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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/LiPb dual-cooled high level waste transmutation (DWT) blanket, the He single-cooled LiPb tritium breeder (SLL) blanket and the He/LiPb dual-cooled (DLL) blanket, the high temperature liquid LiPb (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 lithium lead 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

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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 lithium–lead 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 LiPb 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 LiPb 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.

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) LiPb 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].

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	2500	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

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 LiPb 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 LiPb blankets including the RAFM steel-structured He-cooled LiPb tritium breeder (SLL) blanket and the RAFM steel-structured He-gas/liquid LiPb 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 LiPb is to be the self-cooled tritium breeder with a high outlet temperature up to 700 °C 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 LiPb 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 LiPb.

The SLL concept is another option of FDS-II blanket considering the SLL blanket could be developed relatively easily with lower LiPb outlet temperature and slower LiPb 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 °C) 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 LiPb flows into the channel, then meanders through the multi-layer flow channel inserts. The temperature of the coolant LiPb is improved step by step, at last it is exported from the blanket in the high temperature above 900 °C.

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 LiPb blankets, the Dual-Functional Lithium Lead–Test Blanket Module (DFLL–TBM) system, which is designated to demonstrate the integrated technologies of both He single coolant (SLL) blanket and He–LiPb 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 LiPb 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 LiPb loops will be proposed to concern material development technologies, the TBM fabrication techniques, the thermo-mechanical/thermo-fluid dynamic performances, the compatibility between flowing LiPb and structural steel, the MHD effects of flowing LiPb, etc. (2) The He–LiPb dual-coolant loop and relevant testing facility is to be constructed to test the middle-scale (1/3 size-reduced) TBM in the EAST super-conducting tokamak (the goal parameters are listed in Table 1 for comparison with ITER operation conditions) in order to validate the design tools and codes for electro-magnetics, thermo-mechanics and partially neutronics, to assess the TBM influence on plasma performances, as well as to demonstrate the feasibility and availability of DFLL–TBM auxiliary system design before the DFLL–TBM system is installed in ITER. (3) The full-scale consecutive TBMs are to be tested on the different operation phases of ITER. Therefore, the 'act-alike' DFLL–TBMs are designed in turn as EM-TBM for test of EM effects, as NT-TBM for performance of neutronics, as TT-TBM for thermo-mechanics and tritium behavior, and as IN-TBM for integrated performance test, respectively. This program will allow to consecutively validate SLL/DLL blanket concepts, technologies and design tools with reliable and safe operation, and finally to demonstrate relevant technologies for the fusion DEMO reactors.

7. Development of design and analysis tools

To carry out the design activities of various fusion related concepts, a series of design and analysis tools have been developed, which cover the areas of neutronics, thermal-hydraulics and MHD effects, thermo-mechanics, economics, safety and risk analysis, etc., and their coupling analysis. Three of the integrated code systems are VisualBUS, NTC and TOPCODE.

7.1. VisualBUS/HENDL: a multi-functional neutronics analysis code system

A multi-functional neutronics analysis code system (named VisualBUS) has been developed by integrating and improving existing codes. Transport calculation, burnup calculation, activation calculation and thermal-hydraulics calculation can be coupled or streamed together to run in a batch way or interactively started, monitored and controlled by the user with the help of Graphical User Interface (GUI). Particle transport can be simulated by using either the Monte Carlo (MC) method or the discrete ordinates (SN) method on the basis of the mutli-dimensional geometry models. The burnup and decay chain equations of isotopes can be solved with the Bateman or Runge-Kutta method for burnup calculation, as well as with the Bateman method for activation calculation. Moreover, Genetic Algorithms (GAs) and Artificial Neural Network (ANN) can be optionally used to optimize the nuclear design parameters. Monte Carlo Automatic Modeling system (MCAM) and SN Automatic Modeling system (SNAM) are two of the functional codes of the integrated modeling environment for Monte Carlo and discrete coordinates particle transport simulation codes and Hybrid Evaluated Nuclear Data Library (HENDL) has been developed to provide nuclear database.

As an interface code between commercial CAD softwares and MCNP code, MCAM supports various neutral CAD file formats such as STEP or IGES, it can convert large complex three dimensional CAD models into the format of MCNP input files and vice versa. Besides the conversion functions and basic modeling ability, MCAM is also a fully featured visualization tool and property editor for MCNP models. It has successfully passed the ITER model benchmark conversion and reverse conversion. This means MCAM can work as an efficient productivity tool in nuclear analysis fields.

Similar to MCAM, the code SNAM is under development to provide a useful tool for the users of SN multi-dimensional neutron transport codes such as VisualBUS, DOORS, DANTSYS, etc. SNAM cannot only meet the need for time-consuming combinational geometry creation and verification, but also can be used to visualize the calculational results from SN codes.

HENDL is a compilation of nuclear data selected from the various national and international evaluated nuclear data files. It includes several working libraries, e.g., the multi-group library HENDL/MG for the SN transport calculation and continuous point-wise library HENDL/MC for the MC transport calculation as well as those for burnup and activation calculations. Some special purpose working libraries, e.g., for self-shielding effects analysis are also custom-tailored. A series of data test analyses have been performed to validate and qualify the HENDL working libraries.

7.2. NTC: a neutronics/thermal-hydraulics coupling code system for transient safety analysis

NTC is a computer program to predict the coupled neutronics and thermal-dynamics behavior of advanced reactors during the transients of Design Basic Accident or Beyond Design Basic Accident. The modeling philosophy is based on the use of general physics to represent interactions of accident phenomena and regimes rather than a detailed representation of specialized situation.

The reactor neutronic behavior is predicted by solving a space-time and energy-dependent neutron conservation equations (discrete ordinate transport). The neutronics and the thermal-dynamics are coupled via feedback of the temperature, the density and the nuclear power distribution. The temperature- and background-dependent cross sections are determined using the Bondarenko formalism. The thermal-dynamics calculations are carried out by solving multicomponent, multiphase, multifield equations for mass, momentum and energy conservation. Magneto-Hydrodynamics (MHD) calculations, where the flows of liquid metal are influenced by strong magnetic field in fusion reactor, are integrated into the NTC code by adding the Lorentz source item in the conservation equations. The development and benchmark of transient safety analysis code NTC is underway.

7.3. TOPCODE: an integrated code system for risk-benefit-cost analysis and system Optimization

A goal of integrated system analysis on fusion systems is to exploit and demonstrate its attractiveness from the viewpoint of energy economics (cost-and-benefit), safety and environmental impact, which can help to improve and optimize the design. Development of an integrated code system named TOPCODE is underway, which is a coupling tool of the fusion system optimization and economics analysis code SYSCODE (System Analysis Code) and the Probabilistic Safety Assessment code RiskA (Risk Analysis Code).

Economics assessment of power plant includes calculation of internal costs and external costs. The detailed cost-and-benefit models not only for pure fusion systems but also for hybrid fusion–fission systems have been developed. Related calculation (e.g., COE and BOE) has been performed by SYSCODE. SYSCODE also have other functions, such as optimization of design, uncertainty and sensitivity analyses, etc.

RiskA has been developed for an advanced general-service tool for PSA (Probabilistic Safety Assessment), including fault tree and event tree analysis, importance analysis, common cause failure analysis, human failure analysis, uncertainty and sensitivity analysis, etc. Development of the fusion-oriented version of RiskA is underway.

8. R&D on materials

The Reduced Activation Ferritic/Martensitic steels (RAFMs) are currently considered as the primary candidates of structural materials for the DEMO fusion plant and the first fusion power reactors because of their attractive properties. Development of the Chinese version of RAFMs, i.e., China Low Activation Martensitic steel (CLAM) was initiated several years ago based on the progress in the other RAFMs, such as EUROFER97, F82H, JLF-1, 9Cr-2WVTa, etc., which were widely studied in the world. A series of R&D activities on CLAM and related technology are being carried out, including melting, controlling of the chemical compositions, property tests, techniques for joining and coating, corrosion properties with liquid LiPb and irradiation effect by plasma, etc. Preliminary tests show CLAM has good properties before irradiation. Besides, simulation and analysis on activation characteristics of CLAM and contributions of impurities to total dose rate have been done. A database for nuclear materials management software FUMDS has been developed. Optimization of composition, property tests after irradiation, related work for large ingots, etc., are underway. The details on R&D status of fusion reactor materials in China can be found in Ref. [13].

9. Summary

Four fusion reactor concepts and four types of liquid LiPb blanket concepts have been developed and assessed in China. DFLL–TBM system and its three-phases-strategy of development have been proposed. An overview of the series of activities on conceptual design and related R&D of materials, design and analysis tools have been given in this contribution.

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