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聚变堆设计与核技术

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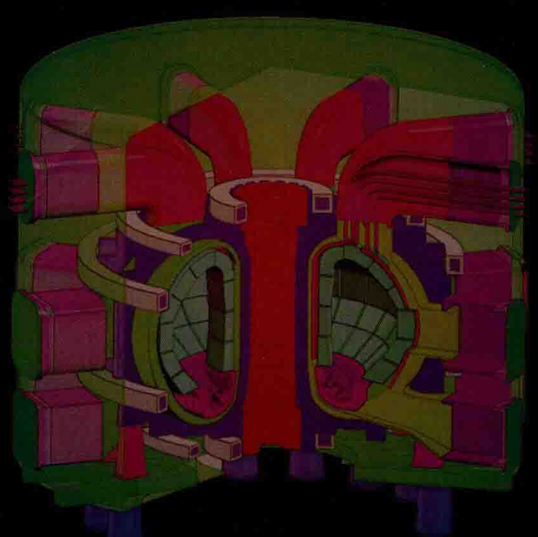

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前　言

聚变能是一种潜在的用之不竭的清洁能源，开展聚变堆设计及相关核技术研发是最终实现聚变能应用的关键环节。随着国际热核实验堆(ITER)计划的实施，聚变堆设计及相关的中子学、材料与包层等核技术研究日益受到重视并取得了重要进展。

《聚变堆设计与核技术》收录了中国科学院核能安全技术研究所·FDS团队在聚变堆设计与核技术方面的部分研究成果。全书分为两篇，上篇侧重介绍系统设计，内容含聚变堆设计、聚变裂变混合堆设计、液态铅锂实验包层(ITER-TBM)系统设计以及设计工具与数据库等，下篇侧重介绍氘氚聚变中子源、抗辐照结构材料、冷却剂和增殖剂等聚变核技术研究进展。

中国科学院核能安全技术研究所是面向核能与核技术应用安全相关领域开展基础性、前瞻性、战略性研究的创新型研究所。多年来，在聚变堆设计与核技术领域已取得诸多重要进展：发展了系列聚变堆和聚变裂变混合堆新概念，设计建造了强流氘氚聚变中子源 HINEG，中子源强在运国际第一；成功研发了中国抗辐照低活化结构钢 CLAM，已成为国际三大低活化钢之一；建成了世界首座多功能液态铅锂包层技术综合实验平台 DRAGON 等等。本书将这些方面研究成果汇集成册，以供广大聚变能研究人员及感兴趣的学者们参考。

由于聚变研究涉及诸多学科，有许多问题待进一步探索，书中存在的不足之处，还请各位专家、读者不吝指正。

谨以此书纪念中国科学院核能安全技术研究所成立五周年。

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一、综　述

Conceptual Design Activities of FDS Series Fusion Power Plants in China

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Abstract: A series of fusion power plants (named FDS series) have been designed and assessed for the examination of the feasibility and the safety, environmental and economical potential of fusion with emphasizing the blanket design optimization on neutronics, thermal-hydraulics, electro-magnetics, material, structural performance analyses in China. Four concepts have been developing, which are the fusion-driven subcritical system (FDS-Ⅰ) with the goal of transmutation of the long-lived nuclear wastes and breeding of fissile nuclear fuels, the fusion power reactor (FDS-Ⅱ) with the goal of electricity generation, the fusion-based hydrogen production reactor (FDS-Ⅲ) and the spherical Tokamak-based compact reactor (FDS-ST). Four types of blankets, including the He/PbLi dual-cooled high level waste transmutation (DWT) blanket, the He single-cooled PbLi tritium breeder (SLL) blanket and the He/PbLi dual-cooled (DLL) blanket, the high temperature liquid PbLi (HTL) blanket, and Test Blanket Modules (TBMs) and the testing strategy have been studied. R&D status on materials, design and analysis tools are also presented.
Keywords: Fusion power plant; Liquid lead lithium blanket; Test Blanket Module; Conceptual design

1 Introduction

A series of fusion power plants (named FDS series) have been designed and assessed for the examination of the feasibility and the safety, environmental and economical potential of fusion with emphasizing blanket design optimization on neutronics, thermal-hydraulics, electro-magnetics, material, structural analyses by the Fusion Design Study (FDS) Team in China. Up to now, four concepts have been developing, which are the fusion-driven subcritical system (named FDS-Ⅰ), the fusion electrical generation reactor (named FDS-Ⅱ), the fusion-based hydrogen production reactor (named FDS-Ⅲ) and the spherical Tokamak-based compact reactor (named FDS-ST).

The purposes of the FDS-Ⅰ and FDS-ST designs are to exploit the possibility of earlier application of fusion energy as volumetric neutron sources. FDS-Ⅱ is a fusion reactor to obtain electrical power based on the technology conservatively extrapolated from ITER and FDS-Ⅰ or FDS-ST. FDS-Ⅲ is designated to assess the potential feasibility and attractiveness of non-electrical application of fusion. Test Blanket Modules (TBMs) for the International Thermonuclear Experimental Reactor (ITER) and the Experimental Advanced Superconducting Tokamak (EAST) and the testing strategy also have been studied to assess the feasibility of technology development.

In this contribution, an overview of design activities of FDS series reactors is presented, including liquid lead-lithium blankets and relevant TBMs, as well as R&D status on materials, design and analysis tools.

2 FDS-Ⅰ: a fusion-driven subcritcal system

The fusion-driven subcritical system, in which a fusion core is used as a neutron source to drive the subcritical blanket, has very attractive advantages, which ease the requirements for the plasma and enable adequate excess neutrons available for breeding fissile fuels, transmuting long-lived fission products and actinides. In addition, there are no risk of critical accident and less danger to nuclear proliferation, as compared to the critical fission systems[1]. If an optimized blanket design was adopted, the requirement for neutron source intensity and subsequently fusion plasma technologies could be lowered. FDS-Ⅰ, a fusion-driven subcritical reactor, is designated to transmute the long-lived nuclear wastes from fission power plants and to produce fissile nuclear fuels for feeding fission power plants as an intermediate step and early application towards final application of fusion energy on the basis of easily-achieved plasma physics and engineering technology, where the He-gas/liquid PbLi dual-cooled high level waste transmutation (DWT) blanket concept is adopted[1-4]. A set of plasma-related parameters of FDS-Ⅰ is given in Table 1. More details on design optimization of fusion plasma core are being carried out.

The FDS-Ⅰ blanket design focuses on the technology feasibility and concept attractiveness to meet the requirement for fuel sustainability, safety margin and operation economy. A design and its analysis on the DWT blanket with Carbide heavy nuclide Particle fuel in circulating Liquid PbLi coolant (named DWT-CPL) has been studied for years. Other concepts such as the DWT blanket with Oxide heavy nuclide Pepper pebble bed fuel in circulating helium-Gas (named DWT-OPG) and with Nitride heavy nuclide Particle fuel in circulating helium-Gas (named DWT-NPG) are also being investigated.

Table 1 The reference plasma-related parameters of FDS series designs

Parameters	Design				
	FDS-Ⅰ	FDS-Ⅱ	FDS-ST	EAST	ITER
Fusion power/MW	150	2 500	100	D-D	500
Major radius/m	4	6	1.4	1.95	6.2
Minor radius/m	1	2	1.0	0.46	2
Aspect ratio	4	3	1.4	4.2	3.1
Plasma elongation	1.78	1.9	2.5	1.8	1.70
Triangularity	0.4	0.6	0.45	0.45	0.33
Plasma current/MA	6.3	15	9.2	1.5	15
Toroidal-field on axis/T	6.1	5.9	2.5	3.5～4.0	5.3
Safety factor, q_{-95}	3.5	5.0	5.5	—	3
Auxiliary power/padd/MW	50	80	19	—	73
Energy multiplication, Q	3	31	5	—	⩾10
Average neutron wall load/(MW · m^{-2})	0.5	2.72	1.0	—	0.57
Average surface heat load/(MW · m^{-2})	0.1	0.54	0.2	0.1～0.2	0.27
Normalized beta, β_N/%	3	5	3	—	2.5

For the DWT-CPL blanket concept, helium gas was adopted to cool the structural walls and long-lived fission product (FP) transmutation zones (FP-zones), liquid metal (LM) PbLi eutectic with tiny particle long-lived fuel to self-cool Actinide (AC) zones (AC-zones) including Minor Actinides (MA) transmutation zones (MA-zones) and Uranium-loaded fissile breeding zones (U-zones). U-zones may be replaced with AC-zones if fertile-free concept is considered. The details on this design can be found in Ref. [2-4].

The DWT-OPG and DWT-NPG blanket concepts are both based on the thermal neutron transmutation concept, in which the helium gas is adopted to cool the structural walls, FP-zones and AC-zones. Compared to the DWT-CPL concept, the relatively low-speed PbLi flow in the DWT-OPG blanket will reduce the MHD effect. In the DWT-NPG blanket, the tiny coated fuel particles suspended in the pressure helium gas will enable the high efficient heat removal compared to the DWT-OPG concept, so the system can operate in the higher power density level than the other two concepts.

3 FDS-Ⅱ: a fusion power reactor for electricity generation

FDS-Ⅱ is designated to exploit and evaluate potential attractiveness of pure fusion energy application, i. e., obtaining a high-grade heat for generation of electricity on the basis of conservatively advanced plasma parameters, which can be limitedly extrapolated from the successful operation of ITER.

The plasma physics and engineering parameters of FDS-Ⅱ are selected on the basis of considering the progress in recent experiments and associated theoretical studies of magnetic confinement fusion plasma (as in Table 1). It is understandable that the FDS-

Ⅱ requirement for plasma technology could be met by the development of ITER and/or FDS-Ⅰ. More details on design optimization of fusion plasma core are being carried out.

Both the feasibility and attractiveness of technology are of concern to the FDS-Ⅱ blanket design, which must meet the requirement for tritium self-sufficiency, safety margin, operation economy and environment protection, etc. Two optional concepts of liquid PbLi blankets including the RAFM steel-structured He-cooled PbLi tritium breeder (SLL) blanket and the RAFM steel-structured He-gas/liquid PbLi dual-cooled (DLL) blanket are adopted for FDS-Ⅱ.

For the DLL design, He gas is used to cool the first wall and blanket structure and liquid PbLi is to be the self-cooled tritium breeder with a high outlet temperature up to 700 ℃ in order to achieve high thermal efficiency. The Flow channel inserts (FCIs), e. g., SiC_f/SiC composite or other refractory materials, are designed and used inside the PbLi coolant channel and manifold, which act both as thermal and electrical insulators to keep the temperature of RAFM structure below the maximum allowable temperature. Coating (e. g., Al_2O_3) on RAFM structure is also considered in the design to reduce tritium permeation and prevent corrosion of PbLi.

The SLL concept is another option of FDS-Ⅱ blanket considering the SLL blanket could be developed relatively easily with lower PbLi outlet temperature and slower PbLi flow velocity and that it allows the utilization of relatively mature material technology. Coating is probably needed to protect the structure and to reduce tritium permeation and also MHD effects. The details on this design can be found in Ref. [5].

4 FDS-Ⅲ: a high temperature fusion reactor for hydrogen production

FDS-Ⅲ aims to obtain the high temperature heat in the blanket of fusion reactor for efficient production of hydrogen using thermo-chemical iodine-sulphur cycles technology based on the current status or promising extrapolation of material technology. This innovative blanket design with "multi-layer flow channel inserts (MFCIs)" is considered to obtain high temperature heat while using the relatively mature and most promising RAFM steel (allowed temperature up to 550 ℃) as structural material, refractory material with low thermal conductivity, such as SiC_f/SiC composite material or other components as the functional material inserted in the flow channel. Low temperature PbLi flows into the channel, then meanders through the multi-layer flow channel inserts. The temperature of the coolant PbLi is improved step by step, at last it is exported from the blanket in the high temperature above 900 ℃.

The details on the blanket design can be found in Ref. [6].

5 FDS-ST: a spherical Tokamak-based subcritcal system

The FDS-ST studies are undertaken to investigate the potential advantages of the low aspect ratio Tokamaks (i. e. , spherical Tokamak-ST)[7].

Theoretical and experimental studies indicated that the performance of Tokamak plasma is substantially improved with decreasing aspect ratio. Low aspect ratio (<2) Tokamaks can potentially provide a high ratio of plasma pressure to magnetic pressure β and high plasma current I at a modest size. The plasma β_T in a ST device can be high so that resistive toroidal-field can be small in order that the manageable Joule losses in TF coils can be achieved. This eliminates the need for a thick, inboard shield for cryogenic toroidal-field coil, so fusion devices with smaller major radius are possible. However, the elimination of inboard blanket needs the introduction of Center Conductor Post (CCP) in the limited space, which is a great challenge for ST because of the high fields and large forces on it.

The outboard can be designed as a subcritical system with a high multiplication of energy due to fission reaction in order to achieve the highly economical operation. This can compensate the large fraction of recirculating power in a ST, mitigate the requirement for the neutron wall loading and thus reduce the irradiation on the first wall (FW).

The CCP in ST reactor will stand severe neutron irradiation and receive high nuclear heating power. Consequently, it is needed to be replaced after a certain years' operation. Four CCP concepts[8-11], i. e. , water-cooled copper (water-Cu), liquid Li self-cooled (Li-SC), water-cooled Li (water-Li) and liquid metal-blanketed copper (LM-Cu) CCPs have been investigated.

6 TBMs and development strategy

To check and validate the feasibility of the China liquid PbLi blankets, the Dual-Functional Lead Lithium-Test Blanket Module (DFLL-TBM) system, which is designated to demonstrate the integrated technologies of both He single coolant (SLL) blanket and He-PbLi dual coolant (DLL) blanket, is proposed for test in ITER[12]. The DFLL design allows the strategy of earlier test of SLL-TBM, evolving to later test of DLL-TBM after the issues on FCIs and MHD effects can be solved. That means that two types of TBMs (i. e. , SLL-TBM and DLL-TBM) are to be tested in ITER with as similar as possible basic structure and auxiliary system except for including FCIs and quicker flowing PbLi in DLL-TBM. The integrated test and validation of the remaining critical issues related to the DWT blanket and the HTL blanket can be conducted after the ITER successful operation.

The TBMs development strategy covers three-phases, i. e. (1) materials R&D and out-of-pile blanket mockup (e. g. , 1/5 size-reduced) test in He and PbLi loops will be proposed to concern material development technologies, the TBM fabrication techniques, the thermo-mechanical/thermo-fluid dynamic performances, the compatibility between flowing PbLi and structural steel, the MHD effects of flowing PbLi, etc. (2) The He-