DESIGN STUDY OF FUSION EXPERIMENTAL REACTOR

M. SUGIHARA, N. FUJISAWA, H. IIDA, S. NISHIO, Y. SEKI, S. YAMAMOTO, J. ADACHI[1], T. HONDA[2], H. HOSOBUCHI[3], S. KASHIHARA, T. KOBAYASHI, N. MIKI, T. MIZOGUCHI[4], S. MORI[1], K. NAKASHIMA, J. OHMORI[5], B.W. RIEMER[6], R. SAITO, M. YAMADA, FER DESIGN TEAM
Naka Fusion Research Establishment,
Japan Atomic Energy Research Institute,
Naka-machi, Naka-gun, Ibaraki-ken,
Japan

Abstract

DESIGN STUDY OF FUSION EXPERIMENTAL REACTOR.

An overall review of the conceptual design studies of a next generation tokamak fusion experimental reactor (FER) at JAERI is presented. Major objectives of the FER are to demonstrate long, ignited D–T burning and the feasibility of key fusion reactor technologies. Two typical design concepts have been studied in detail, one based on the best physics databases and the other on rather conservative physics bases. Various flexibility scenarios and capabilities of the extension to enhance the reactor core performance have also been developed and incorporated into the design, in accordance with an overall plan of a phased construction and operation programme for the FER.

1. INTRODUCTION

In this paper, we report on an overall review of the design study of FER at JAERI undertaken during the past two years [1]. The study has been conducted in line with the national research and development programme recommended by the Subcommittee on the Next Step Device under the Fusion Council of Japan, in which long, ignited burning (about 800 s) with the assistance of non-inductive current ramp-up and moderate neutron fluence (0.3 $MW \cdot a \cdot m^{-2}$) are set as the primary goals. Accordingly, instalment of tritium producing blankets will not be necessary, and tritium breeding and recovery tests are planned to be carried out by blanket test mod-

---

[1] On leave from Kawasaki Heavy Industries, Ltd., Japan.
[2] Toshiba Corporation, Japan.
[3] On leave from Hazama-gumi, Ltd., Japan.
[4] On leave from Hitachi, Ltd., Japan.
[5] On leave from Toshiba Corporation, Japan.
[6] On leave from Oak Ridge National Laboratory, Oak Ridge, TN, USA.

ules. In conducting the FER design studies, we have employed the following guiding principles, mainly in order to cope with a great uncertainty in the physics database:

(1) Wide ranges of device parameters should be covered, for reference option;
(2) The machine should have sufficient flexibility for upgrading and extension of the operation regime.

Two sets of design options have been studied, following guiding principle (1). Throughout the design studies, a major effort has been focused on flexibility studies to enhance the machine capabilities in order to achieve higher reactor core performance, following guiding principle (2). The major categories of flexibility include: (i) plasma size enlargement; (ii) heating and current drive scheme flexibility; (iii) operational flexibility. The proper incorporation of these flexibility scenarios is examined in accordance with a well devised overall programme of phased FER construction and operation.

TABLE I. MAJOR PHYSICS GUIDELINES AND RESULTANT MAJOR DEVICE PARAMETERS OF FER OPTIONS 1 AND 2 DEVICES

	Option 1	Option 2
Operation mode	Non-inductive	Inductive
OH coil flux	75	130
Burn time (s)	800	500
Major/minor radius (m)	4.42/1.25	5.1/1.7
Plasma elongation	1.7	2.0
Safety factor, q_ψ	2.6	3.0
Field on axis (T)	4.9	4.7
Plasma current (MA)	8.74	15.8
Total beta (%)	5.0	5.6
Ion temperature (keV)	12	12
Ion density (10^{20} m^{-3})	1.1	1.05
Effective charge	1.5	1.8
Fusion power (MW)	410	733
Lifetime fluence (MW·a·m^{-2})	0.3	0.3
Neutron wall loading (MW·m^{-2})	1.1	1.1
Divertor	single null	double null
Breeding blanket	test modules	test modules
Ignition margin with Mirnov type scaling	1.0	1.5

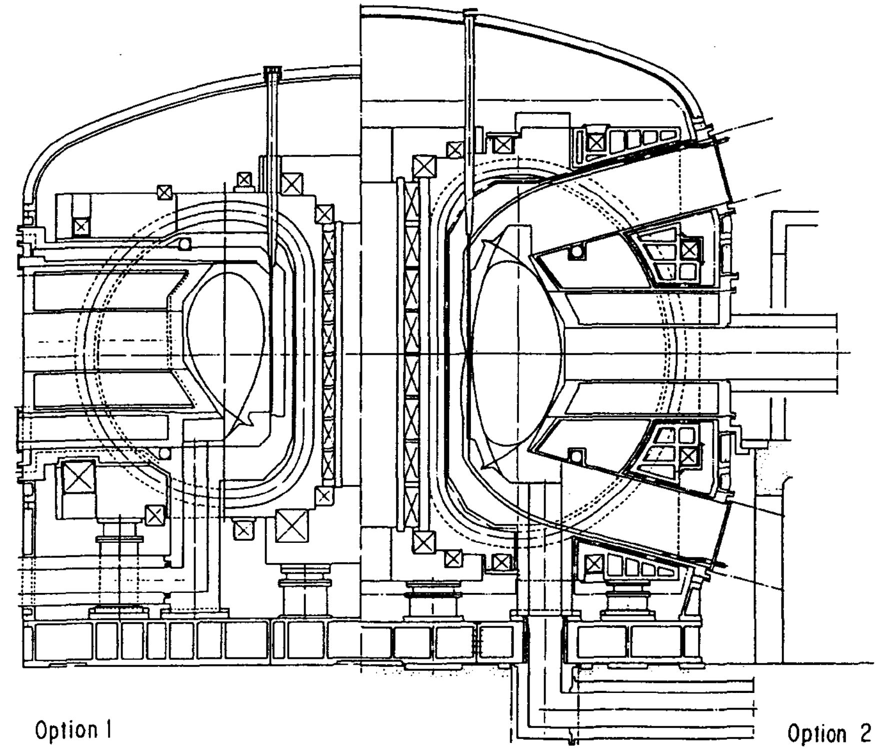

FIG. 1. Cross-sectional view of FER Options 1 and 2 device.

2. DESIGN OPTIONS

Considering the great uncertainties in the present physics database, we have prepared two sets of physics guidelines. The major physics guidelines and the resultant major parameters are summarized in Table I. Option 1 is based on the optimistic physics database, i.e. H-mode class of confinement, low safety factor and high beta value, etc. [2]. Cost effectiveness and machine reliability are pursued as recommended by the Subcommittee by reducing the reactor size with narrow ignition margin and with the assistance of non-inductive current ramp-up. Ignition will be reached when the assumed best operation and non-inductive current ramp-up are successfully realized.

In contrast, Option 2 is based on rather conservative physics guidelines, i.e. a wider ignition margin and conservative operational limit, in order to reduce the uncertainties in realizing the self-ignition condition. Fully inductive current ramp-up capability is also provided. Both design concepts have the potential for extension of plasma dimension and operation limit so as to enhance plasma performance, following guiding principle (2). In the Option 2 device, operations with rather high Q value will be expected from these extensions, even in the case of L-mode or slightly improved L-mode confinement. Non-inductive current ramp-up is to be used for this purpose or for extending the burn time in Option 2. A cross-sectional view of Options 1 and 2 is shown in Fig. 1.

One of the major differences between Options 1 and 2 is the choice of plasma elongation κ (κ = 1.7 for Option 1 and κ = 2.0 for Option 2). This choice was made by a careful assessment of the impact of κ on the overall concept and on various system quantities of the device, with the aid of a systems code [3]. The major reason for this choice is as follows: In Option 1, reliable controllability of the vertical position and a simple maintenance scheme for the radial straight motion of the core component are pursued so that a moderate elongation ($\kappa \approx 1.7$), which does not require PF coil location near the outer midplane, is employed. In Option 2, the device size is reduced, giving rise to some difficulties in position control and the maintenance scheme (non-horizontal motion), which leads to a greater elongation ($\kappa \approx 2$), with allocating the PF coils near the outer midplane.

3. FER DESIGN OPTION 1

3.1. Device and operational flexibility [4]

We have studied several sets of device and operational flexibility scenarios and incorporated them in the design in order to enhance the capabilities for higher reactor core performance.

(i) Plasma size enlargement scenario

In this scenario, an enlargement of the plasma dimensions is pursued mainly to back up the uncertainties in the confinement and operational limits. This enlargement is realized by the concept of replaceable reactor core components to be described in greater detail later. With this enlargement, the plasma minor radius and the plasma current are increased up to 1.43 m and 11.5 MA from their respective base values of 1.25 m and 8.7 MA, which results in an increase of the ignition margin up to 1.5 from a base design value of unity. The operational capability of this enlarged plasma is examined, and it is found that over one hundred seconds of burning are still available, only with increasing the capacity of the top solenoid and the outermost PF (ring) coil by about 30%. This result is mainly due to the fact that the magnetic null points approach the PF coils, which compensates for the increase in the plasma current.

(ii) Heating and current drive scheme flexibility with 'flexible port'

This flexibility scenario ensures the installation of the optimum heating/current drive method to realize a D–T burning state and a variety of operation and control

schemes for MHD activities, and a profile and burn temperature control scheme at every stage of the planned phased FER operation. One of the major points of this scenario is to ensure the installation and replacement of tangential neutral beam injector and RF systems with a 'flexible port' at any phase of machine operation. Another point is several backup scenarios for non-inductive current ramp-up. The first backup scenario is the increase of LH power by increasing the number of LH ports by the concept of a 'flexible port'. Another backup scenario consists of the introduction of inductive assistance for the operation. For example, the plasma current is ramped up to 4 MA non-inductively and subsequently ramped up to 8.7 MA inductively with maintenance of 100 s of burning. A fully inductive operation up to 4 to 6 MA is also considered for the initial phase of machine testing and aging.

(iii) Operational flexibility

This flexibility scenario provides the capability of operating with a variety of plasma shapes, e.g. modified elongation and triangularity and modified magnetic null configuration, as well as the possibility of extending the operational limit, i.e. of plasma current, beta, etc. Operations with enlarged plasma dimensions and inductive operation as mentioned above are directly tied up with this flexibility scenario. A major point in this scenario is the proper modification and enhancement of the poloidal field coils and their power supply systems. In the present studies, operation conditions of the poloidal field coils for all operation scenarios specified are examined throughout the whole period of operation. The optimum coil design for all operation scenarios has been done. An example for the operation of an enlarged size plasma in the plane of the supplied flux, ψ, and the plasma current, I_p, is shown in Fig. 2. The solid and dotted lines are the limiting lines for the designed operation conditions in high and low beta states for each coil. In this case, the limiting coils are Nos 10 (solenoid) and 14; the other coils are no limiting coils.

An incorporation of all flexibility scenarios in their complete forms from the very beginning should cause a substantial increase in the total device cost, and the replacement of the reactor core component requires considerable effort, especially after the D–T operations, even if it is scheduled. Thus, in the actual application of these flexibility scenarios, it is of primary importance to employ the idea of phased construction and operation. Accordingly, the possible occasions of replacing the reactor core component during the D–T experiment phase can be diminished, which may, in its turn, lead to a reduction of the potential difficulties of replacement and also to a reduction in the total project cost. Nevertheless, keeping the possibility of replacing the core components even after D–T operation is quite desirable in our flexibility considerations. The same idea is applied to the phased construction of PF coil power supply systems for operational flexibility. In fact, the PF power supply becomes 2.5 GW if all operational flexibilities are incorporated from the beginning, which is 1 GW for the reference operation scenario. By introducing the idea of

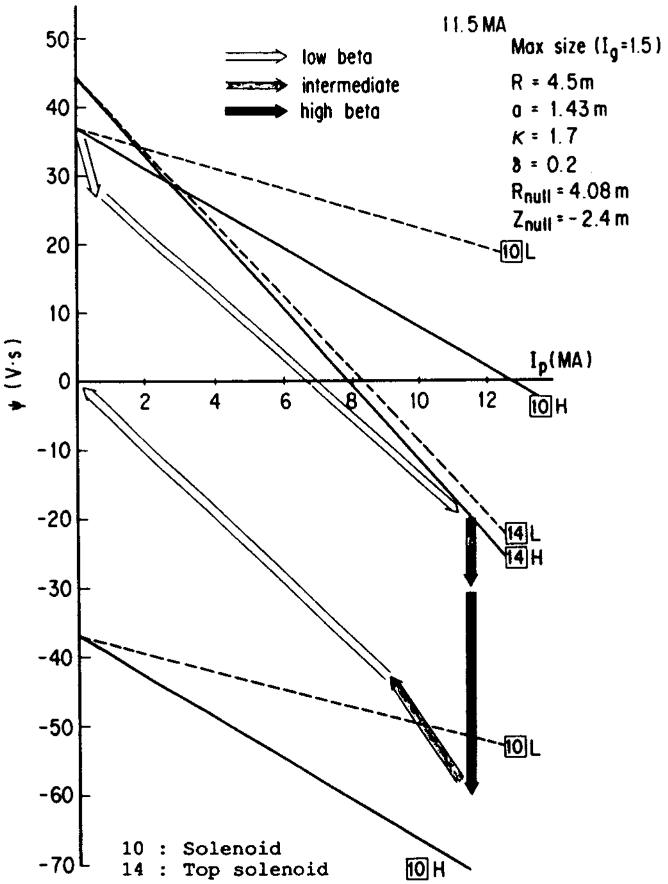

FIG. 2. Operation space in the plane of supplied flux, ψ, and plasma current, I_p. Solid and dotted lines are limiting lines for designed operation condition in high and low beta states for each coil, respectively.

phased construction and operation, PF coil power supply systems can be reinforced step by step, starting from the reference capacity of 1 GW.

3.2. Reactor configuration [5]

In developing the concept of reactor configuration for the Option 1 device, the following design philosophies are employed:

(i) The maintenance scheme should be simple and highly reliable.
(ii) Sufficient operational flexibility should be provided to accommodate a wide range of plasma operations.

(iii) A reduction in the capital cost should be considered for PF coil allocation, a simplification of the overall reactor structure, etc.
(iv) All plasma facing components should be replaceable, and some protection should be provided for particularly vulnerable portions.

In these philosophies, some important features should be mentioned: Passive shell conductors for vertical position control are installed in the outboard movable shield, and active control coils are placed in the permanent shield, which ensures the possibility of replacing all plasma facing components without removing the active coils. The outboard shield modules are water tank type, filled mainly with water for structural simplicity.

In design philosophy (iv), 'sacrificial' guard limiters are installed on the inner and upper first wall area, protruding into the plasma to protect the first wall against off-normal plasma conditions. Details of the guard limiter are shown in Fig. 3. The

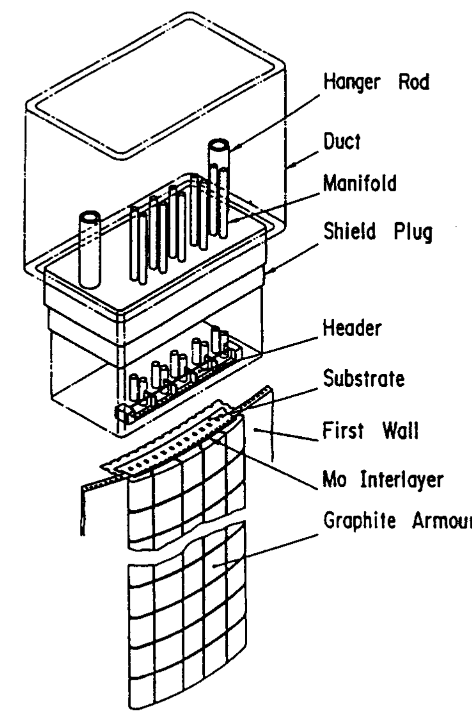

FIG. 3. Guard limiter concept to protect the first wall, which is an easily replaceable 'sacrificial' limiter.

upper guard limiter is also used as a startup limiter in the non-inductive current ramp-up phase. Most of the torus internals are suited to a simple straight line assembly/disassembly procedure through the space between adjacent TF coils for reliable maintenance. Inboard guard limiters are designed to be easily replaceable without breaking the plasma vacuum necessary for machine availability. A biological shield concept is employed to achieve 2.5 $mrem \cdot h^{-1}$ for personnel access into the reactor hall one day after shutdown.

When an enlargement of the plasma dimensions is required, the following changes from the basic structure are planned:

- Replacement of the L-type inboard and upper movable shields by a thinner tungsten shield;
- Replacement of the outboard movable shields;
- Replacement of the inclined divertors by flattened ones.

4. FER DESIGN OPTION 2

4.1. Plasma performance considerations

The basic parameters of the Option 2 device are based on a rather conservative operational limit with a considerable ignition margin of about 1.5, where the H-mode class of the confinement scaling law is evaluated (Mirnov type). By applying the same concept as is valid for the device and operational flexibility in Section 3 to this device, the plasma performance could be improved substantially. We have considered two types of extension. The first one is the enlargement of the plasma dimensions by procedures similar to those of replacing the reactor core components. The plasma minor radius is extended up to 1.9 m for the divertor configuration and to 2 m for the limiter configuration, which results in an increase of the plasma current up to 18 to 22 MA within the assumed operational limit in the basic Option 2 device. The second extension is that of the operational limit. For example, the safety factor q_ψ and the Troyon coefficient G are assumed to be extended to the same regime as in Option 1, i.e. $q_\psi = 2.6$ and $G = 3.5$. The plasma current is increased up to 21 to 25 MA by this extension of the safety factor. In addition, if we restrict the burn time below several seconds, helium ash accumulation might be neglected so that the effective charge could be lowered from 1.8 to, say, 1.5. With these extensions of the device and operational regimes, nearly ignited or rather high Q operation could be within our scope, even when an L-mode or a slightly improved L-mode class of confinement dominates the plasma. In these fully extended plasmas, however, the total fusion power becomes 2 to 3.5 GW, so that heat removal problems become quite serious. One of the possible countermeasures is to restrict the burn time below several seconds during which the plasma facing components should be able to survive.

4.2. Engineering considerations

Although the machine size of the Option 2 device is considerably larger than that of Option 1, basically the same design philosophies are pursued for the reactor configuration concept. Since, however, a plasma of higher elongation is employed and the PF coils are allocated to the optimum position for PF capacity in Option 2, maintenance by simple, straight line motion only becomes difficult, so that some modification of the scheme employed in Option 1 is required. A typical example developed for the Option 2 maintenance scheme is as follows: part of outboard shields, which will interfere with the outermost PF coils during maintenance, are permanent. The first wall area of that shield is, however, to be slid upward and downward and will be able to be removed through the hole of the outboard shield structure near the midplane. By using this part of the permanent structure, the passive shell conductors and the active control coils can be installed near the plasma in this area. The controllability of the vertical position is much improved by this concept. The inboard movable shields are divided into upper and lower parts. Each shield is designed to be replaced through the upper and lower divertor holes after the removal of divertor modules. These shields are toroidally segmented into twelve sectors and are moved on a relatively simple arc trajectory guide rail by a transfer machine. These guide rails are fixed on the inboard permanent shield; the guide structures are also expected to support the shields against the electromagnetic force during disruptions.

5. CONCLUSIONS

FER design studies have been described. A wide range of device concepts has been studied with the aid of system studies; two typical design options were selected. One option is based on a rather optimistic physics database; here, emphasis is placed on the enhancement of cost effectiveness. The other option is based on a rather conservative physics database; emphasis lies on enhancing the physics performance. Furthermore, various device and operational flexibility scenarios have been examined and incorporated into the design to further enhance physics performance. Engineering studies have also been carried out to demonstrate the feasibility of these device concepts.

ACKNOWLEDGEMENTS

The authors are grateful to Drs S. Matsuda and Y. Shimomura for valuable discussions and comments. They are also grateful to Drs S. Tamura, M. Yoshikawa, K. Tomabechi and S. Mori for continuous encouragement.

REFERENCES

[1] FER Design Team, Conceptual Design Study of Fusion Experimental Reactor (FY87FER) — Summary Report, Rep. JAERI-M 88-090, Japan Atomic Energy Research Institute (1988).
[2] FUJISAWA, N., et al., Conceptual Design Study of Fusion Experimental Reactor (FY86FER) — Main Physics Features Driving Design Concept and Physics Design Constraints, Rep. JAERI-M 87-093, Japan Atomic Energy Research Institute (1987).
[3] MIZOGUCHI, T., et al., Conceptual Design Study of Fusion Experimental Reactor (FY86FER) — Development of Tokamak Reactor Conceptual Design Code (TRESCODE), Rep. JAERI-M 87-120, Japan Atomic Energy Research Institute (1987).
[4] SUGIHARA, M., et al., Consideration of Device and Operational Flexibility in FER, Rep. JAERI-M 87-216, Japan Atomic Energy Research Institute (1987).
[5] KOBAYASHI, T., et al., Conceptual Design Study of Fusion Experimental Reactor (FY86FER) — Reactor Configuration/Structure Design, Rep. JAERI-M 87-139, Japan Atomic Energy Research Institute (1987).

PRELIMINARY CONCEPTUAL DESIGN FOR A TOKAMAK ENGINEERING TEST BREEDER

J.H. HUANG, G.Zh. SHENG, G.Q. LI, Ch.L. YOU, X.R. WANG,
H.W. SHI, Zh.Q. HUANG, Zh.Y. XIE, Y.X. YANG, L.Q. WU
Southwestern Institute of Physics,
Leshan, Sichuan,
China

Abstract

PRELIMINARY CONCEPTUAL DESIGN FOR A TOKAMAK ENGINEERING TEST BREEDER.

The outline of a preliminary conceptual design for a tokamak engineering test breeder is given. Some pertinent results in the areas of plasma physics, neutronics, thermohydraulics, magnets and reactor structure are described, and preliminary reactor parameters are presented.

1. GENERAL CONSIDERATIONS

A key issue in the development of nuclear energy in China is the supply of nuclear fuel. A programme has been launched to assess the potential of solving this problem by fusion breeders. One of the milestones along the path of this development would be building a tokamak engineering test breeder (TETB). It is expected that this breeder will be built during the first decades of the next century. This time schedule determines some features of the TETB. For example, the U–Pu fuel cycle is chosen because of the availability of the technologies in question. The structure of a tokamak reactor itself is rather complex, and the additional requirements of fuel breeding complicate the structure further. Hence, a self-cooling liquid tritium breeder blanket, which can alleviate this problem, is a preferable option. There is no fissile breeding at the inboard blanket, which results in a simpler blanket without significant sacrifice of the overall breeding performance. The size of the TETB should be as small as possible in order to reduce the cost. A tokamak reactor is, however, characterized by a rather large size. As a result, its fusion power should not be too small; otherwise, the power density in the blanket would be too low for valuable test results to be achieved. This TETB would produce about 200 kg Pu per year, sufficient to demonstrate the required breeding performance.

Among the liquid tritium breeders, liquid lithium is chosen because of its superior neutronics performance. On the other hand, the drop in MHD pressure is a major concern. Further theoretical and experimental research in this area is being carried out.