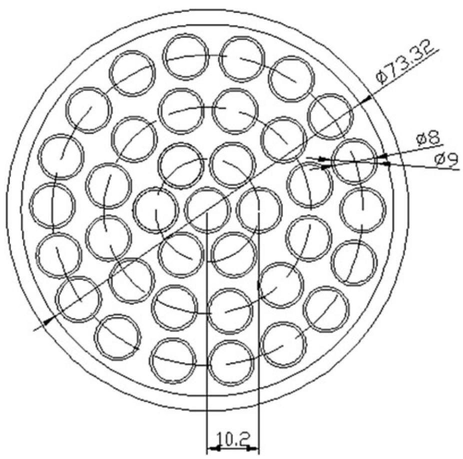

Fig. 4. Cross sectional view of the outer blanket (mm).

where, t is the thickness of pressure tube wall; the equivalent diameter D is 73.32 mm; the pressure p is 15.5 MPa; the allowable stress σ is 450 MPa. Considering enough allowance, the thickness of pressure tube wall is set to be 3.5 mm.

The material composition and the dimensions of the blanket are given in Table 2. Vanadium alloy (V4Cr4Ti) is adopted as the first wall and structure material due to its excellent anti-radiation and neutronics performance (Haci Mehmet Sahin, 2007). Stainless steel SS316 is used as shielding material. Li_2O is applied for tritium breeding due to its high breeding potential, low vapor pressure at high temperature, and low activation (Abdou and The APEX Team, 1999). The enrichment of ^6Li is set to be 90%. The first wall, tritium breeding zone and structural material are cooled by helium and the shielding layer by light water.

Spent nuclear fuel discharged from 33GWD/tU LWRs and natural uranium oxide are employed for energy and neutron multiplication. The composition of the nuclear fuel is given in Table 3. From Table 3, it can be seen that in SNF the total atomic density of fissile isotopes (^{235}U and ^{239}Pu) is higher than that in natural uranium. For the sake of simplicity, the case using SNF is denoted by Case 1, and the case using natural uranium oxide is Case 2. In the outer blanket, the total mass of nuclear fuel is about 250 tons. The material of cladding and pressure tube wall is zircalloy.

Table 2
Material composition and dimension of the blanket.

	Material and volume fraction (%)	Thickness (cm)
Outboard		
First wall	V4Cr4Ti (50)+He (50)	2
Structure material	V4Cr4Ti (50)+He (50)	2
Tritium breeding material	Li_2O (80)+He (20)	20
Shielding material	SS316 (75)+H_2O (25)	15
Cladding/pressure tube material	Zricalloy (100)	0.05/3.5
Inboard		
First wall	V4Cr4Ti (50)+He (50)	2
Tritium breeding material	Li_2O (80)+He (20)	20
Shielding material	SS316 (75)+H_2O (25)	15

3. Numerical results

3.1. Calculation tools

The neutronics calculations have been performed by a home developed code system based on the conventional two-step scheme widely used in the design of fission reactors. The computational flow chart is presented in Fig. 5. Firstly, the lattice calculations are carried out to obtain the detailed heterogeneous flux distribution within a pressure tube bundle, and the detailed flux distribution is used to calculate the homogenized and condensed cross sections for the bundles. Secondly, the transport calculation of the homogenized blanket proceeds using the homogenized and condensed cross sections. The former is conducted by the lattice code DRAGON4 (MARLEAU et al., 2011) which is widely used to generate the homogenized and condensed cross sections for pressure tube bundle calculations (Varin and Marleau, 2006). In the lattice calculation, a WIMSD-IAEA 172-group data library (Lopez Aldama et al., 2003) derived from ENDF/B-VII is used. The multigroup Monte Carlo code (Li et al., 2003) is employed to perform the blanket transport calculation for its statistically converged results of sufficient precision and the powerful capability to describe the complex geometry of the blanket. Between

Table 3
Atomic densities of nuclear fuel.

	Nuclides	Atomic density ($10^{24}/cm^3$)
SNF	^{241}Am	8.8152E-07
	^{242}Am	1.5450E-08
	^{243}Am	2.1917E-06
	^{242}Cm	3.1209E-07
	^{244}Cm	6.7419E-07
	^{237}Np	9.7922E-06
	^{238}Pu	3.3011E-06
	^{239}Pu	1.4285E-04
	^{240}Pu	5.4187E-05
	^{241}Pu	3.4434E-05
	^{242}Pu	1.1484E-05
	^{234}U	4.7844E-08
	^{235}U	2.0162E-04
	^{236}U	8.8416E-05
	^{238}U	2.1880E-02
	^{16}O	4.1240E-02
Natural uranium oxide	^{235}U	1.7673E-04
	^{238}U	2.4265E-02
	^{16}O	4.8884E-02

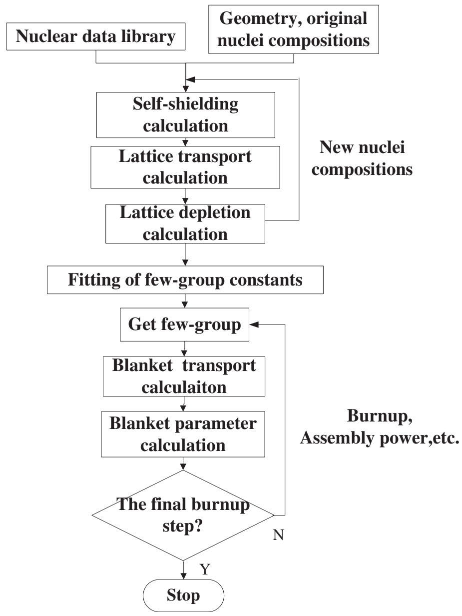

Fig. 5. Computational flow chart of neutronics code system.

the two codes, there is a link procedure to fit the homogenized and condensed cross sections to a function of different variables. Following the blanket transport calculation, another procedure is used to calculate some important parameters such as energy multiplication factor, tritium breeding ratio and the first wall loading. The code system has been widely tested, and has a good agreement with the references.

The thermo-mechanics calculations are performed using the finite element code ANSYS to analyze the temperature and thermal stress distributions in the first wall. A subchannel code is developed to calculate the thermal-hydraulic feature of the fission zone, and this code has been verified by comparing with the ANSYS CFX.

3.2. Thermo-mechanics performance of the first wall

The first wall which is an important part of the blanket needs very detailed design. Due to the adoption of pressure tube, the first wall would not suffer the high pressure from the fission zone, which allows a great reduction in the first wall thickness. Helium is chosen as the coolant of the first wall. The volume fraction of helium is 50%. The cross section view of the first wall is shown in Fig. 6. Helium in adjacent channels flows in opposite directions, which is beneficial to improving the thermal efficiency. The inlet temperature of helium is set to be 250 °C. The pressure of the helium is 8 MPa, and the mass flow rate is 1.9×10^{-3} kg/s for each channel.

Thermo-mechanics calculations are performed to analyze the temperature and thermal stress distributions in the first wall by the finite element code ANSYS. Considering symmetry and assuming that the flow was symmetric, three channels are adopted in the calculations. The hexagonal mesh is generated with the selected first wall model as shown in Fig. 7. In the calculations, the number of the elements is 177345.

The temperature distribution and stress distribution in the first wall are shown in Fig. 8 and Fig. 9, respectively. It can be seen from the results that the max temperature of the first wall is 377 °C, lower than the operation limited temperature of V4Cr4Ti (~700 °C (Abdou and The APEX Team, 1999)). The max thermal stress is 370.19 MPa, satisfying 3 Sm criteria of the material.

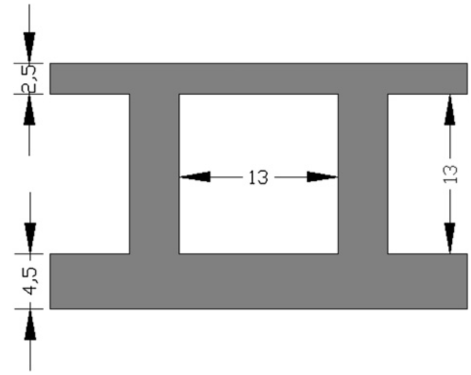

Fig. 6. Cross sectional view of the first wall (mm).

3.3. Neutronics performance

The neutronics performance parameters such as effective neutron multiplication factor (k_{eff}), energy multiplication factor (M), tritium breeding ratio (TBR) and the neutron energy spectra in fission zones have been calculated.

Light water has strong moderation effect, so it is important to calculate k_{eff} to ensure that the hybrid reactor will not reach critical conditions ($k_{eff} = 1.0$) in any case. The values of k_{eff} changing with time are displayed in Fig. 10. It can be seen that k_{eff} are less than 1.0 during the lifetime. The values of k_{eff} drop at the beginning two days due to the neutron absorption by the poison isotopes ^{135}Xe and ^{149}Sm. k_{eff} of Case 1 are greater than that of Case 2, because the total atomic density of the fissile nuclides in the former is greater than that in the latter. The evolution of k_{eff} is affected by the changes of fissile isotopes.

Similar to k_{eff}, energy multiplication factor is influenced by the content of the fissile isotopes and the accumulation of the fission products. The results of M during the lifetime are given in Table 4. The minimum of M is 6.7 appearing at the beginning of the lifetime in Case 2. The corresponding fusion power and first wall loading are 450 MW and 0.51 MW/m^2, respectively. They are both lower than those of ITER. In the design of FFHR, an important parameter limiting the lifetime of the system is the radiation damage in the first wall. The lower first wall loading is helpful to reduce radiation damage and extending the lifetime of the system.

To obtained tritium self-sufficiency, tritium breeding ration must be greater than 1.05. The results of tritium breeding are shown in Table 4. The minimum of TBR in the two cases are 1.18 and 1.10, respectively. The blanket can maintain tritium self-sufficiency during the lifetime.

The relative neutron spectra tallied in 361 energy groups (per source neutron) in the fission zones are shown in Fig. 11. From the results, it can be seen that both fast neutrons and thermal neutrons have formed peaks. The fertile isotope ^{238}U is converted to ^{239}Pu by the absorption of fast neutrons. And then ^{239}Pu and ^{235}U with a large fission cross section in the thermal neutron spectrum will be

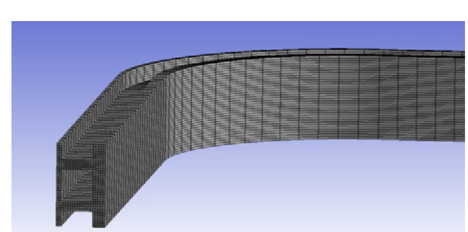

Fig. 7. Mesh of the first wall.

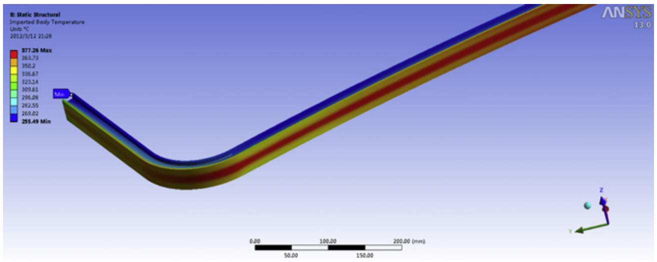

Fig. 8. Temperature distribution in the first wall.

burned. Because of the fissile Pu in the spent fuel, the neutron spectrum in Case 1 is harder.

3.4. Changes of fissile isotope

The changes of the fissile isotopes are depicted in Fig. 12. The atomic densities of ^{235}U without supplement from fertile isotopes decrease in the both cases during the fuel burnup. In the Case 1, ^{239}Pu is burned more slowly than ^{235}U, for which, the reason is that ^{238}U is constantly converted to ^{239}Pu as discussed above. As for Case 2, the atomic density of ^{239}Pu increases during the lifetime.

In a power plant, it is indispensable to examine whether there is a risk of nuclear proliferation. Due to the high spontaneous fission cross section of even plutonium isotopes, such as ^{238}Pu and ^{240}Pu, the presence of a sufficient amount of these isotopes renders the nuclear fuel to a reactor grade. In Case 1, after five years operation, the contents of ^{238}Pu and ^{240}Pu are 5.36% and 19.0%, respectively, and the weight percent of ^{239}Pu is 52.0%. In Case 2, at the end of lifetime, the contents of ^{238}Pu and ^{240}Pu are 2.45% and 17.9%, respectively. The content of ^{239}Pu is 61.8%. In the study (Sahim and Ubeyli, 2004), it was indicated that the ^{240}Pu content must only be $<5\%$ in plutonium fuel of weapon grade. In the two cases, the ^{238}Pu and ^{240}Pu contents are high enough to make the plutonium never reach nuclear weapon grade quality.

3.5. Radiation damage in the first wall

An important parameter limiting the lifetime of the hybrid reactor is radiation damage in the first wall. The radiation damage for first wall material mainly includes displacements of the atoms from their lattice sites due to collisions with highly energetic fusion neutrons and gas production in the metallic lattice resulting from diverse nuclear reaction. In this paper, two parameters: helium production rate and displacements per atom (DPA) are calculated. The average He-production rate and DPA per year for each case are given in Table 5. Case 2 calling for higher fusion power to maintain the 3000 MWth output has larger DPA and He-production rate.

Assuming that V4Cr4Ti can withstand an accumulated DPA of about 100 (Haci Mehmet Sahin, 2007), the first wall have a lifetime

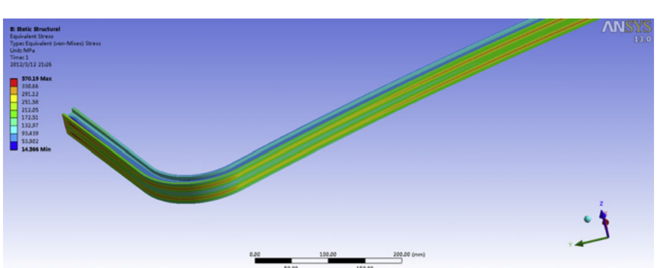

Fig. 9. Thermal stress distribution in the first wall.

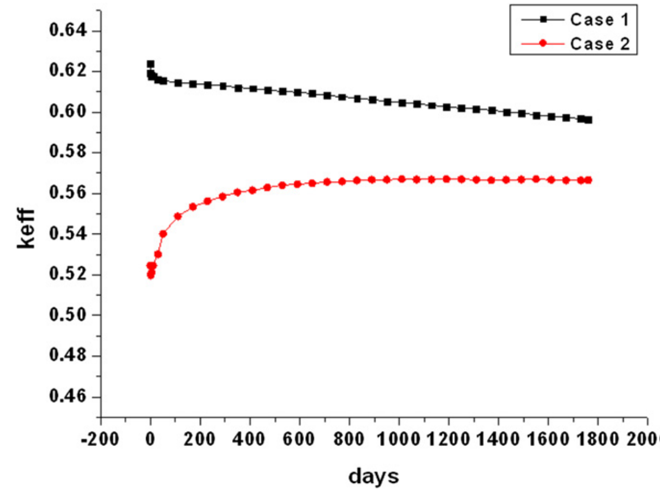

Fig. 10. k_{eff} as a function of time.

of 14.6 years and 11.2 years for the two cases, respectively. If 500appm is considered as the criterion for helium production in the first wall (Haci Mehmet Sahin, 2007), the first wall will be replaced every 29.8 years and 21.3 years for the two cases.

3.6. Thermal-hydraulics performance

Since the power distributions in the two cases are similar with each other, the thermal-hydraulics results based on Case 1 are given in this paper. In a sector, the total thermal power in the first row is 85.9 MW and it is 64.1 MW in the second row. The preliminary results show that the axial power distribution is quite uniform while the radial power distribution of the fusion–fission hybrid reactor blanket is very uneven. The non-homogeneous power factor is 1.34. The coolant pressure was assumed to be 15.5 MPa which is the same with the PWR. Coolant inlet temperature was assumed to be 292 °C.

Fig. 13 shows the corresponding variations of max cladding surface temperature and outlet coolant average temperature along the mass flow rate. As expected, a lower mass flow rate leads to a higher surface temperature and a higher outlet coolant average temperature. The temperature of the cladding material must be lower than its max operation temperature of 330 °C to avoid that the heat flux from the fuel rods to the coolant exceeds the critical heat flux value. And considering enough allowance, the average outlet temperature of the coolant is set to be 312 °C. Based on this, the inlet mass flow rate of the coolant in the first row is set to be 8.0 kg/s from Fig. 13. The pressure drop is 0.067 MPa.

Because the total power in the second row is lower than that in the first row, the inlet mass flow rate of the second row is less than the first row to ensure that the average temperature rise of the coolant in the both rows is equal to each other. The inlet mass flow rate is calculated to be 6.0 kg/s. The different mass flow rate is obtained by adding flow distribution plate at the entrance.

Table 4
M and TBR as a function of time.

Year	Case 1		Case 2	
	M	TBR	*M*	TBR
1	9.4	1.26	6.7	1.10
2	9.1	1.23	7.4	1.15
3	8.9	1.22	7.5	1.15
4	8.7	1.20	7.5	1.14
5	8.5	1.18	7.4	1.14

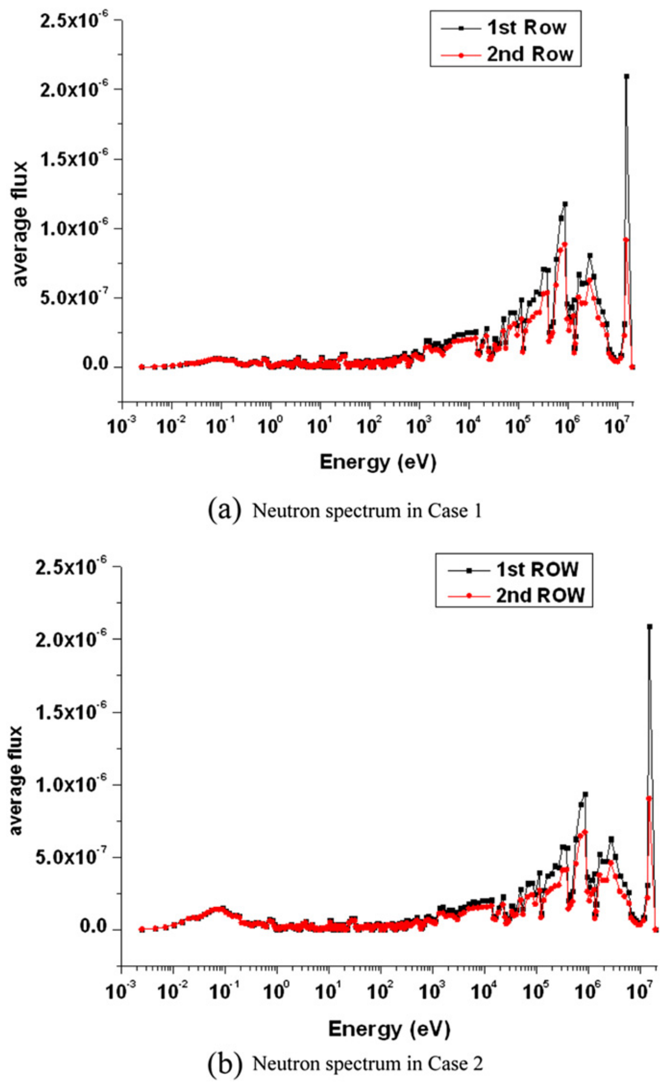

Fig. 11. Neutron energy spectra (per fusion source) in fission zones.

3.7. Some other discussions

An economically competitive fusion system requires a high power density, simpler technology, lower material constraints and so on (Abdou and The APEX Team, 1999). The proposed hybrid reactor has some advantages which will translate into good economics.

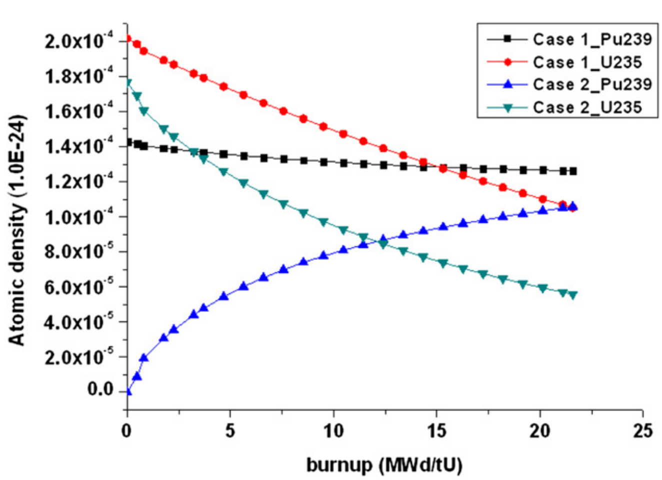

Fig. 12. Atomic densities of the fissile isotopes as a function of burnup.

Table 5
Average DPA and He-production rate per year.

	DPA/year	He-production (appm/year)
Case 1	6.84	16.75
Case 2	8.90	23.45

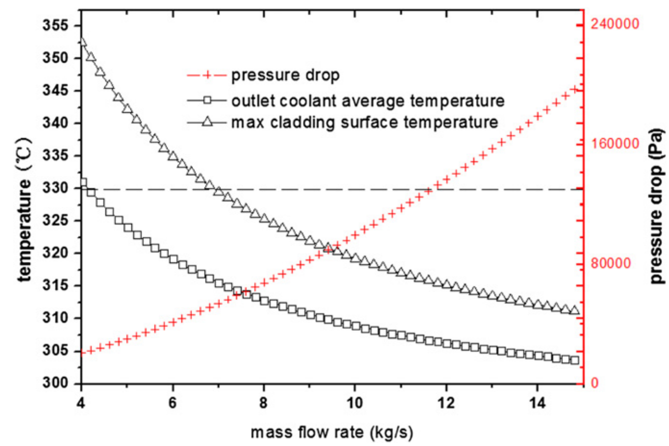

Fig. 13. Values of temperature versus mass flow rate.

Firstly, the average power density in the blanket is about 34 MW/m^3. In an ITER type pure fusion reactor, the core power density is about 1.2 MW/m^3 (Abdou and The APEX Team, 1999). The proposed FFHR is higher by a factor of 28. Secondly, from the numerical results, the needed fusion power is less than 450 MW, and the corresponding first wall loading are is less than 0.51 MW/m^2. The lower requirements are effective to simplify the technologies and lower the material constraints. Thirdly, the cheap light water is used as the coolant in the fission blanket. The well-developed cooling technologies and energy conversion technology can be easily employed in the hybrid discussed in the paper. So the investment cost and technology risk will be reduced. All these factors will improve the economics of the proposed hybrid reactor.

4. Conclusions

In this paper, a water-cooled pressure tube blanket for the fusion–fission hybrid reactor is proposed for energy production using the plasma conditions and configurations of the ITER, and the well-developed pressurized light water cooling technology. The spent nuclear fuel (SNF) discharged from 33GWD/tU LWRs and natural uranium oxide are employed for energy multiplication. Li_2O is used for tritium breeding. The major results of this study are summarized as follows:

(1) The first wall with the thickness of 20 mm is sufficiently safe. The maximum temperature and thermal tress in the first wall is lower than the limited values of the selected material.
(2) The blanket can operate for 5 years without refueling. The values of k_{eff} are always less than 1.0, during the lifetime. The minimal energy multiplication factor is 6.7 and the corresponding maximal fusion power and the first wall loading are 450 MW and 0.51 MW/m^2, respectively. The required parameters are within the ITER levels. The minimal TBR is 1.10 in the lifetime.
(3) According to the radiation damage analysis, the first wall can have a lifetime of more than five years.

(4) After five years operation, the content of fissile ^{239}Pu is 52.0% when the spent nuclear fuel is loaded in the blanket; and it is 61.8% when natural uranium is used. The final plutonium cannot reach a weapon grade.
(5) The thermal-hydraulic analysis indicates the pressurized light water cooling technology can be used in the blanket. The cladding temperature and the coolant temperature are well below the limited values and a large safety margin is provided.

The numerical results indicate the pressurized light water-cooled pressure tube blanket for the fusion–fission hybrid reactor is feasible for energy production.

Acknowledgments

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