OVERVIEW OF FUSION-FISSION HYBRID REACTOR DESIGN STUDY IN CHINA

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The motivation for developing fusion-fission hybrid reactors is discussed in the context of electricity power requirements by 2050 in China. A detailed conceptual design of the Fusion Experimental Breeder (FEB) was developed from 1986–1995. The FEB has a subignited tokamak fusion core with a major radius of 4.0 m, a fusion power of 145 MW, and a fusion energy gain Q of 3. Based on this, an engineering outline design study of the FEB, FEB-E, has been performed. This design study is a transition from conceptual to engineering design in this research. The main results beyond that given in the detailed conceptual design are included in this paper, namely, the design studies of the blanket, divertor, test blanket, and tritium and environment issues. In-depth analyses have been performed to support the design. Studies of related advanced concepts such as the waste transmutation blanket concept and the spherical tokamak core concept are also presented.

KEYWORDS: *fusion-fission hybrid, fusion breeder design, fusion transmutation*

I. INTRODUCTION

Since 1986, hybrid reactors have been seriously considered as a national project in China because it is predicted that with a population of 1.5 billion, 1200 to 1500 GW(electric) power will be required by 2050. The fusion-fission hybrid system is considered to be a step toward pure fusion. A series of fusion-fission hybrid studies has been made in China.[1–9] Previously, in 1980–1985, physics concepts studies were conducted, and in 1986–1995, a detailed conceptual design of the Fusion Experimental Breeder[6] (FEB) was completed. The FEB has a subignited tokamak fusion core with a major radius of 4.0 m, a fusion power of 145 MW, and a fusion energy gain Q of 3. The blanket compositions are liquid lithium or Li_2O/helium/ferritic steel/uranium/beryllium for tritium breeder/coolant/structure/fissile fuel/neutron multiplier, respectively. The FEB is tritium self-sufficient, and the annual fuel breeding capability is ~50 kg. Superconducting coils are employed. In 1996–2000, based on the FEB design, an engineering outline design study of the FEB, FEB-E, was performed. In addition, designs of the fusion-driven hybrid system—a multipurpose experimental hybrid reactor that could perform many functions such as breeding nuclear fuel, transmuting long-lived wastes, producing tritium for fusion fuel cycling—as an alternate approach to utilize fusion energy technology based on previous hybrid reactor studies in China are being conducted.[9]

In this paper, the main design activities and results are presented, especially including the outlined engineering design of the fusion core, fuel breeding blanket, divertor, related tritium and environmental issues, and the related advanced concepts such as the waste transmutation blanket concept and the spherical tokamak core concept.

II. FUSION CORE

The fusion core design studies in China have included the conceptual design of the conventional tokamak experimental reactor FEB and the conceptual optimization of the spherical tokamak in addition to the

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continuous experimental efforts on tokamak devices. The FEB core design study jointly performed by the Southwestern Institute of Physics and Institute of Plasma Physics, Chinese Academy of Sciences, Joint Design Team has covered numerical simulation and analyses on plasma heating, current drive, energetic alpha-particle confinement, fueling and ash exhausting, plasma burning control, plasma evolution process, divertor design with gas puffing and electromagnetic plug, sensitivity of plasma parameters to uncertainties, etc. A set of optimized design parameters is given in Table I (Ref. 1).

III. BLANKET

The blanket has been designed in detail with regard to its fabrication, attachment lock inside the vacuum vessel, and maintenance. The pebble bed structure is chosen for the complex tokamak geometry. LLi is adopted as the tritium breeder to improve the heat conduction within the pebble bed, and ferritic steel HT-9 is adopted as the structural material. Figure 1 shows the cross section of the outboard blanket module at the midplane. It is a long curved box along the poloidal direction; four panels with built-in cooling channels are used for heat removal. The blanket module is optimized for low-energy multiplication and a high total breeding ratio, $T + F$, using a one-dimensional neutronics transport code. Then, a three-dimensional Monte Carlo calculation was carried out using FENDL/MG and a data library for fission fuels based on ENDF/B-VI. The result is $T + F = 1.35$ and 0.08 fission/fusion neutron. The share between T and F can be adjusted by varying the enrichment of ^{6}Li in Li (around natural enrichment). So, the FEB can be tritium self-sufficient while it produces 50 kg/yr Pu, assuming a fuel breeding ratio F of 0.2 and a loading factor of 0.4. Figure 2 gives the exploded view of the blanket module.

TABLE I

Main Parameters of the FEB Core

Major radius, R (m)	4.0
Minor radius, a (m)	1.0
Plasma current, I_p (MA)	5.7
Toroidal field, B_t (T)	5.2
Average density, $\langle n_e \rangle$ (10^{20} m^{-3})	1.1
Average temperature, $\langle T \rangle$ (keV)	10
Plasma volume (m^3)	134
Fusion power, P_{fu} (MW)	143
Auxiliary power, P_{aux} (MW)	50
Neutron wall loading, P_w (MW/m^{-2})	0.43
D-T neutron rate (n/s)	1×10^{19}
Fusion energy gain, Q	$\sim$3
Operation availability (%)	40

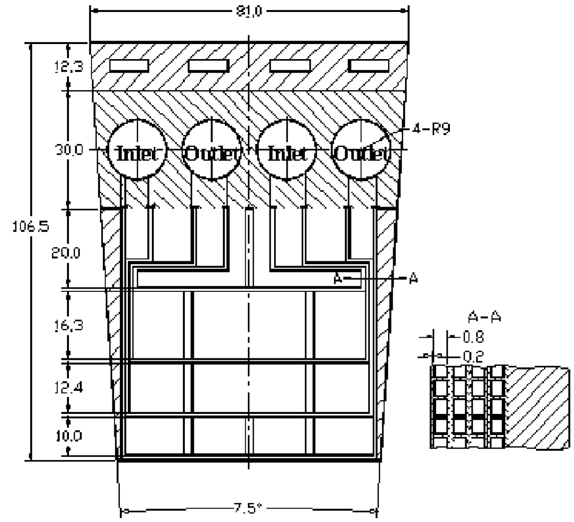

Fig. 1. Cross section of the outboard blanket module.

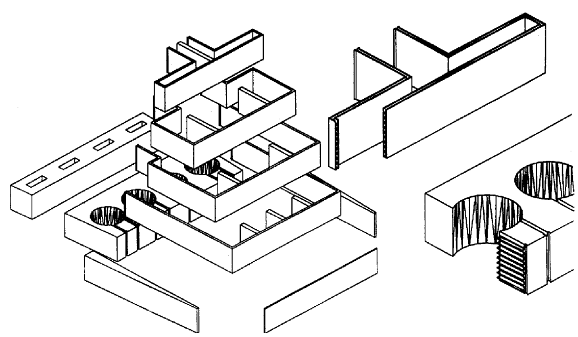

Fig. 2. Exploded view of the blanket module.

The structure of the blanket is quite simple and robust. Fabrication of the panels with built-in cooling channels via machining, hot isostatic pressing, and forming has been demonstrated. The joining of panels to the blanket module frame appears feasible using advanced joining technology such as e-beam welding. So, fabrication of the blanket module is possible. Compared with the FEB design, the helium pressure is enhanced from 5 MPa for the FEB to 10 MPa for the FEB-E. Three-dimensional thermal and structural analyses were done. The results are shown in Fig. 3. The thermal stress in the strengthening ribs in the high power density zone is too high because of the lack of cooling. The support of the first wall by these ribs needs to be improved. Hydraulics calculation for the blanket was also done, giving a pumping power fraction in the total blanket thermal power of 2.7%.

The attaching lock of the blanket in the vacuum vessel should be strong enough to withstand the tremendous electromagnetic load generated during plasma current transient events as well as flexible enough to minimize

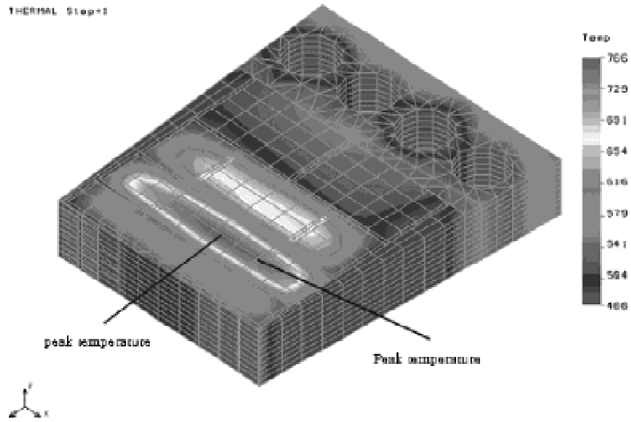

Fig. 3. Temperature distribution in the blanket module.

the thermal stress and to ease the replacement during maintenance. The concept of a belt-bar[10] is accepted. Strong bars above 15 cm in diameter are connected end 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. These belt-bars are used to withstand the electromagnetic load on the modules. Figure 4 displays the attaching lock of the blanket inside the vacuum vessel, showing the blanket modules, belt-bars, upper plug, and lower supports. A two-dimensional axisymmetric model for eddy current analysis is employed to simulate the whole arrangement. The time-space distributions of the toroidal eddy current in various components were obtained, as well as the generated electromagnetic loads.

A first-wall life calculation (FWLC) code was developed to study the first-wall failing modes.

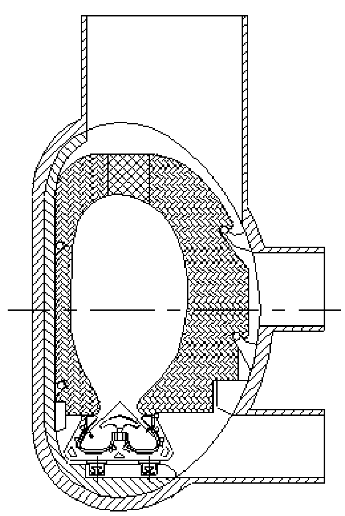

Fig. 4. Attaching lock of the blanket in the vacuum vessel.

IV. DIVERTOR

A close-type divertor was designed to replace the open-type one in the FEB. The transport of impurities and plasma along the separatrix is simulated under combined impurity injection and gas puffing. The injection of impurity is fixed in the vicinity of the target, while the gas puffing location can be changed along the separatrix to study its effect. The gas puffing at the midplane results in the highest radiation power; hence, there is a low heat flux at the target plate.

High-mode detached plasma has been observed in experiments.[11] It is characterized by a large irradiation power fraction in the edge plasma and a sharp drop of plasma pressure along the magnetic line in the scrape-off layer (SOL). Let the pressure drop fraction $f_p = 2P_d/P_s$, where P_d and P_s are the plasma pressure at the divertor target and at the stagnation point, respectively. A two-point model in the SOL is used to estimate the pressure drop of the plasma along the magnetic line. According to the calculation result, the plasma density at stagnation point n_s should be in the range of 2×10^{19} to 8×10^{19} m^{-3}, and the irradiation power fraction f_{rad} should be 0.2 to 0.8 in order to obtain a low heat flux at the divertor target plates.

A divertor structure is designed; the structure is shown in Fig. 5, and the installation and maintenance of the divertor cassettes are shown in Fig. 6. The target plates are cooled by 4-MPa pressurized helium. The original peak heat flux at the target plate is calculated to be 40 MW/m^2. A reduction factor of 2 to 4 results from the oblique arrangement of the plates; the further reduction relies on the detached plasma operation mode. As a result, the peak heat flux is estimated to be 4.5 MW/m^2. The temperature at the target is calculated as shown in Fig. 7. The minimum and maximum temperatures of the Be armor are 254 and 452°C, respectively, and the temperature of the Cu alloy as substrate material is 317°C, all of which are within the temperature window required.

The toroidal field coil (TFC) shielding has been recalculated for the FEB-E. Figure 8 gives the calculation

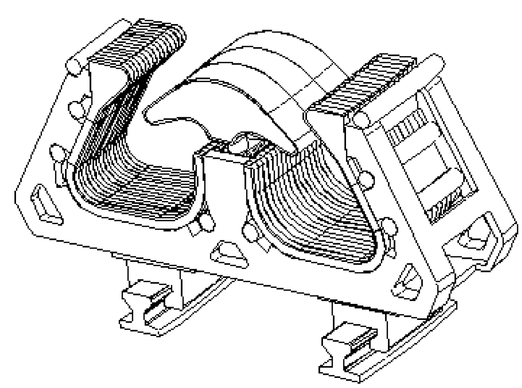

Fig. 5. Divertor structure.

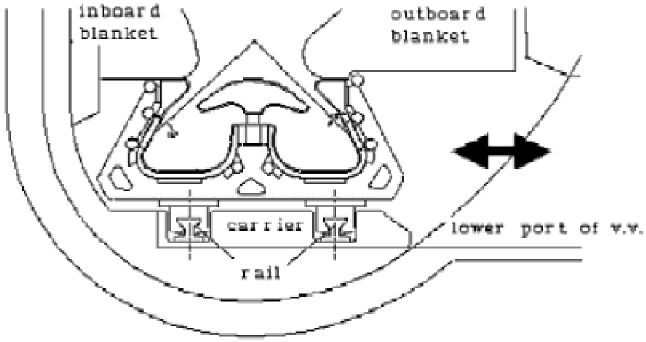

Fig. 6. Installation of the divertor module.

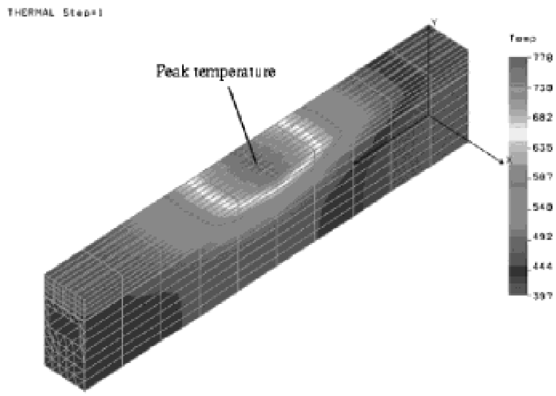

Fig. 7. Temperature distribution in the target plate.

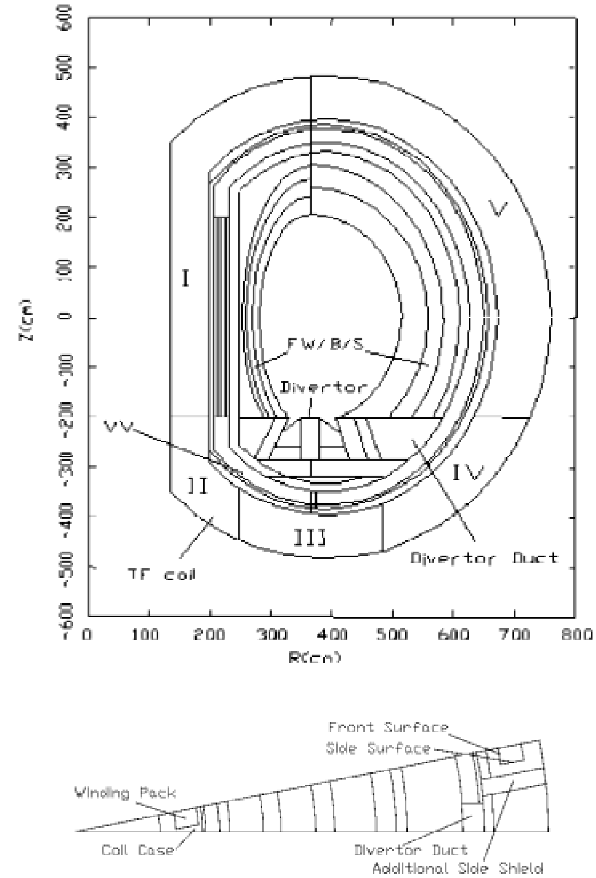

Fig. 8. Neutronics calculation model.

model. Since the streaming neutrons are effectively reduced, this divertor configuration is beneficial. The total heating power in the TFC, calculated to be 4.2 kW, mainly results from gamma heating and comes from region IV (see Fig. 8). In the outboard, the peak nuclear heating in the winding pack, the peak insulator dose, and the peak fast neutron fluence is 0.2 mW/cm^3, 1.4×10^6 Gy/4.5 full-power year (FPY), and 1×10^{18} neutrons/cm^2 4.5 FPY, respectively. The corresponding values in the inboard are nearly four times that in the outboard because of the thin inboard space. Thus, the TFC shielding is adequate.

V. TRITIUM ISSUES

The SWITRIM code has been developed to simulate the whole circulation process of tritium through various subsystems, including burning, breeding, isotope separation, extraction, and storage processes to assess the tritium inventory in these subsystems. The coupled equations for time-dependent tritium inventories in all subsystems were established. The discrete transfer of the tritium from the blanket to the tritium extraction system is considered. Using the SWITRIM code, the tritium inventory and its distribution were obtained. The time-dependent inventories in various subsystems are shown in Fig. 9. Under full-power operation of the FEB-E, 0.5 kg of initial tritium inventory will meet the requirement of circulation. Furthermore, the tritium leakage issue is analyzed. Under normal operations, tritium leakage from the helium coolant is negligible, and from the plasma exhaust system is low. Under accidents, when the blanket temperature goes up to 1000°C, tritium leakage increases because of the enhanced permeation from the lithium side, but it is still allowable. However, leakage from the plasma exhaust system is a concern in accidents.

VI. TEST BLANKET MODULE

A high-power density blanket module is designed for the FEB-E as a test module. The concept of LiPb eutectic/transuranium oxide suspension is adopted to reach an aiming power density of 50 to 100 W/cm^3. A structure is designed within the FEB-E vacuum vessel.

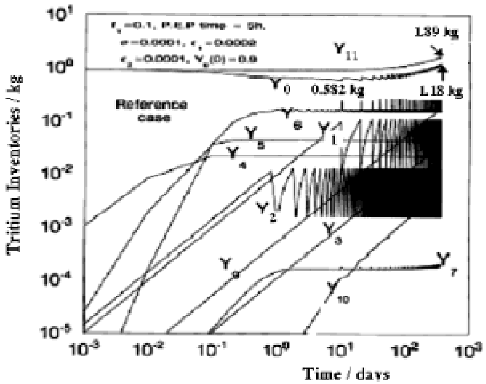

Fig. 9. Time variation of the tritium inventory in FEB-E subsystems: Y_0 = inventory in tritium storage, Y_1 = in outboard blanket, Y_2 = in inboard blanket, and Y_{11} = total tritium inventory.

One-dimensional neutronics calculation gives a k_{eff} of 0.84, an energy multiplication of 37, and a rather flat power density distribution throughout the blanket. The peak power density is 70 W/cm^3. The cross section of the module is shown in Fig. 10. Multiple cooling panels are introduced to reduce the peak temperature of the blanket. Each cooling channel has two passes through the blanket. The panels are held flexibly by the neighbor panels at the position of the turnings to keep the spacing. This also provides a strong structure to guarantee reliability. The first wall is supported by a second cooling panel and ribs to strengthen this weakest part of the module. Mechanical analyses will help to improve this first-wall design. Measures to be taken are to shape the first wall in a curved form and to thicken the second cooling panel. In order to avoid high temperature hence high stress at the ribs, no fissile materials are put into this zone. This configuration is suitable for power density flattening by arranging a different proportion of fissile fuel along the radial direction. In spite of up to 15 cooling panels, the blanket module is still simple and seems feasible to fabricate.

The temperature distribution in the blanket module is calculated using the ANSYS code and analysis. The calculation result is shown in Fig. 11. The peak temperature of LiPb and the peak interface temperature between LiPb and stainless steel is 810 and 460°C, respectively. The result is consistent with the analytical calculation.

VII. ENVIRONMENT ISSUES

The waste disposal rating and remote maintenance rating (RMR) are calculated for the FEB-E. The activation calculation and analyses of all long-lived radioactive nuclides are performed. The results indicate that the first wall and blanket structure materials of the FEB-E can meet the nuclear waste disposal criteria after a few weeks from shutdown. The RMR of the first wall and blanket at different times after 1 FPY of operation are calculated. The results indicate that based on the present design, an additional 25-cm lead layer is needed to meet the requirement of hands-on maintenance at the outer surface of the TFC shielding.[12,13]

Finally, based on the FEB-E design, a 1:10 model has been built, showing the details of the reactor structure including the cooling tubes, manifolds, and channels.

VIII. SPHERICAL TOKAMAK TRANSMUTATION CONCEPT

To seek a new and efficient way to realize a volumetric neutron source, the spherical tokamak concept has been studied in China. Spherical or low aspect ratio

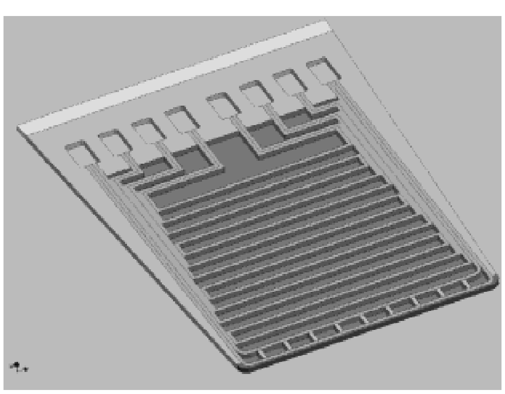

Fig. 10. Cross section of the high-power test blanket module.

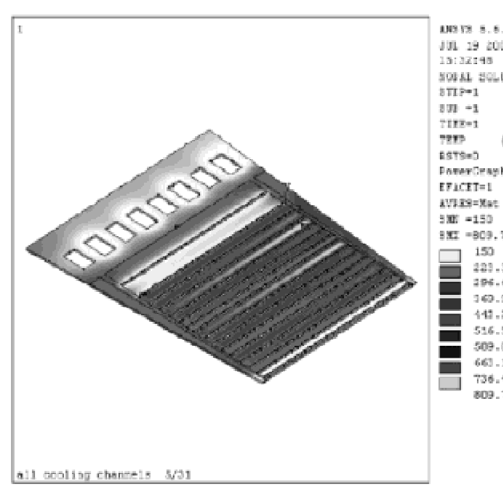

Fig. 11. Temperature distribution in test blanket module.

tokamaks[14] with an aspect ratio (A) in the range of 1.2 to 2.0 offer the possibilities of compact volumetric fusion neutron sources as well as fusion reactors requiring relatively low external fields. The conceptual study on the spherical tokamak has been oriented to the level of the volumetric neutron source for the purpose of nuclear waste transmutation. The optimized core parameters of the spherical tokamak neutron source are listed in Table II.

A design with an aspect ratio near the lower limit (due to limited space) requires an unshielded center conductor post (CCP) as part of the TFC circuit. The fully exposed CCP will receive severe neutron damage, resistive and nuclear heating power, which is one of the key components and requires replacement periodically. The analysis results for the CCP considering neutron radiation effects such as radiation damage, transmutation, nuclear and resistive heat removal, induced radioactivity, and blanket tritium breeding ratio in a spherical tokamak reactor compared with those estimated for the first wall of conventional tokamak reactors have been studied to see if the technical requirements are comparable.[15] The analyses have shown that the severity of neutron radiation damage and transmutation in the CCP is comparable with that estimated for the first wall of conventional tokamak reactors. Thus, radiation damage, transmutation effects, and other relevant problems of the CCP need to be studied further if a design with high neutron wall loading is adopted.

Since 1990, China has performed many studies on transmutation of long-lived nuclides in the blankets of hybrid reactors in addition to studies on breeding nuclear fuel using hybrid reactors. The studies have covered various blanket concepts[15–22]:

1. a thermal fission blanket concept with ^{239}Pu fissile neutron multiplication material for transmutation of fission products
2. hard neutron spectrum blanket concepts for transmutation of actinides
3. a thermal neutron spectrum blanket concept for transmutation of actinides
4. a helium and LiPb eutectic dual cooled fast fission blanket concept simultaneously for transmutation of actinides and long-lived fission products
5. a multifunction fuel cycle blanket concept.

TABLE II

The Optimized Core Parameters of Spherical Tokamak Volumetric Neutron Source

Total fusion power, P_g (GW)	0.1
Major radius, a (m)	1.4
Neutron wall loading, P_w (MWm^{-2})	1
Plasma current, I_p (MA)	9.2
Center post current (MA)	9.0
Aspect ratio, A	1.4
Elongation, κ	2.5
Triangularity, δ	0.45
β_N	6.5

In the studied concepts, fissile ^{239}Pu and ^{33}U have been put 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 to 1.0 MW/m^2, which is between the levels of that achieved in the Joint European Torus (JET) tokamak device and that in the ITER engineering design.

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