DISCUSSION

B.G. LOGAN: By what factor does steady-state current drive reduce the overall costs for your advanced design?

M. YOSHIKAWA: We have not really examined the cost of the reactor itself, and we can only guess at the difference in radius: it seems likely, however, that the radius would be reduced from 5 m to 4 m, so that the economic gain might be expressed as $5^2 - 4^2/5^2$, or about 35%.

S.O. DEAN: Do you think that a device on the scale of FER will be the next one built in Japan, or will it be something less ambitious? And do you think the next device will be built as an international project?

M. YOSHIKAWA: Except where D-T burn is concerned, we have made rather a large step with JT-60, which could be complemented by international co-operation. Thus, with parallel development of nuclear-related technologies, I personally feel we would do better to make a further large step, as represented by FER, rather than something like TFCX.

THE TOKAMAK FUSION CORE EXPERIMENT STUDIES

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Abstract

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The basic objective of the next major step in the US fusion programme has been defined as the achievement of ignition and long pulse equilibrium burn of a fusion plasma in the Tokamak Fusion Core Experiment (TFCX) device. Preconceptual design studies have seen completion of four candidate versions to provide the comparative information needed to narrow down the range of TFCX options before proceeding to the conceptual design phase. All four designs share the same objective and conform to common physics, engineering and costing criteria. The four base options considered differed mainly in the toroidal field coil design, two employing superconducting coils and the other two copper coils. In each case (copper and superconducting), one relatively conventional version was carried as well as a version employing more exotic toroidal field coil design assumptions. Sizes range from R = 2.6 m for the smaller of the two copper versions to R = 4.08 m for the larger superconducting option. In all cases, the plasma current was about 10 MA and the toroidal field about 4 T.

1. INTRODUCTION

Design studies have been performed over the last several years to establish the characteristics of those options that will satisfy the mission and objectives of the next large fusion device in the US tokamak program. This device is called the Tokamak Fusion Core Experiment (TFCX). The design studies have been performed by a design team with representatives from almost all major fusion laboratories of the US fusion program.

The mission established for TFCX is as follows. The essential objective of the TFCX is to achieve ignition and long-pulse equilibrium burn. To the extent that resources permit, the TFCX project should serve as a focus for the development of future fusion technologies.

Constraints related directly to the mission statement are: (1) the device should have an ignition margin of 1.5 under the assumed confinement scaling; (2) the device should be capable of at least 2×10^5 seconds of full parameter D-T burn; and (3) the device should have sufficient volt-seconds during the burn phase to replace 70% of the plasma internal inductance. The most important physics assumptions used were: (A) a modified form of GMS confinement scaling including the degradation of confinement as β approaches the ideal ballooning limit in an elongated, D-shaped, cross-section; (B) the assumed use of lower hybrid current drive to rampup the plasma current; (C) ICRH to supplement the heating available from lower hybrid power for heating to ignition; and (D) the use of a pumped limiter or a poloidal divertor for impurity exhaust and particle control, with the pumped limiter being the primary option. Engineering criteria were developed to ensure uniform assumptions regarding: (1) materials properties; (2) radiation resistance; and (3) design allowables. A cost data base was accumulated and cost algorithms developed which were then applied to each design.

Several candidate TFCX options have been developed that satisfy the mission. For each candidate option, a common basis has been used to develop the design. A common physics basis has been developed and applied. A set of design specifications was established and applied uniformly to each candidate option in performing the engineering design and analysis. Finally, a common cost data base was developed and applied consistently to each option. This process has provided a broad examination of the candidate TFCX options

on a consistent and uniform basis to permit a valid comparison of the results.

Trade studies were conducted to examine the sensitivity of the design to certain assumptions, in particular to variations in confinement scaling, to the use of full inductive drive instead of lower hybrid, and to the use of a divertor rather than a pumped limiter.

2. PHYSICS

The candidate TFCX options have been designed with common primary physics basis in the areas of ignition, pulse length, rf current drive, rf heating and impurity control. They can be characterized as follows:

The plasma current and toroidal field are chosen to provide a safety margin for ignition.

Current drive with an rf system at a frequency (several gigahertz) near the lower-hybrid range of frequencies (LHRF) will be used to provide most of the power to rampup the plasma current.

The LHRF will be augmented with rf heating near the ion-cyclotron resonance frequency (ICRF) to heat the plasma to ignition.

Both limiters and divertors have been analysed to determine their compatability with the TFCX designs. The baseline designs for TFCX feature the limiter option.

TFCX is designed for long-pulse (several hundred seconds or more) operation. The pulse length criterion will be discussed below.

In order that all options be comparable and consistent with mission requirements, a zero-dimensional physics model was developed to evaluate plasma performance. Key plasma parameters in this model that were fixed for preconceptual design studies are shown in Table I.

The elongation and triangularity of the plasma shape were chosen to take advantage of the improved beta values that can be attained with enhanced shaping while not

TABLE I. TFCX DEVICE PARAMETERS

Elongation (κ)	1.6
Triangularity (δ)	0.3
Safety Factor (q):	
on edge	2.4
on axis	1.0
Temperature (T)	10 keV
Ignition parameter (C_{ig})	1.5
Peak to average edge ripple	1.5 %

compromising plasma stability or the mechanical configuration of the device. The safety factor at the plasma edge of 2.4 was a compromise between plasma performance and stability. The plasma temperature of 10 keV was chosen to be near optimal for ignition and fusion burn. The ignition parameter (ratio of fusion heating rate to energy loss rate) is the most important measure of plasma performance with respect to the TFCX mission, and was set at 1.5 to provide some margin in sizing the various options with respect to the energy-confinement scaling discussed below.

The ignition parameter was determined by

$$C_{ig} = 0.295 \times \tau_{Eo} \times B_t^2$$

where B_t is the field-on-axis, and τ_{Eo} is the energy-confinement time at zero β (the total volume-averaged beta). The coefficient in the equation has already been adjusted to reflect a decrease in confinement time at ignition beta. The form used for the reduction in confinement with beta was $\tau = \tau_{Eo} \exp[-(\beta/\beta_c)^2]$. The β_c was determined from consideration of ideal MHD ballooning modes and given by

$$\beta_c = 0.2 \times (1 + \kappa^2)/(2 \times A \times q)$$

where A is the aspect ratio. For a fixed plasma shape and safety factor, the critical β depends solely on aspect

ratio. The energy confinement time at low beta was determined by a modified form of GMS:

$$\tau_{Eo} = 0.39 \times 10^{-6} \times a \times I_p$$

where a is the plasma radius, I_p is the plasma current, and the elongation has already been factored into this equation. Using these confinement and beta assumptions, INTOR would have an ignition margin of about 0.7. Assuming an INTOR design beta of nearly 6%, its ignition margin could be restored to about 1.5.

During current rampup the density will be maintained at a low value to facilitate rf current drive, and the electron temperature needs to be minimized to accommodate current profile evolution in a reasonable time-scale (~ 50 s). Some induction voltage will be applied by the poloidal field system due to the increase in the vertical field.

During the burn phase, the discharge will be maintained inductively while the profiles evolve to steady-state conditions in the plasma resistive time-scale. The inductive volt-seconds provided by the poloidal field (PF) system has been sized to allow for this evolution, namely to replace 70% of the plasma internal flux during burn. The actual burn time ranges from about 300 s to 600 s among the four design options.

3. DESIGN FEATURES

Four primary TFCX options were developed; many additional trade studies branching from these options were performed to examine numerous interesting technical design or performance questions. Two of the options use superconducting toroidal field (TF) coils and two options use copper TF coils. In all but the smallest copper option, superconducting poloidal field (PF) coils are also used. In the smallest copper TF option a copper central solenoid was used.

The two superconducting TF options are designated as nominal and high performance respectively. The nominal superconducting option is designed along traditional thinking as developed in the FED/INTOR design studies. This option provides sufficient shielding to limit the maximum nuclear heat load to the superconductor to 1 mW/cm^3. In order to reduce the overall size of the device, in the high performance superconducting option the maximum nuclear heat load to the superconductor was increased to 50 mW/cm^3. This allows the thickness of the shield to be reduced from about 60 cm in the nominal design to about 30 cm in the high performance design. The major radius decreases significantly as a result. The higher nuclear heat load complicates the design of the TF coil winding, because of the need to provide a high helium throughput. A feasible design concept was developed that uses a pancake wound coil with the first third of the pancake windings individually cooled, then the next third of the winding cooled every two turns, and the last third of the windings are cooled every five turns. (In the nominal design, the turns are cooled every five turns throughout.) The combined winding/cooling concept is feasible and the associated refrigeration requirements can be satisfied for the TFCX application since the duty factor is low (<3%). At high-duty factor operation the refrigeration requirement for such an option would be prohibitive; therefore, this coil cooling option is not attractive for reactor application.

For both superconducting options, the peak field at the coil is 10T. A forced-flow conductor design is used patterned after the Westinghouse conductor in the Large Coil Program (LCP) design. To achieve higher cavity winding current densities, changes in the conduit sheath material, the helium void fraction, and the winding/reacting process have been considered.

The configuration layout for each of the superconducting options is similar and based on the many design developments from the FED/INTOR studies of the last five years. The configuration features 16 TF coils, six superconducting PF coils all located external to the TF coils, intercoil support structure that provides access to the torus at the device midplane region, adequate external device shielding to permit hands-on capability external to the device, a single combined vacuum boundary between the plasma chamber and the magnet system, a plasma chamber vacuum boundary located outboard of the torus, separate structure for warm and cold components, and a simplified gravity support system. An elevation of the nominal superconducting option is given in Fig. 1.

The prescribed total DT burn time is 2×10^5 burn seconds. The integrated dose to the TF insulator is less

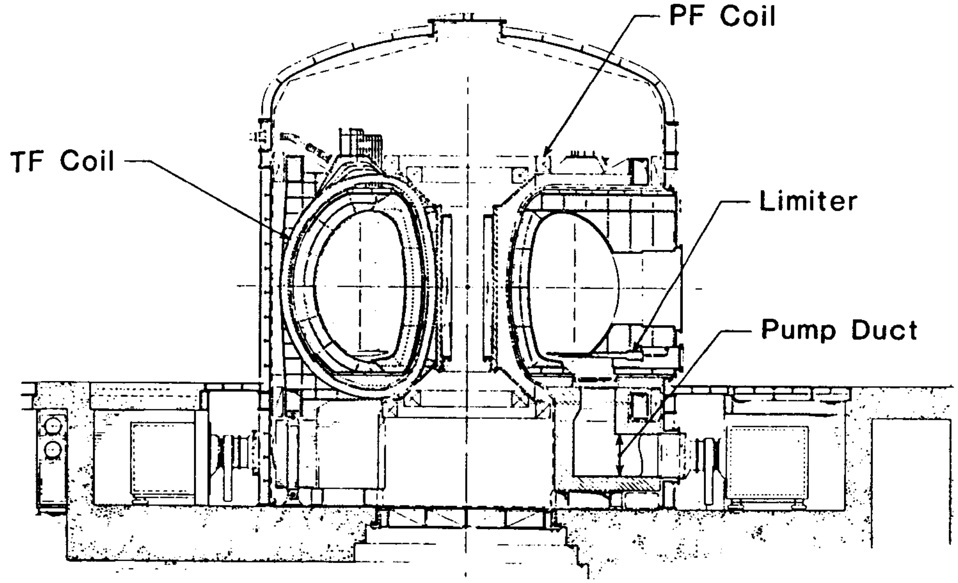

FIG.1. Nominal superconducting option: elevation.

than 2×10^7 rad for the nominal superconducting option, which is well below the allowable limit. For the high performance superconducting option, the total integrated dose to the TF insulator is about 6×10^8, which although higher than the nominal case is still a factor of four below the allowable established for the design.

For the nominal superconducting option, 32 MW of LHRF power is required to increase the plasma current to its full value of 11.2 MA in the prescribed 50 s for current rampup. Initially, 6 MW of ICRF power will be installed which, in conjunction with the LHRF power, should be able to heat a DT plasma to ignition. This power should be able to heat a DD plasma to approximately 50% of the full beta value. Upgrade of the heating power to a total of about 50 MW to allow full-beta operation in non-DT operation could easily be done at a later time. For the high performance superconducting option, the initial complement of RF is 26 MW of LHRF and 10 MW of ICRF. Later upgrade to a total RF power of 36 MW would permit full-beta non-DT operation.

The two copper TF options (also designated nominal and high-performance) are distinguished by the mechanical design, the configuration approach, and the materials used

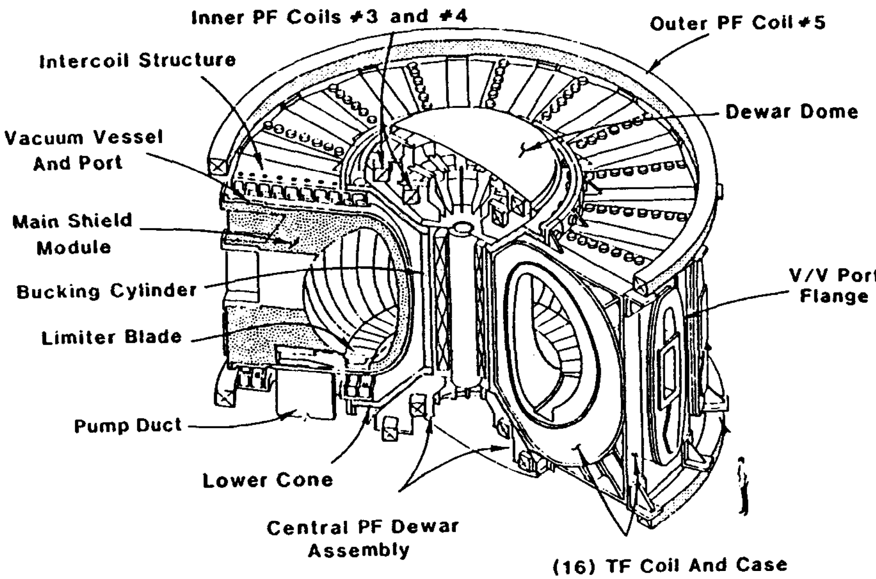

FIG. 2. TFCX machine configuration.

in the TF coils. The nominal copper option uses plate copper coils in a complete coil case. The conductor is made of oxygen-free, high conductivity (OFHC) copper. The TF coils center on a bucking cylinder. The configuration arrangement is similar to that used for the two superconducting options. An isometric of this option is given in Fig. 2. One difference in the configuration is that a copper solenoid is used; however, the remaining PF coils are superconducting.

The high performance copper option differs from the nominal option in several fundamental ways. The high performance copper option uses coils made from copper plates which, in the nose region, are high strength beryllium-copper alloy (50% IACS electrical conductivity, Cu-1.8% Ni - 0.4% Be, (C17510)) tapered to permit the coils to react the loads by wedging. No case or bucking cylinder is used in the nose region. The high performance TF configuration is shown in Figure 3. The size of this option is senstive to the strength and conductivity of the TF conductor alloy. The present alloy was selected because of the available data base and whose material properties are well established for larger size stock. The copper material in the outboard leg is OFHC copper with a "strong back" coil case. This

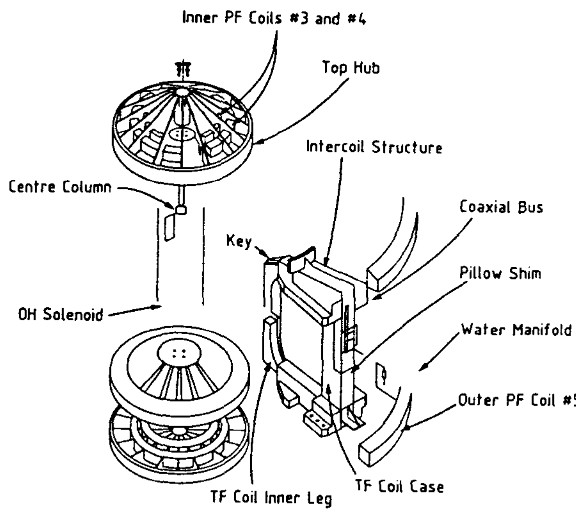

FIG. 3. High performance machine structural configuration.

approach minimizes the overall size of the device; however, it increases the complication of the TF coil and limits access which, in turn, complicates maintenance and repair.

The inboard shield for the copper TF options is based on the dose to the TF insulator and not on the instantaneous nuclear heat load as in the superconducting options. For low fluence devices such as TFCX, this results in much thinner shields and overall smaller devices. In the nominal copper option, the inboard shield is 12 cm thick (compared to the 30-60 cm in the superconducting options). The dose to the TF insulator is 1×10^{10} rad (the allowable dose). In the spirit of the high performance options, a higher allowable dose was specified, allowing the inboard shield to be smaller. A dose limit of 1×10^{11} rad was established and an inboard shield thickness of 1.5 cm was required.

In the nominal copper option, a cavity current density of 2740 A/cm^2 was used. To minimize the resistive power in the TF coils, the current density in the outboard leg was reduced to about 700 A/cm^2, resulting in a resistive power in the TF coils of 405 MW. In the high performance copper option the resistive TF coil power is 333 MW.

In the nominal copper option, 26 MW of LHRF power is required initially to rampup the plasma current. The

TABLE II. TFCX OPTION CHARACTERISTICS

Parameter	Superconducting Nominal	Superconducting Hi Perf.	Copper Nominal	Copper Hi Perf.
Major Radius (m)	4.08	3.61	3.35	2.60
Minor Radius (m)	1.52	1.30	1.30	1.04
Aspect Ratio	2.69	2.77	2.58	2.49
Field on Axis (T)	3.73	4.23	4.00	4.50
Inboard Shielding (m)	0.62	0.36	0.12	0.015
Fusion Power (MW)	267	270	229	197
Wall Load (MWm^{-2})	0.69	0.92	0.85	1.17
Plasma Current (MA)	11.2	10.5	10.9	10.4
Pulse Length (s)	618	452	458	298
LHRF Power (MW)	32	26	26	19
ICRF Power Initial/Final (MW)	6/31	10/36	7/28	7/26
TF/PF PWR (MW)	-	-	405/51	333/108
Operating Beta (%)	5.51	5.35	5.76	5.95
C_{ig} Mirnov	1.5	1.5	1.5	1.5

initial complement of ICRF is 6 MW of power. For the high performance copper option, the LHRF power requirement is 19 MW, and the initial complement of ICRF is 14 MW. Similar arguments to those for the superconducting options apply to provide the total RF power capability to achieve full-beta non-DT operation.

Selected key parameters for all four options are given in Table II.

4. COSTS

Cost estimates were developed for each of the four options on a consistent basis using a cost data base developed from past experience in the US fusion program. The cost for many systems and components were estimated on a per unit pricing (e.g. dollars per watt, dollars per kilogram, etc.). Costs for other systems were estimated on a "bottoms up" technique. Total costs, including R&D and contingency, and sited at an existing DoE installation, range from about $1 billion[1] for the high performance copper device, to a little over $1.3 billion for the nominal superconducting option.

[1] One billion = 10^9.

ACKNOWLEDGMENT

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DISCUSSION

W.M. LOMER (*Chairman*): Have you prepared relative cost estimates for superconducting as opposed to copper-coil machines?

J.A. SCHMIDT: The high-performance copper option was cheapest and was estimated to cost about US $850 million (1984) if constructed at an optimum existing site. The nominal superconducting TF option would cost US $1150 million (1984).