DESIGN ACTIVITIES OF A FUSION EXPERIMENTAL BREEDER *

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Abstract

The fusion reactor design studies in China are under the support of a fusion-fission hybrid reactor research Program. The purpose of this program is to explore the potential near-term application of fusion energy to support the long-term fusion energy on the one hand and the fission energy development on the other. During 1992–1996 a detailed consistent and integral conceptual design of a Fusion Experimental Breeder, FEB was completed. Beginning from 1996, a further design study towards an Engineering Outline Design of the FEB, FEB-E, has started. The design activities are briefly given.

1. INTRODUCTION

Fusion energy is the long-term goal in China. However, it seems unlikely that commercial fusion plants will be available in China before 2050, mainly due to the fact that fusion energy based on the typical tokamak is not economically competitive with other energy sources. It is of vital importance that an essential intermediate application of fusion energy is implemented at an earlier time. Fission energy is practical. The fusion-fission hybrid reactor has the potential to provide plenty of commercial fissile fuel and also to provide a valuable means of long-lived high level radioactive waste disposal through transmutation, both benefit the development of fission energy, and thus becomes an essential intermediate application of fusion energy. This potential is worth studying, and a fusion-fission hybrid reactor program was proposed and approved. An important milestone in the course of fusion breeder development is a Fusion Experimental Breeder, FEB.

2. DETAILED CONCEPTUAL DESIGN OF THE FUSION EXPERIMENTAL BREEDER[1]

The fusion reactor design studies in China are mainly carried out at the Southwestern Institute of Physics, SWIP, and the Academia Sinica Institute of Plasma Physics, ASIPP, under the support of a fusion-fission hybrid reactor research Program. Based on two conceptual designs separately performed by SWIP and ASIPP during1988--1991, in order to be ready for the engineering design that follows, a detailed consistent and integral conceptual design was completed by a joint effort in 1993–1996.

2.1. The basic device

The FEB is He-cooled with a major radius of 4 m, minor radius of 1 m, magnetic field on axis 5 T, plasma current 5.7 MA, and a neutron wall loading of 0.42 MW / m^2. Liquid lithium is adopted as tritium breeder in the blanket to improve the heat conduction in the pebble bed. The perspective drawing of the FEB is given in Fig.1. Effort has been made to give a consistent design of the FEB: the blanket inlet/outlet channels were arranged carefully for blanket maintenance purpose, the fitness of various components was checked using computer animation software, and comprehensive analyses were carried out focused on common objects, e.g., blanket or TFC.

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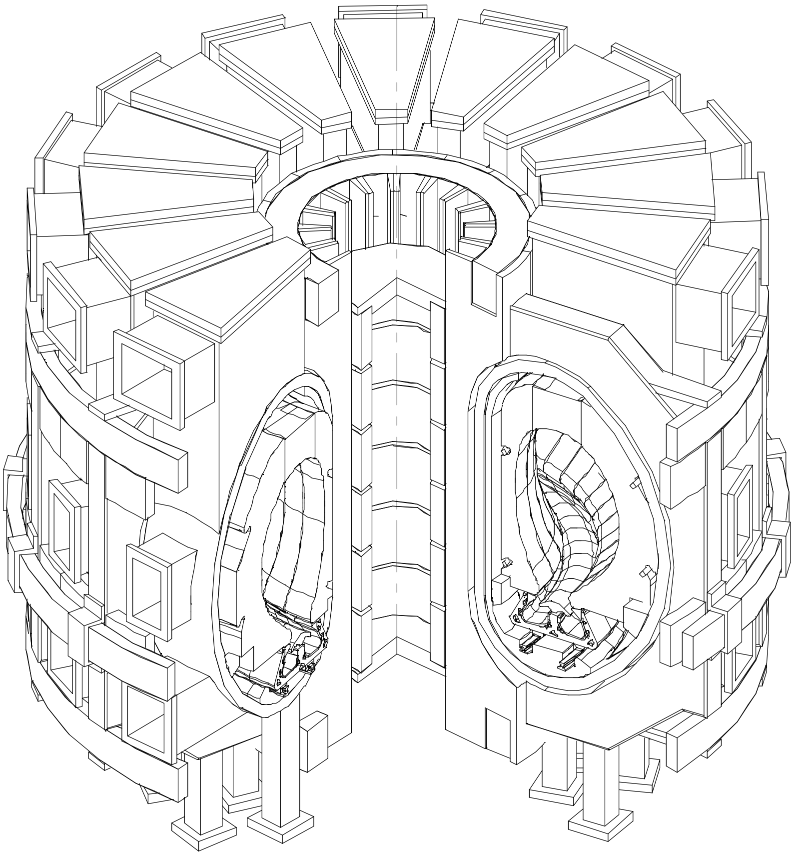

Fig.1. The perspective drawing of the Fusion Experimental Breeder, FEB.

2.2. Blanket / shield

The pebble bed structure is adopted for the complex tokamak geometry. To minimize the size of the reactor and to enhance reliability, a thin inboard blanket is chosen and the fuel breeding is limited only in the outboard blanket. Sixteen blanket sectors correspond to 16 TFC. Each sector consists of 2 inboard and 3 outboard blanket modules. 3-D neutronics calculation was carried out using FENDL/MG [2]plus data library for fission fuels based on ENDF/B-VI. The helium pressure is 5 MPa. The pumping power is calculated to be 5.9 MW giving a He circulation power fraction of 2.7% of the total thermal power. Following 3-D neutronics analyses, some local shield layers are added to attenuate the streaming neutrons at the lower port of the vacuum vessel.

2.3. Environment analysis

The radioactivity related parameters of the FEB were calculated. Furthermore, people are concerned about concentrations of some actinides: ^{237}Np for its decay daughters with high toxicity and long lives, and ^{232}U for being high toxic and long-lived α-decay emitter. The calculation gives the concentration of ^{237}Np and ^{232}U in uranium after one-year operation and two years after shutdown is $8.2{\cdot}10^{-5}$ and $4.4{\cdot}10^{-10}$, respectively, which is quite acceptable in fuel reprocessing.

Besides, a safety analysis of LOCA/LOFA in the outboard blanket was made and the capital cost of the FEB assessed. A PRA was carried out to assess the failure probability of a series of sequences.

The peak temperature evolution was obtained assuming that the reactor is shutdown immediately after a primary cooling system failure. Combining this with results of the PRA, a first level PRA was completed resulting in the probability and consequence for a series of sequences. The total capital cost of FEB was calculated to be B$2.1 (1989 US $), assuming the construction time is 8 years.

3. ENGINEERING OUTLINE DESIGN OF THE FEB, FEB-E

After this, an engineering outline design of the FEB has been started from 1996. This design study will form a transition from conceptual to engineering design in our research. The task is to improve the FEB design, to add safety margins with respect to core plasma and reactor operations, and to enhance the safety and reliability of the fusion experimental breeder. It is also required to perform the engineering outline design of the key components. The design study should deal with engineering and technological issues as far as possible. The design should be feasible from the engineering point of view with regard to materials, structure, fabrication, maintenance etc. In-depth analyses should be performed to support the design.

blanket design and analyses

The attachment lock of blanket in the vacuum vessel should be strong enough to withstand the tremendous E-M load generated during transient events of plasma current, while flexible enough to minimize the thermal stress and to ease the replacement during maintenance. The concept of belt-bar [3] is accepted. Strong bars above 15 cm in diameter are connected ends to end to form a belt. Four belts are firmly fixed on the inner wall of the vacuum vessel. Blanket modules are attached to the belts along an inclined direction. In-depth thermo-mechanics analyses will be made to examine and improve this design scheme. Three-dimensional thermal analysis was performed. The helium pressure is enhanced from 5 MPa to 10 MPa for the FEB-E. Compared with the FEB design, the peak temperatures are lowered by about 150 °C. Then, a stress analysis was made which included the He pressure and thermal loads. The peak stress is 185 MPa, located in the strengthened ribs in the fuel zones.

Divertor design and analyses

A close type gas box divertor is adopted in the FEB-E design. Joint operation of gas puffing and impurity injection is proposed for the purpose of plasma temperature and density control. For the purpose of impurity retention and retaining the beneficial effect of redeposition of the impurity, boron is injected near the target plate. The position of gas puffing is optimized using a plasma transport code. A divertor module was designed. The fabrication and maintenance of the divertor have been considered. Thermo-mechanics analyses were made for the target plate. With finned cooling channels (extended surface concept[4]) the effective heat transfer coefficient is calculated to be $1.0{\cdot}10^4$ $W/m^2{\cdot}K$. For a peak heat flux of 4.5 MW/m^2, the peak and lowest temperatures of the Be are 452°C and 254°C, respectively, and the peak temperature of copper is 317°C. Stress calculation was performed with the loading of temperature difference and helium pressure. The calculated peak stress is 203 MPa.

3.1. Eddy current analysis

The in-vessel components must be capable of withstanding the impacts from current quench during a plasma disruption without being damaged. A 2-D axisymmetric model was set up to simulate the whole arrangement of the in-vessel components. The induced eddy currents during a centered plasma disruption are calculated. A current quench time of 10 ms is taken. The time-space distributions of toroidal eddy current in various components were obtained, as well as the E-M load generated, which will be loaded to the modules of blanket and divertor for mechanics analyses.

3.2. Tritium inventory

A SWITRIM code has been developed to simulate the whole circulation process of tritium through various subsystems, including burning, breeding, isotope separation, tritium extraction, and storage processes to assess the tritium inventory in these sub-systems. The coupled equations of tritium inventory variations in all sub-systems are established. Discrete transfer of tritium from blanket to tritium extraction system is considered. Using the SWITRIM code, the tritium inventory and its distribution were obtained. Under full power operation of the FEB-E, 0.5 kg initial tritium inventory will meet the requirement of circulation.

3.3. TFC shield

In accordance with the close-type divertor design, the TFC shielding has been recalculated. Since the streaming neutrons are effectively reduced, this configuration is beneficial. Satisfactory shielding is provided. The total nuclear heating power in the TFC is calculated to be 4.2 kW.

3.6. High power density test blanket module

For fusion-based transmutation reactors and commercial fusion reactors as well, the power density of the blanket should be high enough to meet the economic requirement. A high power density blanket is being designed as a test module of the FEB. The concept of LiPb eutectic/ minor actinide oxide suspension is adopted to reach an aiming power density of 50–100 MW/m^3. The structure of the module is similar to that of the FEB but with much more cooling panels to account for the much higher power density. This structure is suitable for power density flattening by arranging different proportion of fissile fuel along the radial direction. The one-dimensional neutron transport calculation gives a k_{eff} of 0.82, an energy multiplication of 40. The power densities appear to be quite uniform ranging from 40 – 70 MW/m^3. The inlet/outlet temperatures of the 10 MPa helium are chosen to be 250/400 °C. A 3-D temperature distribution is calculated, giving a peak temperature of 1210 °C in LiPb and 570 °C in structural material. Further design measures should be taken to lower the temperature..

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