A Fusion-Fission Reactor Concept Based on Viable Fusion and Fission Technologies

Y. Wu 1, 2), J. Jiang 1, 2), M. Wang 2), M. Jin 1, 2), J. Liu 2), Z. Chen 1, 2), Y. Chen 1, 2), Y. Wang 1, 2), Q. Zeng 1, 2), Y. Bai 1, 2), S. Liu 1, 2), H. Chen 1, 2),Q. Huang 1, 2), Y. Song 1, 2), C. Li 1, 2), L. Hu 1, 2), P. Long 1, 2), FDS Team

1) Institute of Plasma Physics, Chinese Academy of Sciences, Hefei, Anhui, 230031, China;
2) School of Nuclear Science and Technology, University of Science and Technology of China, Hefei, Anhui, 230031, China

E-mail of main author: ycwu@ipp.ac.cn

Abstract. A multi-functional fusion-fission reactor concept named FDS-MF simultaneously for nuclear waste transmutation, fissile fuel breeding and thermal energy production based on viable technologies i.e. available or limitedly extrapolated fission and fusion technologies, from the experimental stage named FDS-MFX to the DEMO stage named FDS-SFB, is developed based on the re-examination of the feasibility, capability and safety & environmental potential of three types of fusion-fission hybrid reactors, i.e. the energy multiplier named FDS-EM, the fuel breeder named FDS-FB, waste transmuter named FDS-WT. The tokamak can be designed based on relatively easy-achieved plasma parameters extrapolated from the successful operation of the Experimental Advanced Superconducting Tokamak (EAST) in China and other tokamaks in the world, and the subcritical blanket can be designed based on the well-developed technology of fission power plant. The conceptual design of fusion plasma, fission blanket and fuel cycle have been presented with the preliminary performance analyses covering neutronics, thermal-hydraulics, and thermo-mechanics. The calculation results have shown that the concept based on viable technologies can meet the requirements of tritium self-sufficiency, effective nuclear waste burner, good fission fuel breeding and sufficient energy gain, and operate for at least 10 years without fuel unloading and reloading.

Key Words

1. Introduction

The world needs a great deal of carbon free energy for civilization to continue. Nuclear power is attractive for helping cut carbon emissions and reducing imports of fossil fuel. Although the recent experiments and associated theoretical studies of fusion energy development have demonstrated the feasibility of fusion power, it is commonly realized that it needs hard work before pure fusion energy could be commercially and economically utilized. Some countries are speeding up the development of their fission industry. In China, the government has decided to continuously develop nuclear power with a mid-term target of 40GWe or more in 2020. If only Pressurized Water Reactor (PWR) is used to meet the huge nuclear capacity requirement, there may be a shortage of fissile uranium to support fission industry and an increase of long-lived nuclear wastes coming from fission reactors. Therefore, any activity to utilize fusion energy technology [1-4] and to solve the problems has been welcome. The fusion-fission hybrid systems/reactors have the potential attractiveness of good nuclear spent fuel burner performances and plenty of fuel and easing the requirement of fusion plasma technology (with a low fusion gain Q) and plasma-facing material technology (with a low neutron wall loading), which represent a possible use of fusion power to build a power reactor with the current understanding of plasma physics. A lot of research activities had been done to evaluate the possibility of the hybrid systems in China and other countries in the world [1-46], however, most of them were based on advanced fusion and fission technologies.

In this contribution, along with the achieved and ongoing efforts to establish fusion as an energy source, there is a renewed interest in fusion-fission hybrid reactors, especially based on the progress in the construction and operation of EAST [47] in China and International Thermonuclear Experimental Reactor (ITER) [48]. Three types of fusion-fission hybrid reactor concepts based on viable technologies, i.e. the energy multiplier named FDS-EM with the goal of energy production, the fuel breeder named FDS-FB with the goal of fissile fuel breeding, waste transmuter named FDS-WT with the goal of transmutation of the long-lived nuclear wastes [49-50], have been used for the re-examination of feasibility, capability and safety & environmental potential of fusion -fission hybrid systems. Then according to the results of the re-evaluation activity, a multi-functional fusion-fission hybrid reactor concept called FDS-MF has been developed simultaneously for nuclear waste transmutation, fissile fuel breeding and thermal energy production based on available or very limitedly extrapolated fission and fusion technologies. The final goal of the concept may be fission reactor Spent Fuel Burning named FDS-SFB. To achieve the final goal, an intermediate stage to develop a multi-functional experimental reactor named FDS-MFX is needed.

2. Fusion Plasma Core

The multi-functional fusion-fission reactor concept FDS-MF based on viable technologies may be achieved through two stages, i.e. the first stage named FDS–MFX is the experimental stage. and the tokamak can be designed based on relatively easy-achieved plasma parameters extrapolated from the successful operation of EAST in China and other tokamaks in the world, e.g. with a tokamak core of the fusion power of ~50MW, the power gain of ~1, the major radius of 4m, the minor radius of 1m, the average neutron wall loading of ~0.2 MW/m^2 as shown in Table I. The second stage named FDS–SFB is the DEMO stage, with a tokamak core of the fusion power of ~150MW, the power gain of ~3, the major radius of 4m, the minor radius of 1m, the average neutron wall loading of ~0.5 (MW/m^2).The fusion power for DEMO is designed based on evaluation activity of the performance of nuclear spent fuel burner, fissile fuel breeding and thermal energy production (See *Section 4.1*).

A set of reference plasma parameters of FDS-MFX or FDS-SFB is selected by using SYSCODE code [51], which is developed by the FDS Team, China. The MHD equilibrium is calculated by using EFIT equilibrium code [52] based on the plasma parameters and *FIG.1.* shows the magnetic configuration. The main plasma parameters of FDS-MF are listed in Table Ⅰ, as well as the parameters of the EAST and ITER for purpose of comparison. It was relatively easy-achieved due to the low plasma parameters and could be extrapolated from the operation of the EAST.

TABLE I: THE CORE PARAMETERS OF FDS-MF, EAST AND ITER

Parameters	ITER	EAST	FDS-SFB	FDS-MFX
Fusion power (MW)	500	-	150	50
Major radius (m)	6.2	1.95	4	4
Minor radius (m)	2	0.46	1	1
Aspect ratio	3.1	4.2	4	4

Plasma elongation	1.85	1.8	1.78	1.7
Triangularity	0.33	0.45	0.4	0.45
Toroidal magnetic field on axis (T)	5.3	3.4-4.0	6.1	5.1
Safety factor / q-95	3	-	3.5	2.03
Plasma current (MA)	15	1.5	6.3	6.1
Average neutron wall load (MW/m^2)	0.57	-	0.49	0.17
Average surface heat load (MW/m^2)	0.27	0.1-0.2	0.1	0.1
Fusion gain	>10	-	3	0.95
Normalized beta, β_N (%)	2.5	-	3	3

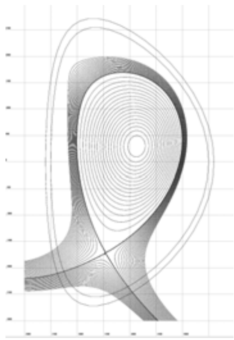

FIG. 1. Magnetic configuration and profiles of current density and plasma pressure of FDS-MFX

3. Blanket Concept

3.1. Fission Blanket

3.1.1. Re-evaluated Blanket Concepts (FDS-EM/ FDS-FB/ FDS-WT)

Three types of hybrid concepts i.e. FDS-EM, FDS-FB and FDS-WT are conceptually designed and re-evaluated based on available or very limited extrapolated fusion (i.e. a fusion power of 50~500MW) and fission technologies (i.e. Water-cooled PWR or He-cooled HTGR technologies).

The conceptual design of FDS-EM, FDS-FB and FDS-WT, which consists of fusion plasma core parameter design and optimization, fission blanket and fuel cycle, and reference blanket module design, has been presented. And the preliminary performance analyses covering neutronics, thermal-hydraulics and thermo-mechanics have been carried out. FDS-EM / FDS-FB / FDS-WT is a practical path to the early fusion application for energy production / fissile fuel breeding / nuclear waste transmutation in a sub-critical reactor, which is based on available fusion technologies (the level extrapolated from the operation of the EAST tokamak) and mature fission reactor technologies such as the PWR or helium-cooled High Temperature Gas-cooled Reactor (HTGR) technologies.

The neutonics analyses showed the maximum energy multiplication factor M (M, which is defined as the ratio of fission power to the source neutron power (80% of fusion power in the deuterium-tritium fusion fuel cycle)) can be ~130, the maximum fissile fuel breeding ratio BSR (BSR, which is defined as the ratio of the fissile plutonium mass bred by FDS-FB to the fissile plutonium mass depleted by a referred PWR (3000MWth PWR with fuel burned to 33 GW.D/T) per year)can be ~10, the maximum waste transmutation ratio TSR (TSR, which is defined as the ratio of the waste mass transmuted by FDS-WT to the waste mass produced by a referred PWR per year) can be ~15, depending on specific designs.

3.1.2. Multi-functional Blanket Concept (FDS-MF)

- **Experimental Stage (FDS-MFX)**

The blanket system of FDS-MFX is designated to check and validate the DEMO reactor blanket relevant technologies. Thus, the design of the blanket is consistent with the design of DEMO reactor blanket. For the inner board, it is only designed to breed tritium. Outer board blanket is designated to demonstrate the hybrid technology with fission materials.

The low activation ferritic-martensitic steel (RAFM), e.g. the CLAM (China Low Activation Martensitic) steel is employed as a candidate structural material. Based on the different purposes of the FDS-MFX, there are two sub-phases of choosing of the fission fuel: natural uranium (NU) for hybrid reactor principle validation phase in prior-period of the experiment, and spent fuel (SF) from PWRs (uranium, plutonium and minor actinides) for engineering validation (such as neutronics, thermal-hydraulics and material validation) phase in the late period of the experiment. The form of carbide and plate-type fuel is adopted in accord with the DEMO reactor design.

The coolant and tritium breeder of FDS-MFX has been chosen in view of the DEMO blanket design. Thus, helium is served as the coolant of fission fuel zone, and Li17Pb83 eutectic severs not only as tritium breeder but also as coolant by itself, where lead is also a kind of neutron multiplier.

Although the whole blanket is not designed to pursue neutronics performance for the experimental object, a sufficient Tritium Breeding Ratio (TBR) of blanket to maintain the fusion core which is necessary has been estimate and achieved. The basic materials composition and radial sizes of the reference module are shown Table II

TABLE II: MATERIALS COMPOSITION AND RADIAL SIZES OF FDS-MFX

Zones	Material Component (%)	Thickness (cm)
Inboard blanket		
First Wall	CLAM (75)+H2O(25)	2
TB zone	LiPb(6Li:90%)(100)	38
Shield layer	CLAM (50)+He (50)	40
Outboard blanket		
First Wall	CLAM (50)+He(50)	3
Fission Fuel zone	**NU**: UC(21)+SiC(21)+He(58)	20
	SF: SF-55 wt % U (9) +SiC(9)+He(82)	20
Structural walls	CLAM (50)+He (50)	3
TB zone	LiPb(6Li:90%)(100)	50
Reflector layer	C(100)	30

- **DEMO Stage (FDS-SFB)**

The general design of the FDS-SFB is to have the subcritical blanket which interacts with a copious source of fusion neutrons provided by the fusion core to achieve tritium breeding and energy multiplication / fissile fuel breeding / waste transmutation. The blanket system has been designed based on the well-developed technologies of Gas-Cooled Fast Reactor (GFR) with helium gas blankets. For FDS-SFB, besides the shielding modules, three types of functional blanket zones i.e. tritium breeding zone, spent fuel burning zone and fissile fuel breeding zone are designed, respectively. For the inner board, it is only designated to breed tritium. The 3D configuration of FDS-SFB is shown in *FIG. 2*.

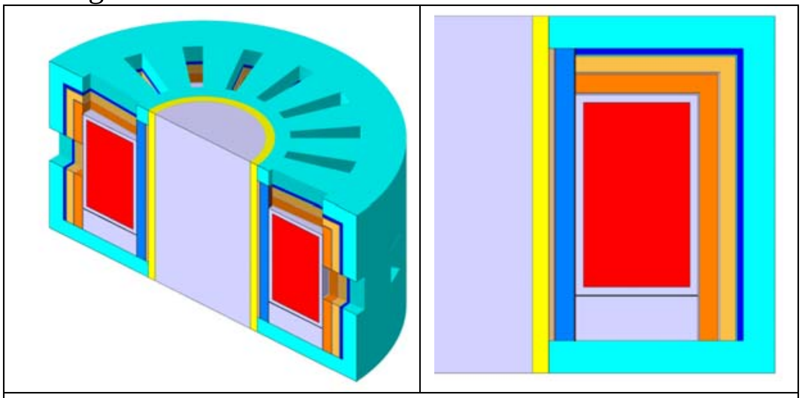

FIG. 2. The 3D configuration of FDS-SFB

The (RAFM) is considered as a candidate structural material. The fuel form of carbide is adopted for FDS-SFB based on two main reasons: one is the form of carbide fuel have high density of heavy metal and good performance on stability and excellent characteristics on compatibility with clad materials Silicon carbide (SiC), SiC carbide provides the mechanical stability to the fuel and is the main diffusion barrier to the release of fission products etc.

The form of carbide and plate-type fuel is adopted which refers to the reference [53]. Helium is served as the coolant of fission fuel zone. $Li_{17}Pb_{83}$ eutectic severs not only as tritium breeder but also as coolant by itself.

The performance parameters such as effective neutron multiplication factor K_{eff}, TBR, M, BSR, and Transmutation Support Ratio (TSR_{LLMA}, which is defined as the ratio of the minor actinide waste mass transmuted by FDS-SFB to the waste mass produced by a referred PWR per year) of the blanket have been calculated by the integrated multi-functional neutronics calculation and analysis code system named VisualBUS [54-56] and the Hybrid Evaluated Nuclear Data Library named HENDL [57]. The time-dependant fuel loading and cycling are preliminarily optimized based on the one-dimensional geometrical models. For these models, the basic materials composition and radial sizes, which are the optimization based on the preliminarily optimized neutronics performances, are shown in Table III.

TABLE III: MATERIALS COMPOSITION AND RADIAL SIZES OF FDS-SFB

Zones	Material component (%)	Thickness(cm)
Inboard blanket		
First Wall	CLAM(50)+He(50)	2
Tritium Breed Zone	LiPb(6Li:90%)(100)	38
Shield layer	CLAM(75)+H2O(25)	40
Outboard blanket		
First Wall	CLAM(50)+He(50)	3
Fuel zone	**50MW P_{fusion}**: SF-55 wt % U (13.7) +SiC(13.7)+He(72.6)	20
	150MW P_{fusion}: SF-55 wt % U (12.5) +SiC(12.5)+He(75)	20
	500MW P_{fusion}: SF-55 wt % U (8.8) +SiC(8.8)+He(82.4)	20
Structure Wall	CLAM(50)+He(50)	3
Fuel zone	DU(50/150/500MW P_{fusion}) : UC(40)+He(40)+SiC(20)	40
Structure Wall	CLAM(50)+He(50)	3

Tritium Breed Zone	LiPb(6Li:90%)(100)	20
Structure Wall	CLAM(50)+He(50)	2
Reflector layer	C(100)	30
Shield layer	CLAM(75)+H2O(25)	50

3.2. Blanket Reference Module Design

- **Experimental Stage (FDS-MFX)**

This blanket shown in *FIG.3* features an outboard blanket reference module. The blanket module poloidal length is 1.2m. These modules are installed in 4 poloidal positions with a toroidal segment of 11.25º (32 modules). Each blanket module divided into two zones along the radial direction, which are the fission fuel zone and the tritium breeding zone, respectively. In fission zone (natural uranium or spent fuel zone), plate type of fuel is proposed. Basically, two ceramic plates enclose the honeycomb structure which contains pellets of the elementary fuel compound. The material choices are a mixed carbide ceramic for the fuel compound and a SiC ceramic for the structures.

Inboard blanket is a banana type pure tritium breeding blanket module of 22.5^{o} section angle, which only consists of LiPb channels to maintain the TBR of the whole reactor.

The LiPb feeds into the blanket from the bottom of the blanket and flow out at the top. The coolant helium collects at bottom, and then feeds into FW and fission fuel zone to cool the FW and fuel assemblies shown in (shown in *FIG.4.*).

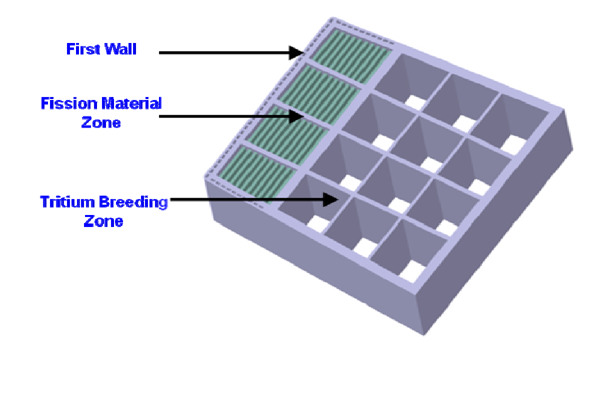

FIG. 3. Skematic view of outboard blanket of FDS-MFX

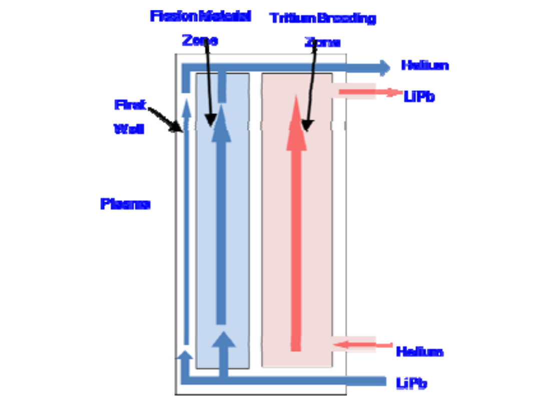

FIG. 4. Coolant flow schme of outboard blanket of FDS-MFX

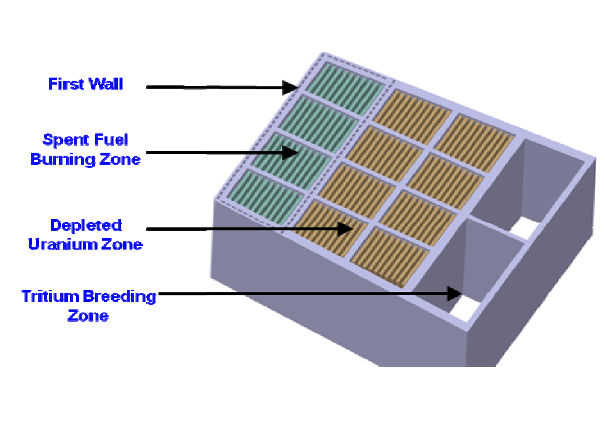

FIG. 5. Skematic view of outboard blanket of FDS-SFB

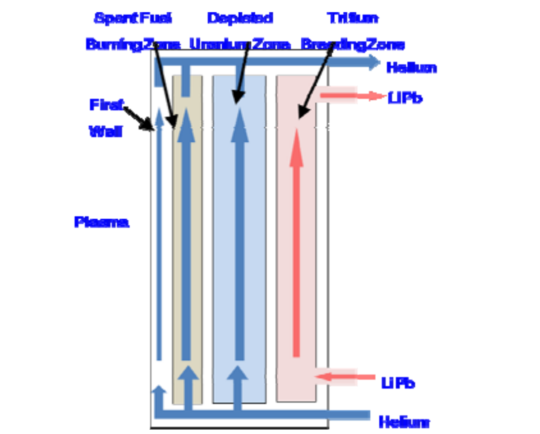

FIG. 6. Coolant flow schme of outboard blanket of FDS-SFB

- **DEMO Stage (FDS-SFB)**

This blanket shown in *FIG.5* features an outboard blanket reference module. The blanket module poloidal length is 1.2m. These modules are installed in 4 poloidal positions with a toroidal segment of 11.25º (32 modules). Each blanket module divided into three zones along the radial direction, which are the spent fuel burning zone, depleted uranium zone and the tritium breeding zone, respectively. In fission zone (the spent fuel burning zone, depleted uranium zone), plate type of fuel which likes the reference design of the GFR fuel structure is proposed.

Inboard blanket is a banana type pure tritium breeding blanket module of 22.5^{o} section angle, which only consists of LiPb channels to maintain the TBR of the whole reactor.

The LiPb feeds into the blanket from the bottom of the blanket and flow out at the top. The coolant helium collects at bottom, and then feeds into FW and fission fuel zone to cool the FW and fuel assemblies shown in *FIG.6*.

4. Performance Analyses

4.1. Neutronics Analysis

The main purpose of neutronics design and analysis is to optimize the composition and spatial arrange in the functional zones and the fuel cycle to achieve tritium self-sufficiency, sufficient energy gain and enough safety margin in the normal operation and all transient events. The main neutronics constraints and objectives of FDS-SFB blanket are listed in Table IV.

TABLE IV: MAIN CONSTRAINTS AND OBJECTIVES OF NEUTRONICS DESIGN PARAMETERS FOR FDS-SFB

Items	Constraints and objectives
Keff	≤ 0.95 (safety margin limit)
PDmax(MW/m^3)	≤ 100 (cooling capability limit)
TBR	≥ 1.05 (tritium sustainability limit)
Fuel Inventory	Minimize
Energy Multiplication(M)	>90 for Pfu=50MW
	>30 for Pfu=150MW
	>9 for Pfu=500MW
Breeding Support Ratio(BSR)	Maximize
Transmutation Support Ratio (TSR_{LLMA})	Maximize

The preliminarily optimized neutronics parameters of FDS-SFB are presented in Table V show as follows:

The energy multiplication factor M can meet the requirements of energy production with different fusion powers. The energy multiplication factor M can be 25~130 depending on specific designs.

A hybrid SFB system based on a plasma core of 150MW fusion power and a helium-cooled blanket loaded with spent nuclear fuel can be designed to support ~4.5 unit of referred PWR.

A hybrid SFB system based on a plasma core of 150MW fusion power and a helium-cooled blanket can be designed to transmute ~3 unit of standard PWR.

TABLE V THE PRELIMINARILY OPTIMIZED NEUTRONICS PARAMETERS OF

FDS-SFB

Coolant	Helium gas		
Fusion power	50MW	150MW	500MW
Keff	0.95	0.91	0.76
TBR	3.52	2.13	1.00
Fuel inventory(SF) (t HM)	124	113	80
Pth(GW)	5.31	8.94	10.58
PDmax(MW/m3)	61.05	99.88	103.32
M	132.60	74.38	26.41
BSR	1.93	4.43	8.55
TSR_{LLMA}	0.09	3.14	0.59

4.2. Thermal-hydraulics Analysis

Thermal-hydraulics analysis of FDS-SFB outer blanket is performed considering the operation requirement limits of structural material and coolant. The lower operation limited temperature of RAFM steel is 200~300 ºC, owing to the low temperature irradiation-induced hardening and embrittlement effects. And the highest operation temperature of RAFM steel is 550 ºC. Thus , the inlet temperature and the outlet temperature of helium is assumed to be 300 ºC and 500 ºC respectively to prevent the temperature of structure material from exceeding the limits. The velocity of helium in the spent fuel burning zone, depleted uranium zone can be calculated to be 41 m/s and 8.9m/s, and the maximum pressure drop is about 0.028MPa. The fission nuclear thermal can be removed effectively; the velocity and pressure drop of helium is acceptable.

We assume that LiPb remove all the heat of tritium breeding zone, as the inlet and outlet temperature of LiPb are 300 ºC and 320 ºC, LiPb velocity is 0.009 m/s. In order to reduce MHD pressure drop of LiPb flow, the coating in the LiPb channel is employed. The pressure drop of LiPb is negligible.

The thermal-hydraulics parameters are summarized in Table VI.

TABLE VI: THE THERMAL-HYDRAULICS PARAMETERS FOR FDS-SFB BLANKET

	Spent Fuel Zone	**Depleted Uranium Zone**	**Tritium Breeding Zone**
Coolant	Helium	Helium	LiPb
Power Density	$100MW/m^3$	$10.7MW/m^3$	$0.41MW/m^3$
Coolant Fraction	50%	30%	100%
Operation Pressure	8MPa	8MPa	/
Inlet Temperature	300℃	300℃	300℃
Outlet Temperature	500℃	500℃	320℃
Velocity	41m/s	8.9m/s	0.009m/s
Pressure Drop	0.028MPa	0.003MPa	/

4.3. Thermo-mechanics Analysis

The calculations and analyses of thermal-stress of first wall, spend fuel burning zone, depleted uranium zone and tritium breeding zone of the FDS-SFB blanket module have been carried out based on the two-dimensional finite element model by using commercial finite element code ANSYS. The thermal and stress distributions are shown in *FIG.7*. The maximum temperature of structure is 521 ºC, lower than the highest operation limited temperature of RAFM steel. The maximum thermal stress is 375 MPa, satisfying 3Sm criteria (396 MPa at 500 ºC) of structure material.

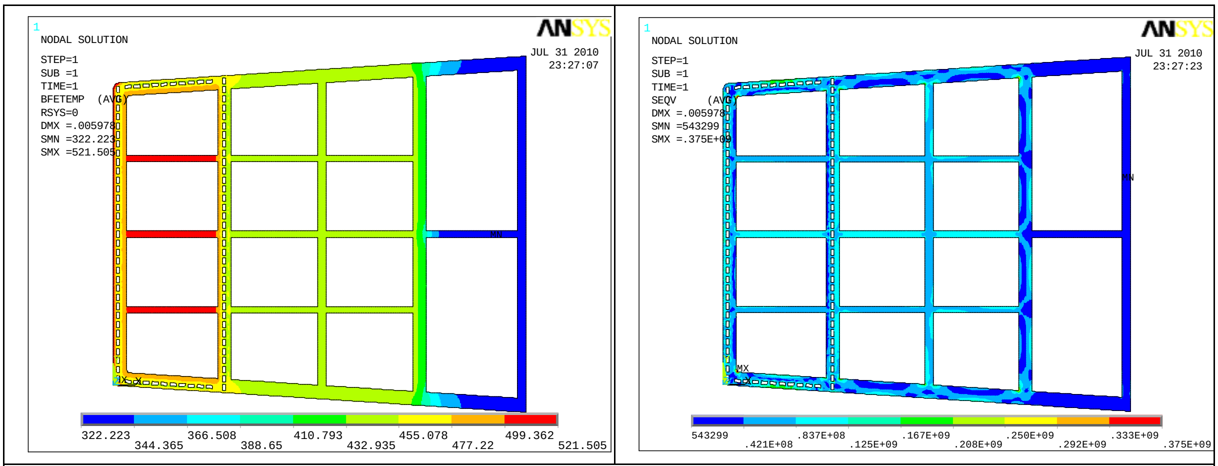

FIG. 7. Distributions of temperature (Left) and stress （Right） of FDS-SFB blanket

5. Fuel Cycle

The goal of the FDS-SFB is to achieve waste transmutation, fissile fuel breeding and energy multiplication through burning spent fuel. The spent fuel, whose composition is given in Table VII, comes from PWR spent nuclear fuel, which already removed a great part of uranium. In this fuel, uranium's mass percent is 55%, reasons to make this choice are: 1) Negative reactivity is introduced due to the fission product accumulation that comes along with the fuel depletion, This negative reactivity must be compensated for by fissile fuel breeding; 2) The consequences of admixing uranium with Transuranic (TRU) waste within fuel to produce additional TRU to reduce the reactivity decrease due to fission product buildup and TRU depletion was also evaluated.

The fuel will be reprocessed using pyroprocessing reprocessing. Pyroprocessing is based on the use of high-temperature fused chloride salts to melt the fuel, along with electro-refining techniques to separate the transuranics and fission products. A key benefit of pyroprocessing is that the actinides are never separated, but are instead recycled as one group, making them highly proliferation resistant.

A set of preliminarily optimized parameters presented in Table VIII show that the blanket system can operate for 10 years without fuel reloading. After operating for 10 years, fuel burn-up achieves 214.6 GWD/T (HM). The highest effective multiplication factor of 0.91 can meet the design limit of less than 0.95. The calculation results show that the preliminary blanket concept design can meet the requirements of tritium self-sufficiency and the sufficient energy multiplication.

TABLE VII: SPENT NUCLEAR FUEL COMPOSITIONS

Isotope	U235	U238	NP237	AM241	AM243	CM244	PU238	PU239	PU240	PU241	PU242
Mass percent (%)	0.48	54.52	2.02	2.31	0.42	0.08	0.63	23.14	10.69	3.54	2.16

TABLE VIII: MAIN NEUTRONICS PARAMETERS FOR FDS-SFB

Years	Keff	TBR	Fuel Inventory(SNF) (t HM)	Pth (GW)	Pdmax (MW/m^3)	M	Burnup(GW.D/t HM)	BSR	TSR_{LLMA}
0	0.91	2.13	113.26	8.94	99.88	74.38	28.81	-	-
1	0.89	1.96	110.29	7.82	81.94	65.03	54.01	4.43	3.14
2	0.87	1.75	107.90	6.79	67.55	56.52	75.89	8.64	4.71
3	0.86	1.64	105.93	6.24	59.34	51.89	96.00	12.34	5.54
4	0.85	1.55	104.21	5.84	59.28	48.55	114.82	15.72	6.04
5	0.84	1.53	102.66	5.67	49.76	47.21	133.10	18.90	6.36
6	0.83	1.44	101.23	5.32	44.93	44.23	150.24	21.93	6.61
7	0.83	1.39	99.91	5.08	41.50	42.28	166.61	24.82	6.83
8	0.82	1.39	98.67	5.04	39.70	41.90	182.85	27.62	7.04
9	0.81	1.36	97.51	4.94	37.65	41.12	198.77	30.33	7.26
10	0.81	1.36	96.40	4.91	36.24	40.90	214.60	32.95	7.46

6. Summary

A multi-functional fusion-fission reactor concept named FDS-MF simultaneously for nuclear waste transmutation, fissile fuel breeding and thermal energy production based on viable technologies i.e. available or limitedly extrapolated nuclear, processing and fusion technologies, from experimental stage to DEMO stage, has been developed.

The tokamak can be designed based on relatively easy-achieved plasma parameters extrapolated from the successful operation of the Experimental Advanced Superconducting Tokamak (EAST) in China and other tokamaks in the world, and the subcritical blanket can be designed based on the well-developed technology of fission power.

The design and optimization of fusion plasma core parameters, fission blanket and fuel cycle have been presented. And the preliminary performance analyses covering neutronics, thermal-hydraulics, and thermo-mechanics have been carried out.

The calculation results have shown that the concept based on viable technologies can meet the requirements of tritium self-sufficiency, effective nuclear waste burner, good fission fuel breeding and sufficient energy gain, and operate for at least 10 years without fuel unloading and reloading.

Acknowledgements

This work was supported by the Chinese Academy of Sciences with Grant No. KJCX2-YW-N35, by the National Natural Science Foundation of China with Grant No. 10675123.and by the Ministry of Science and Technology with Grants No. 2009GB101000 and 2010GB111004.

References

[1] Jose M. Martinez-Val, Mireia Piera, “Nuclear fission sustainability with hybrid nuclear cycles”, Energy Conversion and Management **48** (2007) 1480-1490.

[2] D.H. Imhoff et al., “A Driven Thermonuclear Power breeder”, CR-6, California Research and Development Corporation (1954).

[3] Leonard Jr BR. “A review of fusion-fission (hybrid) concepts”, Nucl Technol **20** (1973)161-178.

[4] J. D. Lee, “Neutronics of sub-critical fast fission blankets for D-T fusion reactors, Proceedings 7th intersociety energy Conversion engineering conference”, American

chemical society (1972) 1294.

[5] J. D. Lee, “Nuclear design of fast hybrid blankets”, Conference:2.DMFE hybrid reactor meeting, Washington, DC, USA, 2 (1977).

[6] Powell C, Hahm DJ., “Energy balance of a hybrid fusion-fission reactor”, Atomkernergie **21** (1973) 175-178.

[7] LIDSKY, L. M, “Fission-fusion systems : hybrid, symbiotic and augean”, Nucl. Fusion **15** (1975) 151-173.

[8] Blinkin VL, Novikov VM. “Symbiotic system of a fusion and a fission reactor with very simple fuel reprocessing”, Nucl Fusion **18** (1978) 893-900.

[9] F. L. TENNEY et al., “A systems study of tokamak fusion-fission reactors”, PPPL-1450, Princeton Plasma Physics Laboratory (1978).

[10] Bethe HA. “The fusion hybrid”, Phys Today **5** (1979) 44-51.

[11] Bender DJ., “Performance parameters for fusion-fission power systems”, Nucl Technol **44** (1979) 381-389.

[12] K. R. Schultz, Gas-Cooled Fusion-Fission Hybrid Reactor Systems, Journal of Fusion Energy **1**(1981) 163-183.

[13] Abdel-khalik S, et al., Impact of fusion-fission hybrids on world nuclear future, Kernforschung Szentrum Karlsruhe GmbH, Karlsruhe, KFK 2998, 1980.

[14] Berwald DH, Maniscalco J., “An economics method for symbiotic fusion－fission electricity generator systems”, Nucl Technol/Fusion **1** (1981)128-137.

[15] Rose RP, “The case for the fusion hybrid”, J Fusion Energy **1** (1981) 185-196.

[16] Lee JD, Moir RW, “Fission-suppressed blankets for fissile fuel breeding fusion reactors”, J Fusion Energy **1** (1981) 299-303.

[17] D. L. Jassby, D. H. Berwald, J. Garner, et. al., “Fast-Fission Tokamak Breeder Reactors”, Journal of Fusion Energy **5** (1986) 171-180.

[18] J. D. LEE, “Neutronic Analysis of a 2500 MWt Fast Fission Natural Uranium Blanket for a DT Fusion Reactor”, Proc.1st Topl. Mtg. Technology of Controlled Nuclear Fusion, San Diego, California, April 16-18, 1974, CONF-740402-P1,p. 223, U.S. Atomic Energy Commission (1974).

[19] L.M.Lidsky, “Fission-fusion symbiosis: general considerations and a specific example”, in Proc. Nuclear Fusion Reactors Conference, 1969 (Culham, England). Culham Laboratory Report CLM-NFR (1969) 41.

[20] J. A. MANISCALCO, D. H. BERWALD, et al., “Recent progress in fusion-fission hybrid reactor design studies”, Nuclear Technology/Fusion **1**(1981) 419-478.

[21] Moir RW, “The fusion breeder”, J Fusion Energy **2** (1982) 351-367.

[22] Takahashi H, et al., “Fissile fuel production by linear accelerator”, Trans Am Nucl Soc **43** (1982)138-140.

[23] Kazoo Furuta, Yoshiaki Oka, Shigehiro, “An Design and safety study of power producing fusion-fission hybrid reactor”, Nuclear Engineering and design/Fusion **1**(1984) 255-263

[24] E.T. Cheng and R.J. Cerbone, “PROSPECT OF NUCLEAR WASTE TRANSMUTATION AND POWER PRODUCTION IN FUSION REACTORS”, Fusion Technology **30** (1996) 1654-1658

[25] S. Taczanowski, G. DomanskaJ, Cetnar, “Fusion-driven transmutations of nuclear waste - a misconception or an incentive for promotion of fusion energy” Fusion Engineering and Design **41** (1998) 455-460.

[26] H. Nifenecker, S. David, J. M. Loiseaux and A. Giorni, “Hybrid Nuclear Reactors”, Progress in Particle and Nuclear Physics **43** (1999) 683-827.

[27] Y. Gohar, “Fusion solution to dispose of spent nuclear fuel, transuranic elements, and highly enriched uranium”, Fusion Engineering and Design **58-59** (2001) 1097-1101.

[28] A. Serikov, G. Shatalov, S. Sheludjakov, et al. “Possibility of fusion power reactor to transmute minor actinides of spent nuclear fuel”, Fusion Engineering and Design **63-64** (2002) 93-99.

[29] W. M. Stacey, J. Mandrekas, E. A. Hoffman, et al., “A fusion transmutation of waste reactor”, Fusion Engineering and Design **63-64** (2002) 81-86.

[30] E. A. Hoffman, W. M. Stacey. “Nuclear and fuel cycle analysis for a fusion transmutation of waste reactor”, Fusion Engineering and Design **63-64** (2002) 87-91.

[31] S. Sahin, M. Ubeyli, “LWR spent fuel transmutation in a high power density fusion reactor”, Annals of Nuclear Energy **31** (2004)871-890.

[32] M. Ubeyli. Transmutation of minor actinides discharged from LMFBR spent fuel in a high power density fusion reactor. Energy Conversion and Management **45** (2004) 3219-3238.

[33] Mustafa Ubeyli, “Utilization of Ceramic Uranium Fuels in ARIES-RS Fusion Reactor”, Journal of Fusion Energy **23** (2004) 41-48.

[34] I. Murata, Y. Yamamoto, S. Shido, et al., “Fusion-driven hybrid system with ITER model”, Fusion Engineering and Design **75–79** (2005) 871-875

[35] D. Ridikas, R. Plukiene, A. Plukis, et al., “Fusion-fission hybrid system for nuclear waste transmutation (I): Characterization of the system and burn-up calculations”, Progress in Nuclear Energy **48** (2006) 235-246.

[36] Y. Wu, S. Zheng, X. Zhu, et. al., “Conceptual design of the fusion-driven subcritical system FDS-I”, Fusion Engineering and Design **81** (2006) 1305-1311.

[37] Huseyin Yapıcı, Nesrin Demir, Gamze Genc, “Time-dependent Neutronic Analysis for High Level Waste Transmutation in a Fusion-driven Transmuter”, J Fusion Energ **27** (2008) 206-215.

[38] Y. Wu, the FDS teams, “Conceptual design activities of FDS series fusion power plants in China”, Fus. Eng. Des. **81** (2006) 2713-2718.

[39] Y. Wu, FDS Team, “Conceptual design of the China fusion power plant FDS-II”, Fusion Eng Des. **83** (2008) 1683-1689.

[40] H. Chen, Y. Wu, et al., “A high temperature blanket concept for hydrogen production”, Fusion Eng. Des. **83** (2008) 903-911.

[41] Y. Wu, “Progress in fusion-driven hybrid system studies in China”, Fusion Engineering and Design **63-64** (2002) 73-80.

[42] S. Zheng, Y. Wu, “Neutronics Optimization of LiPb-He Dual-Cooled Fuel Breeding Blanket for the fusion-Driven sub-critical System”, Plasma Science & Technology **4** (2002) 1421-1428.

[43] Yican Wu, “A Fusion Neutron Source Driven Sub-critical Nuclear Energy System : A Way for Early Application of Fusion Technology”, Plasma Science and Technology **3** (6) (2001) 1085-1092.

[44] Yican Wu, Lijian. Qiu, Yixue Chen, “Conceptual Study on Liquid Metal Center Conductor Post In Spherical Tokamak Reactors”, Fusion Engineering and Design **51-52** (2000) 395-399.

[45] L.J. Qiu, Y.C. Wu, B.J. Xiao, Q. Xu, Q.Y. Huang, B. Wu, Y.X. Chen, W.N. Xu, Y.P. Chen, X.P. Liu, “A Low Aspect Ratio Tokamak Transmutation System”, Nuclear Fusion **40**(3) (2000) 629-633.

[46] W. M. Stacey, “Transmutation missions for fusion neutron sources”, Fusion Eng. Des. **82** (2007) 11-20.

[47] S. Wu, the EAST Team, “An overview of the EAST project”, Fusion Engineering and Design **82** (2007) 463–471.

[48] What is ITER?, http://www.iter.org/

[49] Y. Wu, J. Jiang, M. Wang, M. Jin, Z. Chen, J. Liu, S. Liu, Y. Bai, H. Chen, Y. Wang, Y. Wan, FDS Team, "Re-Evaluation of Fusion-Fission Hybrid Reactors for Energy Production, Fuel Breeding and Waste Transmutation", Presentation at the 3rd IAEA Technical Meeting on "First Generation of Fusion Power Plants Design and Technology", July 13-15, 2009, Vienna, Austria.

[50] Y. Wu, J. Jiang, H. Chen, S. Liu, et. al., "The Fusion-Fission Hybrid Reactor for Energy Production: A Practical Path to Fusion Application", presented at the 22nd Int. Conf. on Fusion Energy (FEC-22) (Geneva, Switzerland, October, 2008) FT/P3-21, International Atomic Energy Agency, Vienna.

[51] D. S. Huang, Y. Wu, D. L. Chu, L. Hu，"Economic analysis of the fusion—driven subcritical system", Chinese Journal of Nuclear Science and Engineering **24** (2004) 184-192.

[52] L. L. Lao, H. St. John, R. D. Stambaugh, A. G. Kellman, W. Pfeiffer, "Reconstruction of Current Profile Parameters and Plasma Shapes in Tokamaks", Nuclear fusion **25** (1985)1611-1622.

[53] P. Dumaz, P. All`egre, C. Bassi, T. Cadiou, A. Conti, J.C. Garnier, Gas-cooled fast reactors—Status of CEA preliminary design studies, Nuclear Engineering and Design **237** (2007) 1618–1627.

[54] Y. Wu, J. Li, Y. Li, Q. Zeng, M. Chen, S, Zheng, et al., "An integrated multi-functional neutronics calculation and analysis code system: VisualBUS", Chinese journal of nuclear science and engineering **27** (2007) 365-373.

[55] Y. Wu, FDS Team, "CAD-based Interface Programs for Fusion Neutron Transport Simulation", Fusion Engineering and Design **84** (2009) 1987-1992.

[56] Yican Wu, Zhongsheng Xie, Ulrich Fischer, "A Discrete Ordinates Nodal Method for One-Dimensional Neutron Transport Numerical Calculation in Curvilinear Geometries", Nuclear Science and Engineering **133** (1999) 350-357.

[57] JIANG Jieqiong, XU Dezheng, ZHENG Shanliang, et al., Integral Data Benchmark of HENDL2.0/MG Compared with Neutronics Shielding Experiments[J], Plasma Science and Technology **11** (2009): 625-631.