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Section 16. Conceptual designs, programs and studies

The fusion-driven hybrid system and its material selection

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Abstract

The fusion-driven multi-functional hybrid reactor system (FDS) is proposed as a middle step toward the final application of fusion energy. The strategic goal and roadmap of the FDS is addressed along with its potential advantages. Selection analysis of required materials are performed as well as those of fusion driver concept, hybrid blanket concept for the FDS in China.

1. Introduction

The fusion–fission hybrid concept dates back to the earliest days of the fusion project when it was recognized that using fusion neutrons to breed nuclear fuel would vastly increase the energy from fusion plant. It appears to receive almost no attention since the mid 80's in the world, except in China who has given very serious consideration and has strong hybrid reactor activities [1].

Since 1986, the hybrid reactors have been given serious consideration in China as a national project supported in the framework of the High Technology Program ('863' Program) since it is predicted to need 1200–1500 GW_e power in 2050 with a population of 1.5 billions. China has strong fusion–fission hybrid studies activities [1–3]. Physics concepts studies of fusion–fission experimental hybrid reactor for the purpose of nuclear fuel production had started in 1980–1985, a detailed conceptual design had been finished in 1986–1995, an outlined engineering design has been performed in 1996–2000. Besides the fuel breeder design activities, a series of conceptual studies on long-lived radioactive waste transmutation using fusion neutron source have been done with the support of the National Natural Science Foundation (NNSF) [4–13]. Recently definition and design of the multi-functional fusion-driven sub-critical hybrid system (FDS), which could perform multi-functions such as breeding nuclear fuel, transmuting long-lived wastes, producing tritium for fusion fuel cycling etc. as an alternate strategy to utilize fusion energy technology based on the recent progress in hybrid reactor studies in China are being carried out under the support of the Chinese Academy of Sciences (CAS) [14–16].

The FDS is proposed as a middle step toward the final application of fusion energy considering available knowledge base of fusion technology and the energy demand in China. The strategic goal and roadmap of the FDS is addressed along with its potential advantages in Section 2. Selection analysis of required materials is performed in Section 5 as well as selection analyses of fusion driver concept, hybrid blanket concept for the FDS are presented in Sections 3 and 4, respectively.

2. Energy strategic goal

Although the recent experiments and associated theoretical studies of fusion energy development have proven the feasibility of fusion power, it is commonly realized that it needs hard work before pure fusion energy could commercially and economically utilized. On the other hand, the fission nuclear industry has been

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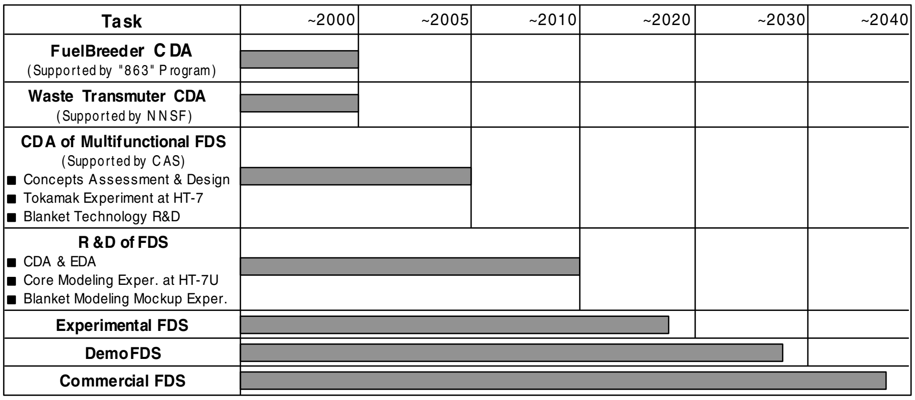

Fig. 1. Roadmap of FDS development in China.

falling on hard times recently since so far there has been no conclusion about how effectively to deal with the long-lived wastes produced from the nuclear spent fuel and about how to solve the shortage of natural uranium ore in addition to nuclear safety and proliferation. It's a natural way to develop fusion–fission hybrid reactors as an alternate strategy to speed up utilization of energy since a hybrid reactor can operate with lower fusion energy gain ratios Q, therefore the design of the fusion core for a hybrid system is easier than for a pure fusion reactor.

This kind of system has the following attractive features [14]:

(1) Having the fusion neutrons improved the overall neutron balance of the fission blanket system, thus enabling adequate excess neutrons available for breeding fissile fuel, transmuting long-lived fission products (LLFP) into short-lived radioactive nuclides (SLRN) or stable nuclides (SN) and minor actinides (MA) into fission products (FP).
(2) Having the fission blanket improved the overall energy balance of the fusion driver, thus easing the requirements from the plasma, subsequently from the materials of the first wall.
(3) Having no critical accident risk and reducing real proliferation dangers as compared to only considering critical fission systems.
(4) Benefiting fusion energy development by providing a test-bed for the development of fusion reactors and giving experience with a large-scale pure fusion device, while at the same time fulfilling a useful purpose of its own, which encourages continued work, and continued progress, toward the larger goal-pure fusion.
(5) Benefiting fission nuclear industry development by solving the long-lived waste disposal and fuel supply limitation.

To meet the energy demand of China in 2050 or so, at least 100 GW_e fission power stations would be built if a fraction of only 10% of total electricity capacity is assumed to come from nuclear power, which would consume a large amount of nuclear fuel and produce a large amount of high-level nuclear wastes [14].

The designed FDS should contain a D-T fusion core as the neutron driver, which could run based on continuous or long-pulse-based way, and a sub-critical hybrid blanket which could be used to transmute the long-lived nuclear wastes from spent fuel of fission power plants, to breed nuclear fuel to supply to fission reactors and to breed tritium for fusion fuel cycling of itself.

Conceptual design activities (CDA) and engineering design activities (EDA) for the experimental FDS and R&D of key components should be performed before 2010 in order to build an experimental reactor within the period of 2010–2020, to build a Demo reactor 2020–2030 and finally to build a commercial reactor 2030–2040. The history and proposed road map are shown in Fig. 1. On the basis of a series of CDA of fuel breeder and waste transmuter driven by fusion neutron source, the definition and CDA of multi-functional experimental FDS are being carried out under the support of CAS.

3. Fusion diver

A fusion core is used to provide a fusion neutron source to drive the sub-critical blanket. If an optimized blanket design would be adopted, the requirement for neutron source intensity and subsequently plasma technologies could be lowered. The studies have shown that the neutron intensities corresponding to the neutron wall loadings (0.2–1.0 MW/m^2) could meet the requirements from serious consideration of the FDS for long-lived waste transmutation or nuclear fuel breeding [2,9,13].

Substantial progress has been made in recent years in achieving the plasma conditions required for a fusion neutron driver. Tokamaks such as JET and TFTR have already generated about 10^{19} neutrons per second and fusion powers exceeding 10 MW (the corresponding neutron wall loading approached 0.2 MW/m^2) and Q_p just under unity in a short pulse using D-T fuel. The tokamak JT-60 has reached the conditions required for $Q_p > 1$ in deuterium plasmas. There now exists a knowledge base sufficient to design tokamak drivers that will achieve $Q_p \gg 1$ and neutron wall loadings $P_w \simeq 1.0$ MW/m^2 with high confidence. The remaining physics development required for the fusion drivers is to achieve much longer burn pulses or continuous operation and higher reliability and availability. A substantial advance in the database for long pulse operation will be supplied by the two superconducting tokamaks HT-7U in China (the designed maximum pulse length is 1000 s) [17] and KSTAR in Korea (the designed maximum pulse length 300 s) under construction using deuterium plasmas [18]. The internationally funded ITER-EDA (Engineering Design Activities) have been successfully performed and a cost-reduced design ITER-FEAT is being performed recently which would provide a prototype experiment which integrates the burning plasma physics and the advanced technology if it could be built [19]. In addition to conventional tokamaks which have a good database to design fusion drivers, other fusion concepts such as spherical tokamak, helical-type device etc. have their own attractive features as the drivers of the FDS and also made a good progress. For example, spherical tokamaks have many attractive advantages such as compact and relatively simple configuration and steady-state operation potential.

The engineering outlined design and R&D activities for the hybrid reactor FEB producing nuclear fuel have been finished recently [2]. The studies on the hybrid reactor transmuting nuclear wastes are being performed at ASIPP based on the standard (regular) tokamak driver concept and the compact (spherical) tokamak concept [3–13], respectively. The main design parameters of two fusion driver concepts based on a regular tokamak (RT) and a spherical tokamak (ST) are presented in Table 1 where the ones of HT-7U, ITER, ITER-FEAT and JET are also included for the purpose of comparison. To meet the requirement of a multi-functional experimental FDS, a fusion tokamak driver with a neutron rate of the magnitude order of 10^{19} n/s and a fusion power gain $\simeq 3$ is enough to achieve the goal of annual production of 100 kg ^{239}Pu based on the operation availability of 50%. Considering the progress in magnetic confinement fusion technology, a regular tokamak should be the most hopeful candidate of the first experimental FDS. The studies of fuel breeding and transmutation systems and other considerations would seem to indicate a tokamak with energy amplication $Q \simeq 3$, neutron wall loading $\simeq$ 0.5 MW/m^2, availability $\simeq$ 50%, very long pulse operation would be useful for realistic application. Meanwhile, a spherical tokamak is another attractive potential candidate since it offers the possibility of compact volumetric fusion neutron sources requiring relatively low-external fields [20,21]. Furthermore, this kind of compact tokamak might offer some attractive advantages such as a cost effective, high-performance (high-stable beta in the first stability boundary) plasma regime [22,23]. Recent progress in spherical tokamak experiments and associated theoretical studies as well as contribution to the mainstream tokamak program [24–28] provided added impetus to the verification of the physics in this regime and the assessment of its reactor prospects. In addition, helical reactors seem to have attractive advantages over tokamaks such as steady operation and no dangerous current disruption owing to inherently current-less

Table 1
Main parameters of HT-7U, FEB, FDT-ST, JET and ITER

Device or design	HT-7U	RT	ST		JET	ITER (ITER-FEAT)
			Option 1	Option 2		
Major radius R (m)	1.7	4.0	1.4	1.4	2.9	6.2
Minor radius a (m)	0.4	1.0	1	1	1.2	2.0
Plasma current I_p (MA)	1	5.7	9.2	7.0		15
Toroidal field B_t (T)	3.5	5.2	2.5	2.5		5.3
Average density $\langle n_e \rangle$ (10^{20} m^{-3})	$\sim$1	1.1	1.6	1.1		1.1
Average temperature $\langle T \rangle$ (keV)	3–5	10	10	9.5	10	10
Plasma volume (m^3)		134	50	50		837
Fusion power P_{fu} (MW)		143	100	50	16	500–700
Auxiliary power P_{aux} (MW)		50	28	19		73–100
Power gain Q		$\sim$3	$\sim$3	$\sim$2.5	$\sim$1	$\sim$10
Neutron wall loading P_w (MW/m^2)		0.43	1.0	0.5	$\sim$0.2	0.5–0.8
D-T neutron rate (10^{19} n/s)	3×10^{-4}	1.1	3.6	1.8	0.57	18–25
Operation availability (%)		50	80	80	–	$\sim$40

plasma. Final selection of fusion driver is one of the current key tasks for the FDS studies.

4. Hybrid blanket

The blanket system is one of the most important components of an FDS because it has a major impact on both the economics and safety of the hybrid system. The general idea of a hybrid FDS is to have the subcritical blanket which is to interact with a copious source of fusion neutrons provided by the fusion core.

The fissile or fertile materials are fissionable and are included in the blanket of FDS. Consequently, fission reactor technologies that have been developed should be applicable to the blanket design. On the other hand, an FDS has the fusion component. Fusion reactor technologies that are under development should be applicable to the blanket design, too.

Many detailed designs of blankets for pure fusion reactors have been performed, particularly, the engineering design and R&D of the internationally funded ITER has been finished. The technologies involved in the above concepts are also applicable to the blanket of hybrid FDS although additional attention should be paid on the fission related technologies.

Two blanket concepts for nuclear fuel production had previously been studied in China. In both the two concepts, the pebble bed structure was adopted for the complex tokamak geometry and Li_2O/liquid lithium and helium were chosen as tritium breeder and coolant, respectively. The inboard blanket is thin and the fuel breeding is limited to the outer board blanket only which minimizes the size of the reactor and enhances the reliability [2,3].

China has performed a lot of studies on transmutation of long-lived nuclides in the blankets of hybrid reactors besides on breeding nuclear fuel using hybrid reactors [15]. The studies have covered various blanket concepts. The studies have involved the thermal fission blanket concept with ^{239}Pu fissile neutron multiplication material for transmutation of FP [4], the hard neutron spectrum-based concepts for transmutation of actinides [5], the thermal neutron spectrum-based blanket concept for transmutation of actinides [5]. In all the studied concepts, fissile ^{239}Pu have been initially introduced into the blankets for neutron multiplication and energy balance adjustment. Thus the requirement for fusion driver technology could be much more easily satisfied, that is, the plasma core parameters and fusion technology requirements are far less stringent. The effective transmutation of long-lived waste nuclides could be achieved based on the requirement for relatively low-neutron wall loadings of 0.2–1.0 MW/m^2 which are between the levels of the one achieved in the tokamak device JET and the one in the ITER engineering design.

An attractive blanket for the FDS has at least the following features: (1) sub-criticality is deep enough to avoid the supercritical accident risk; (2) the fuels are self-sustainable with tritium for DT fusion reactions (the tritium breeding ratio $TBR > 1$) and Pu for fission neutron breeding (the Pu produced by the blanket approximately equal to (if only for waste transmutation) or more than (if also for fuel breeding) the Pu burned by the blanket); (3) the fuel cycle period is as long as possible to minimize the fuel and waste reprocessing; (4) the thermal power density is acceptable and the swing of thermal power output is as little as possible during the operation period; (5) the efficiency of waste transmutation is as high as possible.

The liquid $Li_{17}Pb_{83}$–He-gas dual-cooled blanket was previously studied for a pure fusion power reactor [29,30]. If this dual-cooled system is combined with a hybrid blanket, more potential advantages could be obtained. The helium gas and LiPb eutectic dual-cooled multi-functional blanket concept simultaneously for transmutation of MA, LLFP and Pu breeding is a very attractive blanket candidate due to its many attractive potential advantages [16]. The preliminary conceptual design showed that it is possible to reach the goals such as the 1-GW$_e$ year transmutation capability of $\simeq$175 kg MA produced by 5-GW$_e$ year PWRs approximately and 40–500 kg LLFP at least equivalent to the LLFP produced annually by the blanket itself when the initial total LLFP fraction is 2–70%, the swing of thermal power output less than 5% annually on the basis of feasible fusion physics and fusion technology level (the assumption of neutron wall loading 0.5 MW/m^2 and the operation availability 50plutonium can be self-sustainable. A preliminary definition of the LiPb–He gas dual-cooled, fuel-sustainable, multi-functional blanket for the FDS is presented in Table 2.

Table 2
Outline of LiPb–He dual-cooled multi-functional blanket

Functional zone	MA transmutation	FP transmutation	Pu breeding	Tritium breeding
Loaded fuel	LLMA	LLFP	U	LiPb
Products	FP + Energy	SLRN or SN	Pu	Tritium
Coolant	LiPb	He gas	LiPb	LiPb + He gas
Neutron multiplicator	Pu	–	–	LiPb
Neutron moderator	–	Graphite	Graphite	–

Further optimized design goal may be that the blanket could breed the amount of fissile plutonium to support 5-GW_e PWRs and transmute the amount of MA and FP from 5-GW_e PWRs as a DEMO FDS on the basis of tritium sustain. The number of PWRs may be reduced to one as an experimental FDS.

5. Material selection

Fusion reactor materials R&D for the FDS have been conducted in China for decades [31,32]. Considering the design requirements for the FDS and the situation of material research, the recommendations of materials for the FDS are given.

5.1. Structural materials

Austenitic stainless steels have been developed in the past years since they are mature and have widely been used in nuclear industry so that it might be the prime candidate for the first fusion-driven reactor. Considering the main purposes of developing FDS is not to output electricity, but to transmute nuclear wastes and to breed nuclear fuel, the operating temperature in the designed blanket may be less 400 °C, 316Ti stainless steel (15% cold worked, 316Ti SS-15%CW) may be used as the structural material for the first wall (FW) and the blanket system of the experimental FDS. 316Ti SS-(15–20)%CW has been developed in China [33–36]. Its irradiation effects have been explored in a wide temperature range and the results indicated that its void swelling behavior was much more improved in comparison with 316L SA (solid-solution annealed) at the same condition while the irradiation swelling resistance was much better than that of the latter. These results have also been proved in other laboratories [37–41]. For example, the irradiation lifetime (DPA at 5% swelling) of 316Ti SS-15%CW developed was 150 dpa at 400 °C. The 316Ti SS-15%CW meets the requirement of the compatibility with liquid $Li_{17}Pb_{83}$ and helium at 400 °C, the corrosion is less than 20 μm/y under the liquid velocity 1.5 m/s. The yield strength reaches saturation at the temperature range of 100–400 °C after 7 dpa and the total elongation remains at 2–3% at temperatures of 300–400 °C. The other characteristics such as most extensive database for nuclear applications, good fabrication/weldability, oxidation resistant and non-ferromagnetic material show the application of 316Ti SS-15%CW to be realistic. However, if the plasma arc-welding will be used for the 316Ti SS-15%CW, the performance of welding materials irradiated should be examined to meet the requirement of design.

Ferritic/martensitic (F/M) stainless steels are also candidates for structural application in FDS because of the properties such as swelling resistance, no helium brittleness, better thermal stress factor and better liquid-metal corrosion behavior than austenitic steel, with a substantial database. Its compatibility with liquid metals is much better than that of austenitic steels under the same conditions: F/M stainless steels can be used in the temperature range up to 550 °C in liquid lithium and 450 °C in liquid $Li_{17}Pb_{83}$, respectively [42], while the operation temperature limits of austenitic steels are 450 and 400 °C, respectively. So F/M stainless steels are especially suitable for liquid metal blankets. In addition, the neutron irradiation resistance of F/M stainless steels is much better than that of austenitic steels. A series of reduced activation ferritic/martensitic steels such as F82H (7.5Cr–2WVTa), JLF-1(9Cr–2WVTa), EUROFER (9Cr–1.1WVTa) have been actively pursued over the world [43]. A few kinds of F/M stainless steels have been developed in China [44] and reduced activation ones, e.g. Fe–CrWVTa alloys, are preferable because of their environmental attractiveness [45].

If the engineering design of the FDS e.g. in the DEMO phase needs to increase the operating temperature, the reduced activation martensitic steel is more suitable to use as the structural material of the FW and the blanket. The shift in the ductile-to-brittle transition temperature is the most critical effect for the martensitic steel. For example, the 9Cr–2WVTa steel exhibits a lower shift in the DBTT which is −56 °C at the damage dose of 28 dpa under 365 °C irradiation [46]. Normally the shift of the DBTT decreases with the irradiation temperature increase. It is necessary to irradiate this materials to reach 100 dpa at 450 °C and to test the DBTT increase. The other issues such as weld procedure requirement and magnetic load, still need to be investigated. Because the performance of martensitic steel is very sensitive to the heat treatment process, the performance of welding section material depends on the welding technique and the heat treatment after welding. The structural design still need to consider the effects of the large inductive current in the FW during the plasma disruption and the magnetic effects on the plasma and the stress of components.

The high-temperature strength of F/M stainless steels is lower than that of austenitic steels but the oxide dispersion strengthening (ODS) technique can increase their high-temperature strength even though the compatibility of ODS ferritic steels is not much better than that of F/M stainless steels. The ODS ferritic steel is an alternative choice for the FW and blanket structure, but the DBTT increase induced by irradiation should be improved in the future.

It is well known that clean nuclear power is very attractive and therefore some inherent low-activation materials have been developed for fusion application. In this meaning, advanced low-activation materials such as vanadium alloys and SiC/SiC composites could be considered for application in the commercial FDS because of their advanced properties and performance not only in mechanics, irradiation resistance, compatibility with liquid metal and low-activation characteristics. Unfortunately, many issues should be resolved before their application in fusion reactors. So, as a near-term application of fusion power, we prefer to consider the reduced activation structural materials in FDS rather than the inherent low-activation materials, e.g. V-based alloys and SiC/SiC composites. On the other hand, considering our ultimate objective, advanced pure fusion power application, we are also recommending to support a few limited research activities on low-activation materials R&D in China.

5.2. Tritium breeding materials and coating materials

The FDS blanket design prefers to adopt the lithium-lead eutectic alloy ($Li_{17}Pb_{83}$) as tritium breeding material because of attractive neutronics performance for a multi-functional hybrid blanket [11,16] besides its other attractive properties. The key feasibility issues in the design of a liquid metal self-cooled blanket include (1) liquid metal compatibility with structural materials; (2) the strong influence of the magnetic field on the liquid metal flow i.e. resulting MHD pressure drop; (3) tritium permeation. R&D studies concerning $Li_{17}Pb_{83}$ blankets suggest the use of coatings as insulators and tritium permeation barriers for the self-cooled blanket [37]. Al_2O_3 is the primary coating for application in a $Li_{17}Pb_{83}$ environment to decrease the MHD pressure drop and to prevent the tritium permeation. In order to obtain an acceptable pressure drop in the cooling system, generally the product of aluminide coating resistivity and the coating layer thickness with the hot-dipping process should be larger than a minimum resistivity value of 10^4 Ωm for a coating thickness of 1 μm, or 10^3 Ωm for a coating thickness of 10 μm [47]. But the major requirements for viable insulator coating are the chemical compatibility of the liquid metal with the structural metal, adequate electrical insulating characteristics, stability under irradiation environment and long-term stability (including self-healing) under thermal cycling conditions. The tritium permeation reduction factor (TPRF) of the tritium permeation barrier (TPB) coating in a $Li_{17}Pb_{83}$ blanket should be larger than 100 compared to the permeation rate for the bare channel wall, while the higher TPRF, e.g. >1000, would be required in the case of tritium gas permeation.

5.3. Nuclear fuel materials

Nuclear fuel (MA/Pu/U) materials in FDS may be developed based on the available fabrication technologies of the pebble bed fuel of the high-temperature gas-cooled reactor (HTGR) [48] and the metal alloy fuel of the liquid metal fast breeding reactor (LMFBR) [49]. Minor actinides (MA), Pu and U may exist in the form of oxide or metal alloy in the FDS blanket, which depends on physical considerations. For example, MA mixed with Pu could be made in appropriate small particles coated with multi-layers to form fuel balls [15]. The first layer is a buffer layer of low-density carbon to adjust the swelling and expansion of fuel and accommodate the fission recoils. The second layer is a dense graphite layer to prevent fission gas release the liquid coolant. The outer layer is SiC which should be very well compatible with liquid $Li_{17}Pb_{83}$. Because the density of (MA + Pu) oxide or metal alloy fuel is higher than that of liquid $Li_{17}Pb_{83}$ (9.43 g/cm^3), the density of fuel balls can be adjusted by the multi-layer coating and comparable to liquid eutectic LiPb, hence the small fuel balls can be uniformly distributed in the LiPb slurry.

5.4. Plasma facing materials

There is a good progress in plasma facing materials (PFMs) meeting the requirements of some fusion experimental devices over the world. For example, Low-*Z* PFMs, mainly carbon-based materials, have been developed and used in tokamak devices in China [50,51]. A multi-element (B-, Si- and Ti-)-doped graphite has been developed and its chemical sputtering (CS) rate is successfully decreased by a factor of 5 and its radiation enhanced sublimation (RES) is also improved in comparison with that of pure graphite [52]. The Development of multi-elements doped graphite with SiC (or tungsten) coating as PFMs should have a good thermal conductivity under irradiation and the factor of self-sputtering on the surface must be less than 1. The SiC coated multi-element doped graphite has successfully been applied as PFM in the HT-7 tokamak and it has been selected as PFM of the superconducting tokamak HT-7U under construction in China [17]. High-*Z* PFMs, e.g. Mo and W, have also been used in tokamaks in China [50]. Their higher threshold energy for sputter, lower erosion rate and good thermal conductivity are more attractive for the application in fusion reactors. Considering the lower activation property, W is preferable in FDS. It has to be noticed that W and current W-alloys have shown a few shortcomings, e.g. embrittlement at lower temperature, poor strength at high temperature and unsatisfactory neutron irradiation resistance, in many tokamak experiments. A new ODS W-alloy proposal has been put forward and it is expected to improve these shortcomings mentioned above. The application of functionally graded materials (FGMs) in PFM and high heat flux components (HFCs) has paid more attention in China. FGMs are better than coatings because of a good interface where the thermal stress much more decreased during thermal shocks. For the same reason, it is also a good joining technique between

PFM and heat sink. Some FGMs have been developed for this purpose in China [53].

6. Summary

Features of the FDS and its driver and blanket have been addressed to show the necessity and feasibility of developing FDS as the next step towards pure fusion energy to help development of fission and to encourage continued progress to the final goal in China. The parameter levels e.g. neutron wall load $\simeq 0.5$ MW/m^2 with the availability of 50% of the fusion driver and the multi-functional blanket are proposed. Preliminary selection analyses were carried out for the materials including structural material, tritium breeding material, nuclear fuel material, coating material and PFM in the FDS. The 316Ti stainless steel is considered as the prime candidate of the FW and structure materials of the experimental FDS blanket. The reduced activation martensitic steel is recommended as an alternate candidate of the structural material of the FW/blanket system although the DBTT shift at high-damage dose ($\simeq$100 dpa) and the welding technique should be further improved. Al_2O_3 coating is selected as insulator and tritium permeation barrier for the LiPb self-cooled blanket. The fuel fabrication of (MA + Pu) oxide or metal alloys may be developed on the basis of the available fabrication techniques developed for the pebble bed fuel of the HTGR and the metal alloy fuel of the LMFBR. The PFMs could be selected based on the progress already and in future made in the tokamak experimental devices.

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