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 and structural and safety 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 waste and breeding of fissile nuclear fuel, the fusion power reactor (FDS-II) with a goal of electricity generation, the fusion-based hydrogen production reactor (FDS-III) and the spherical tomakak-based compact reactor (FDS-ST). Four types of blankets, including the He gas/liquid lithium-lead dual-cooled high level waste transmutation (DWT) blanket, the He-cooled LiPb tritium breeder (SLL) blanket and the He/LiPb dual-cooled (DLL) blanket, the high temperature liquid lithium-lead (HTL) blanket probably considering SiC_f/SiC composite or other refractory materials as the insulator, and TBMs (test blanket modules) scenarios and the test strategy have been studied. In this contribution, a design overview of FDS series reactors including lithium-lead blankets and relevant TBMs are presented, as well as a proposed roadmap of fusion energy development in China.

0. 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, thermalhydraulics, electro-magnetics, material and structural analyses by the FDS (Fusion Design Study) 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 tomakak-based compact reactor (named FDS-ST).

The purpose 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 conversional 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. TBMs(Test Blanket Modules) for ITER (the International Thermonuclear Experimental Reactor)[1] and EAST (the Experimental Advanced Superconducting Tokamak) 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, including liquid lithium-lead blankets and relevant TBMs and the design and analysis tools development are presented as well as a proposed roadmap of fusion energy development in China.

1. FDS-I: a Fusion-Driven Subcritcal System

FDS-I, a fusion-fission hybrid reactor, is designated to transmute the long-lived nuclear wastes from fission power plants and to produce fissile nuclear fuel 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 RAFM (Reduced Activation Ferritic/Martensitic) steel or optionally the austenitic stainless steel-structured He-gas/liquid LiPb dual-cooled high level waste transmutation (DWT) blanket concept is adopted [2-6].

1.1 Necessity of Sub-critical System Development

Although the recent studies and experiments of fusion energy development have proven the feasibility of fusion power, more work still needs to be done before pure fusion energy could be finally utilized commercially and economically. On the other hand, the fission nuclear industry has been falling on hard times recently in the world since so far there has been no conclusion about how 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 is a natural way to develop fusion-fission hybrid reactors including fuel producing reactors and waste transmuting reactors as an alternate strategy to speed up the time for producing energy since a hybrid reactor as a subcritical system can operate with lower fusion energy gain ratios Q and the design of the fusion core for a hybrid system is easier than for a pure fusion reactor. This philosophy should be true particularly in China**[1,2]**, at least considering the two points i.e. the energy demand in China and the features of a hybrid system, which are addressed in the following two sub-sections. To avoid confusing the conventional fusion-fission hybrid reactor only for fuel production purpose or energy production, which will operate with high plasma parameters like a normal fusion power reactor, and the multi-functional hybrid system, which may operate with low plasma parameters as a volumetric fusion energy source, also for waste incineration and even a testing bed for fusion energy technology development, the expression "fusion-driven subcrtical system" has been adopted[3,4].

(1) Energy demand in China

Nuclear energy is safe, clean, economical and can be a substitute of fossil fuels on a large scale. Nuclear development in China is important in restructuring the coal-dominating energy mix, mitigating environmental pollution induced by coal combustion and supplementing power supply in costal areas with power demand growing rapidly. The nuclear power scale in China is targeted to reach 40 GW by 2020 on the basis of 7 GW in 2004, accounting for about 4% of the total installed capacity 1000GW of electric power, and 240GW, which is equivalent to 68% of the overall installed capacity of the world at present, by 2050 if it would conservatively be assumed to be 16% of the total installed capacity 1500GW of electric power, the average level of the world at present. This situation will certainly need to solve the challenges for a large amount of fissile fuel supply and long-lived nuclear waste disposal, which may be solved by using the proposed sub-critical systems.

(2) Features of a subcritical system

The fusion-driven subcritical system has very attractive advantages in principle because of its potential ability to achieve effective transmutation of long-lived radioactive wastes from spent fuels of fission industry, efficient breeding of fissile fuel in order to supply fuel for fission industry and other near-term fusion applications based on feasible fusion plasma physics and technology. The advantages can be summarized as follows[2]:

(1) The fusion neutrons improve the overall neutron balance of the fission blanket system, thus enabling adequate excess neutrons available for breeding fissile fuel, transmuting long-lived minor actinides and fission products.

(2) The fission blanket improve the overall energy balance of the fusion driver, thus easing the requirements for the plasma, subsequently for the materials of the first wall, and enabling an earlier contribution of the fusion research and development program to the energy economy.

(3) There are no risk of critical accident and less danger to nuclear proliferation as compared to the critical fission systems.

(4) It may provide a test-bed for the development of fusion reactors.

Obviously, the development of the subcritical system with multi-purposes is beneficial to both fission nuclear industry and fusion energy development. On the other hand, the fusion-driven subcritical system as a kind of advanced nuclear system has its own uncertainty of technology because it partly integrates the complexity of a fusion system related to plasma technology and a fission system related to high power density component with fission materials.

In the current studies, two kinds of fusion drivers, i.e. a normal tokamak based system (FDS-I) and a spherical tokamak based system (FDS-ST), are considered and evaluated.

1.2 Plasma Core

The major objective of FDS-I is to demonstrate the feasibility of early application of fusion energy technology. A fusion core is used as a neutron source to drive the subcritical blanket. If an optimized blanket design was adopted, the requirement for neutron source intensity and subsequently plasma technologies could be lowered. The plasma physics and engineering parameters of FDS-I are selected on the basis of the progress in recent experiments and associated theoretical studies of magnetic confinement fusion plasma and the progress in studies of blanket concepts optimization to reduce the requirement for neutron source intensity and subsequently plasma technologies. A set of plasma-related parameters of FDS-I is given in **Table 1.** Those of ITER and other FDS series designs are also shown in the table for the purpose of comparison. It is understandable that the FDS-I requirement for plasma technology could be met by the development of ITER. More details on design optimization of fusion plasma core are being carried out.

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

1.3 Blanket

The blanket system is one of the most important components of FDS-I because it has a major impact on both the economics and safety of the hybrid system. The general idea of a fusion-fission subcritical system is to have the subcritical blanket which is to interact with a copious source of fusion neutrons provided by the fusion core to achieve its multi-functions such as nuclear waste transmutation, fissile fuel and tritium breeding. 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 series of design scenarios, with emphasis on circulating particle or pebble bed fuel configurations considering geometry complexity of tokamak, frequency of fuel discharge and reload (including design of an emergency fuel discharge sub-system to improve the safety potential of the system), are being evaluated and optimized. A design and its analysis on the helium-gas and liquid LiPb eutectic Dual-cooled Waste Transmutation (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.

(1) DWT-CPL

For the DWT-CPL blanket concept, helium gas was adopted to cool the structural walls and long-lived fission product (FP:^{99}Tc, ^{129}I, ^{135}Cs) transmutation zones(FP-zones), Liquid Metal (LM) LiPb eutectic with tiny particle long-lived fuel to self-cool Actinide (AC: MA, Pu, U etc.) zones (AC-zones) including Minor Actinides (MA: ^{237}Np, ^{241}Am, ^{243}Am, ^{244}Cm) 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. LiPb in AC-zones serves as coolant, tritium breeder, neutron multiplier and fuel circulating carrier. High energy neutrons from D-T fusion reactions and AC fission reactions are moderated in FP-zones with graphite.

TRISO(Tri-ISOtropic)-like coated AC carbide particle suspended in the LiPb slurry is considered as one of the options in the design of AC fuel form referring to the maturity of HTGR (High Temperature Gas cooled Reactor) fuel fabrication. The circulating fuel particles have the advantages of good compatibility with complex geometry, easy control of fuel assembling or unloading and fast response to emergency discharge etc. The carbide fuel is preferable for its good neutron economy and superior thermal conductivity. Silicon carbide (SiC) as the peripheral cladding is considered as the most compatible material with LiPb currently. The moderator-coated LLFP and graphite pebbles are loaded in the separated FP zone respectively. According to the different capture cross sections of three isotopes ($\sigma_{Tc^{99}} > \sigma_{I^{129}} > \sigma_{Cs^{135}}$), CsCl, NaI and Tc are distributed in radial direction respectively considering the comprehensively consistent transmutation. The LLFP pebble fuel cooled by helium gas has the advantages of online refueling and fast response to emergency fuel removal.

The details on this design can be found in Ref.[4,5,6]

(2) DWT-OPG

The DWT blanket with Oxide heavy nuclide Pebble bed fuel cooled in circulating helium-Gas (named DWT-OPG) is based on the thermal neutron transmutation concept, in which the helium gas is adopted to cool the structural walls, FP zones and AC zones. LiPb in the LM zones is only uesd as tritium breeder. Compared to the dual coolant concept in the DWT-CPL, the single coolant concept will not only simplify the flowing channel designs but also bring good safety performance by eliminating the influence of coolant void coefficient. Meanwhile, the relatively low-speed LiPb flow will reduce the MHD effect in the blanket..

Considering the fuel pebbles need to be loaded homogeneously in the complex geometry of blanket, the proper pebble size and spatial distribution must be optimized not only to satisfy the efficient nuclear heat removal but also to provide enough space for helium flowing arrangements. Unlike the high helium outlet temperature in the HTGR (>750°C), the heat removal capability will limit the power density level due to the steel structure instead of ceramic container. The optimized gas cooling system is needed to improve the power density limit in the DWT-OPG.

The experiences of well-known HTGR technology including fuel fabrication and pebble bed technology will completely applied in this design. Oxide fuel is preferable for its mature application in the HTGR fabrication. The most generic concept for DWT-OPG in terms of refueling is the online, multi-recirculating feeding system in which pebbles are continuously removed and controlled with regard to their burn-up and mechanical integrity, then they are transported back to the top of the reactor core if they have not yet reached the burn-up target **[7]**.

(3) DWT-NPG

The DWT blanket with Nitride heavy nuclide Particle fuel in circulating helium-Gas (named DWT-NPG) is based on the fast neutron transmutation concept, in which the helium gas is adopted to cool the structural walls, FP zones and AC zones. LiPb in the separate LM zones is only used as the tritium breeder. 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. The proper helium flowing system will be needed not only to cool the fuel particles and structural materials but also to disperse the tiny particle in the complex geometry of blanket. The proper cooling channel inlet/outlet design is needed to prevent the jam of the tiny particles. The designs of fuel composition and particle size must satisfy the dynamics effect of gas-solid two phases flow.

The AC nitride with TiN coating as a fuel in a fast transmutation system offers potentially enhanced performance compared to the conventional oxide fuel due to the better neutron economy, higher thermal conductivity, good FP retention and mutual solubility. The pyrochemical methods can be applied in the reprocessing of nitride fuel, which shows that actinides are reasonably separated from fission products, and the high level wastes are of nearly actinides-free form in the back-end fuel cycle **[8].** The porous TiN probably needs to be coated on the outer layer in order to prevent the fuel damage by the collisions of the particles. Highly enriched ^{15}N would have to be used for the nitride fuel in order to prevent the formation of hazardous ^{14}C through the reaction $^{14}N(n, p)^{14}C$. Although the nitride as an advanced fuel concept has been envisaged in many institutes especially in Europe and Japan, the technology maturity of fuel fabrication is still in the basic R&D or laboratory levels **[9]**.

2 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) [10]. Theoretical and experimental studies indicate 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 [11-13]. Therefore, plasma core of ST can be used as a compact volumetric external neutron source for subcritical syetem FDS-ST. 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[14] 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, i.e. water-cooled Copper (water-Cu)[15,16], liquid Li self-cooled (Li-SC)[17], water-cooled Li (Water-Li)[18] and liquid metal-blanketed Copper (LM-Cu) CCPs[19] have been investigated.

3 Electricity Generation with a Fusion Power Reactor

The ultimate goal of the fusion program is the development of large-scale power plants for the production of electricity. 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 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.

Two optional concepts of liquid lithium-lead 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 FCIs (Flow Channel Inserts), e.g. SiC_f/SiC composite, are designed and used inside the LiPb coolant channel, which act both as thermal and electrical insulators to keep the temperature of RAFM structure below the maximum allowable temperature. Coating 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 that 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. It uses quasi-static LiPb flow instead of quick flowing LiPb as in DLL design. Coating is probably needed to protect the structure and to reduce tritium permeation and MHD effects. The lower LiPb outlet temperature leads to a lower thermal efficiency compared with the DLL blanket design. The details on this design can be found in Ref [19,20].

4 FDS-III: a High Temperature Fusion Reactor for Hydrogen Generation

Hydrogen is considered as the most potential energy carrier in the future. It is clean, powerful, renewable and environmentally benign. But it scarcely exists in the nature, it must be produced with raw material and energy. A promising method for the production of hydrogen is that nuclear power would be used as a provider of electricity in the electrolysis process or as a provider of high-temperature heat in the thermo-chemical cycles technology, which needs the high temperature range above 900 ^{o}C to achieve high efficiency of hydrogen production.

To achieve a high temperature above 900 ^{o}C, one of the most challenging issues about the fusion reactor is the structural material under irradiation. Some materials with good properties at high temperature, such as SiC_f/SiC composite, V-alloy and W-alloy etc., are considered as the candidate structural materials for high temperature fusion reactors. But some issues for these materials limit their application in fusion, including the unsolved fabrication and joining technology, rather low thermal conductivity for SiC_f/SiC composite, the high induced radioactivity and afterheat for W-alloy, bad compatibility of V-alloy with coolant etc.. As a result, the development of high temperature fusion reactors is limited by the current status of material technology.

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, for example, an optimized blanket design with innovative idea (e.g. special multi-layer FCI design etc.) is considered to obtain high temperature heat based on the relatively mature and most promising RAFM steel (allowed temperature up to 550 ^{o}C) as structural material and SiC_f/SiC composite or other components made of refractory materials with low thermal conductivity (allowing temperature up to 900~1000℃) as flow channel electrical and thermal insulators in a dual-cooled liquid LiPb blanket. The details on the blanket design can be found in Ref. [22].

5 Test Blanket Module

To demonstrate and validate the feasibility of the candidate blankets, the strategy for TBMs (test blanket modules) development has been proposed, which covers three-phases e.g. Out-of-pile experimental Mockup in liquid LiPb experimental loops, EAST-TBM (the Test Blanket Module for EAST, which is expected to operate in 2006, the goal parameters are listed in Table 1 for comparison with ITER operation conditions) and ITER-TBM (named DFLL-TBM: Dual-functional Lithium-Lead - Test Blanket Module). The reference preliminary scenarios of three typical TBM designs are being designed at ASIPP in wide collaboration with various institutions in China [23,24].

To balance the reduction of potential risk and the pursuit of potential attractiveness of the technological development, the two-step testing strategy of DFLL-TBM is proposed, including the early SLL(Quasi-Static Lithium Lead)-TBM testing and late DLL(Dual-cooled Lithium Lead)-TBM testing. DFLL (Dual-Functional Lithium Lead)-TBM system is designed with a similar basic structure and auxiliary system for SLL and DLL testing except for including FCIs and more quickly flowing LiPb in DLL-TBM.

The DLL-TBM consists of a 626 mm x 1832 mm x 476 mm steel box, reinforced by two radial-poloidal (rpSP) and six ‘⏋’ shape toroidal-poloidal (tpSP) stiffening plates, containing the self-cooled LiPb breeder / multiplier as schematically shown in Fig.1. The module box is formed by a U-shaped FW with toroidal helium gas cooling channels.

The pressurized helium gas at 300□ is delivered to the TBM through concentric pipes with the outlet temperature of 410□. The helium gas mainly transfers the nuclear heat of the TBM structure to water through heat exchanger.

The LiPb with the inlet temperature of 480□ and the outlet temperature of 700□ will carry the nuclear heat of the breeding zones to the LiPb/He heat exchanger which is installed in the transporter.

6 Development of Design and Analysis Tools

To carried 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, and their coupling analysis. Two of the integrated code systems are VisualBUS and TOPCODE.

6.1VisualBUS/HENDL: a Multi-functional Neutronics Analysis Code System

The analyses of complex nuclear systems such as fusion, fission, hybrid reactors and accelerator systems require particle transport simulations to determine the engineering response functions. With advanced computer performances, particle transport simulation calculations are preferentially performed with integrated physical procedures and multi-dimensional geometric models, which is able to handle the problem physics accurately in complicated geometries.

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 GUI (Graphical User Interface). 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 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. MCAM (MCNP Automatic Modeling) and SNAM (SN Automatic Modeling) are two of the functional codes of the integrated systems and HENDL (Hybrid Evaluated Nuclear Data Library) is 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 model into the format of MCNP input file. and vice versa. Besides the conversion functions and basic modeling ability, MCAM is also a fully featured visualization tool and property editor for MCNP model. 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, and DANTSYS etc.. SNAM can import CAD data from most leading solid modeling system and generate the neutronics calculation model automatically, eliminating the need for time-consuming combinational geometry creation and verification. The neutronics calculation model can be converted into input files of SN codes and vice versa. SNAM can also be used to visualize the calculation results from SN codes..

HENDL has been developed to meet the need for calculation and optimization of fusion, fission and fusion-fission hybrid systems. It is a compilation of nuclear data selected from the various national and international evaluated nuclear data files. Several working libraries are prepared, e.g. the coupled neutron gamma-ray multi-group library HENDL/MG for the SN transport calculation and continuous point-wise neutron data 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.

6.2 TOPCODE: An Integrated Code System for Risk-Benefit-Cost Analysis and System Optimization

A goal of integrated system analysis for fusion system 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. The Risk-Cost-Benefit analysis on fusion can provide an improved methodology for investigating the trade-offs between the health & environment risks and economic benefits of fusion by incorporating public attitude to fusion energy in the analysis.

Development of an integrated code system named TOPCODE is underway to achieve comprehensive Risk-Cost-Benefit analysis on fusion related systems, including analysis on different scenarios considering carbon dioxide emission restrictions, price of fuel for fission and fusion, waste disposal, public attitude to risk and acceptance of fusion. TOPCODE is a coupling tool of the fusion system optimization and economics analysis code SYSCODE (System Analysis Code) and the PSA (Probabilistic Risk Assessment) code RISKA for the Risk-Cost-Benefit analysis.

Economic assessment of power plant includes modeling and calculation of internal costs (TCC: Total Capital Cost, C_{OP}: total cost of annual operation and COE: Cost of Electricity) and external costs. The internal costs of electricity generation do not include costs such as those associated with environmental damage or adverse impacts upon public and occupational health which are included in the "external costs" [25]. The economics difference between a hybrid fusion-fission system and a pure fusion system mainly comes from the extra benefit (BOE: Benefit of Electricity) of a multifunctional blanket which can transmute high level wastes and produce additional fissile fuel and tritium if needed. The net cost of electricity (COE') can be represented by COE' = COE –BOE. The detailed cost-and-benefit models for not only pure fusion but also hybrid fusion-fission have been developed. Uncertainty and sensitivity analyses can be carried out based on the models. SYSCODE integrating the details of physical, engineering and financial models has been developed for the Cost-and-Benefit calculation of fusion and fusion-fission hybrid systems with a function of multi-objective parameters optimization based on the Generic Algorithms.

For a complex and new system such as a fusion energy system, a comprehensive and integrated assessment of the safety is necessary, including the probability, progression and consequences of equipment failures or transient conditions to derive numerical estimates that provide a consistent measure of the safety. RISKA (Risk Analysis Code) has been developed for an advanced general-service tool for PSA, including fault tree and event tree analysis, important analysis, common cause failure analysis, human failure analysis, uncertainty and sensitivity analysis etc. Development of the fusion-oriented version of RISKA is underway.

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Table 1 Main core parameters of FDS series design

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	4.0	5.3
Safety factor /$q_{\_95}$	3.5	5.0	5.5	/	3
Auxiliary power /P_{add}(MW)	50	80	19	/	73
Energy multiplication /Q	3	31	5	/	≥10
Average neutron wall load($MW \cdot m^{-2}$)	0.5	2.72	1.0	/	0.57
Average surface heat load ($MW \cdot m^{-2}$)	0.1	0.54	0.2	0.2	0.27

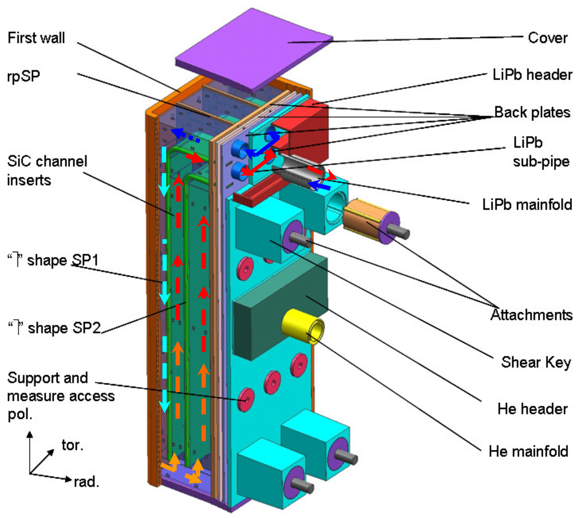

Fig. 1 3D Structure View of DFLL-TBM