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NEXT TOKAMAK FACILITY*

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ABSTRACT

Design studies on a superconducting, long-pulse, current-driven, ignited tokamak, called the Toroidal Fusion Core Demonstration (TFCD), are being conducted by the Fusion Engineering Design Center (FEDC) and Princeton Plasma Physics Laboratory (PPPL) with additional broad community involvement. Options include the use of all-superconducting toroidal field (TF) coils, a superconducting-copper hybrid arrangement of TF coils, or all-copper TF coils. Only the first two options have been considered to date. The general feasibility of these approaches has been established with the goal of high performance (ignition, ~390 MW; wall loading ~2.2 MW/m^2) at minimum capital cost. The preconceptual effort will be completed in early FY 1984 and a selection made from the indicated options. The TFCD is judged to represent a reasonable necessary step between the Tokamak Fusion Test Reactor (TFTR) and the Engineering Test Reactor (ETR).

INTRODUCTION

Studies are being carried out at the Fusion Engineering Design Center (FEDC) and the Princeton Plasma Physics Laboratory (PPPL) to establish the characteristics of the next large tokamak experimental facility. The mission for this facility, as established by the Magnetic Fusion Advisory Committee (MFAC), is to provide a long-pulse, ignited demonstration of the plasma and technology core for a tokamak fusion reactor. The cost of this facility, to be called the Tokamak Fusion Care Demonstration (TFCD), will be minimized by eliminating the requirement for high-fluence nuclear testing. It has been judged that the nuclear testing role is best relegated to the Engineering Test Reactor (ETR) facility, following the development of the physics and supporting technology for a long-pulse ignition demonstration on TFCD.

Devices with the following TF coil options are being considered for the TDFC facility: (1) a copper-superconducting hybrid option. (2) an all-superconducting option, and (3) an all-copper option. A choice will be made following a preconceptual design for these candidates. These preconceptual design studies will be carried to a level that not only ensures design feasibility within the subsystem envelopes developed, but also gives acceptable confidence in the associated subsystem costs.

ISSUES

The TFCD, along with complementary, non-fusion facilities, must address the remaining issues requiring resolution before a final commitment is made to an ETR. The physics and technology issues to be addressed by TFCD will be those remaining after Doublet III Upgrade, Tokamak Fusion Test Reactor (TFTR), and Joint European Torus (JET) operation.

Most of the issues addressed by TFCD will be related to ignition and/or long-pulse operation. Information will be required on post-ignition confinement and burn control, on maintaining a high beta configuration for a period long compared with magnetic diffusion times, on impurity control and power/particle handling at long pulse and with a reactorlike core plasma, on maintaining steady-state or quasi-steady-state plasmas using noninductive current drive, and on ash accumulation and removal. Closely related technology issues are testing of an ion cyclotron range of frequencies (ICRF) launcher concept designed for the radiation environment, for tritium containment, and for maintainability and management of disruptive energy dumps at more nearly reactor-relevant levels (5-10 MA and >100 MJ).

Considerable research and development (R&D) is still required for large, reliable, superconducting magnets. Reactor-relevant TF coils must

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be employed on operating tokamaks; however, no existing or planned U.S. tokamak uses superconducting coils.

The next tokamak facility, along with all advanced fusion devices, must provide information on component and system reliability. Enlightenment on maintenance is likely to come exclusively from operating devices. Before a next tokamak, only TFTR must face up to remote maintenance, and, unfortunately, the TFTR configuration is unsuitable for reactor use. Thus, remote maintenance as well as component and system reliability will be important issues addressed by TFCD. Other systems-related issues are tritium management and the development of diagnostics, data acquisition, and control systems for the full fusion environment, including ignition and long-pulse burn.

The International Tokamak Reactor (INTOR) and Fusion Engineering Device (FED) studies have shown the adverse impact on the cost and schedule of a next tokamak when a comprehensive nuclear mission is specified. Therefore, a coordinated program using complementary nonfusion facilities will be employed to satisfy nuclear testing requirements. The role of the next tokamak facility in this program will be resolution of its assigned nuclear issues along with confirmation of the nonfusion testing.

OPTIONS

As indicated above, most of the issues to be resolved by the next tokamak facility are associated with ignition and long-pulse burn. Therefore, there are three preferred options for the next tokamak facility that can provide ignition and long-pulse burn capability. These three options are distinguished by the type of TF magnet employed: all-copper, all-superconductor, or a copper-superconductor combination.

Using all-copper TF coils removes the superconducting design requirement for shielding these coils, thus reducing size and complexity, and allows higher central fields, thus allowing an additional decrease in size for equivalent performance. In addition, the need for a superconducting coil development program, which is a critical path item for a superconducting device, is eliminated. Therefore, the overall schedule is shorter by about one year. The principal advantages of the all-copper device are the perceived simplicity, reliability, and a somewhat shorter schedule. The principal disadvantage arises directly from the advantages: the copper coils, the attendant simplicity, and the lack of an accompanying superconducting coil development program causes this option to be least relevant to the unresolved issues of an ETR.

The TFCD option with a pure superconducting TF coil is clearly relevant to future reactor development. However, substantial neutron shielding is required to reduce the nuclear heating in the superconducting coil even for low-fluence operation. In addition, the maximum reliable operating field realizable with superconducting coils is substantially less than for copper coils. This reduction in field results in an increase in size for a given ignition parameter. The increased plasma major radius and aspect ratio associated with superconducting coils incur a potential cost penalty relative to the copper coil option.

Using copper-superconductor combination TF coils provides a potentially attractive hybrid option. Copper TF coils located inside the superconducting TF coils provide shielding for the superconducting coils and higher central fields, resulting in smaller size and potentially lower cost than the all-superconductor option. This option's advantage over the all-superconductor approach is the higher central field at the same or lower cost. The advantage over the all-copper option is its relevance to the ETR issues: it does develop superconducting coils and its design is based on these coils. Disadvantages relative to the all-copper option are a somewhat longer schedule and less reliability due to more complexity.

DESIGN

Physics Requirements

The overall mission for TFCD dictates that the physics requirements for the device be compatible with a long-pulse, ignited discharge. The ignition parameter is dependent on the energy confinement and maximum operating beta. During initial studies we have assumed conventional plasma shapes (D shapes) and associated modest betas. Parallel studies are being carried out to develop a more advanced (higher beta) shape compatible with a realistic poloidal field (PF) design. These advanced shapes will be integrated into the design during the next phase of the studies.

Experiments on present devices indicate a soft beta limit, which will provide a stable operating point during burn. However, some method of varying the operating point, such as varying plasma current, ripple, or impurities, may be required.

A single-null poloidal divertor configuration was developed as part of the INTOR design studies with minimum impact on the PF coil design and mechanical configuration. This configuration can be accommodated within about the same envelope required for a pumped limiter placed at the bottom of the plasma chamber. Due to the reasonable compatibility of these two configurations, we have based the TFCD preconceptual design studies and the associated option selection on a pumped limiter design. Ongoing divertor and limiter experiments will form the basis for selecting the impurity control option during the conceptual design.

Current drive and ramp up at low density with rf heating near the lower hybrid frequency (LHRH) has been successful in several recent tokamak experiments. This technique will be used with a modest ohmic heating (OH) voltage to ra_ the plasma current during startup. This will alleviate the inductive volt-second requirement for startup and conserve the volt-seconds for use during the burn. The requirement is for a 300-s ignited burn. About 10 MW of LHRH appears sufficient for current drive purposes. The remaining 20 MW required for heating to ignition is provided with ICRF heating. A modest amount (0.1-1 MW) of electron cyclotron resonant heating (ECRH) may be required for preionization.

The basic parameters for the superconducting-copper hybrid TF coil option are listed in Table 1. The high central field (7 T) provided by the hybrid TF coil facilitates the realization of a compact device with high neutron wall loading.

Table 1. DCT-8 parameter list.

Parameter	Units	DT Sc & Cu
Minor radius, a	m	0.75
Major radius, R	m	3.60
Scrapeoff		
Inboard	m	0.13
Outboard	m	0.07
Elongation, κ		1.6
Triangularity, δ		0.3
Aspect ratio, A		4.0
Average ion temperature $\langle T_i \rangle$	keV	13.0
Average electron temperature $\langle T_e \rangle$	keV	13.0
Average electron density, $\langle n_e \rangle$	10^{20} m^{-3}	1.90
Field on axis, B_t	T	7.0
Plasma current, I_p	MA	5.9
Safety factor, q_ψ		2.0
Beta, β		0.04
Poloidal beta, β_p		1.5
Energy confinement time, τ_E	s	1.0
Plasma edge neutron wall loading	MW/m^2	2.2
Fusion power	MW	390.0
Ignition parameter		1.0

Engineering Design

Effort to date has concentrated on developing preconceptual designs to demonstrate the engineering feasibility of the candidate options. Primary effort has focused on the development of a viable superconductor-copper hybrid TF coil option as well as the development of an all-superconducting TF coil option. Although an all-copper TF coil option may be a viable candidate, no preconceptual design effort has been performed.

Configuration Options

Five candidate options were selected for study:

1. An all-superconducting TF coil configuration.

2. A hybrid superconducting-copper TF coil system with water-cooled copper coils located in the shield region in both the plane of the TF coils and the removable torus sector between the TF coils.

3. A hybrid superconducting-copper TF coil system with water-cooled coils located in the shield region only in the plane of the TF coils. The removable torus sectors between the TF coils contain no copper coils.

4. A hybrid superconducting-copper TF coil system with water-cooled copper coils located in the shield region only in the removable torus sector between the TF coils. No copper coils are located in the plane of the TF coils.

5. A hybrid superconducting-copper TF coil system with cryogenically cooled copper coils located within the bore of the superconducting coils in the same vacuum region.

Figure 1 schematically indicates the locations of the copper coils and the superconducting coils for Options 2 through 5.

Option 5, generally speaking, is derived from Option 1. Although it offers some flexibility for using the available magnetic field, its estimated cost appears larger than that of Option 1. It has been concluded that such an approach would be feasible because of the similarities to Option 1, so limited effort was devoted to this option. Option 2 appears to be more complicated than either Option 3 or 4, so it has received little attention. The bulk of the preconceptual design effort on a hybrid TF

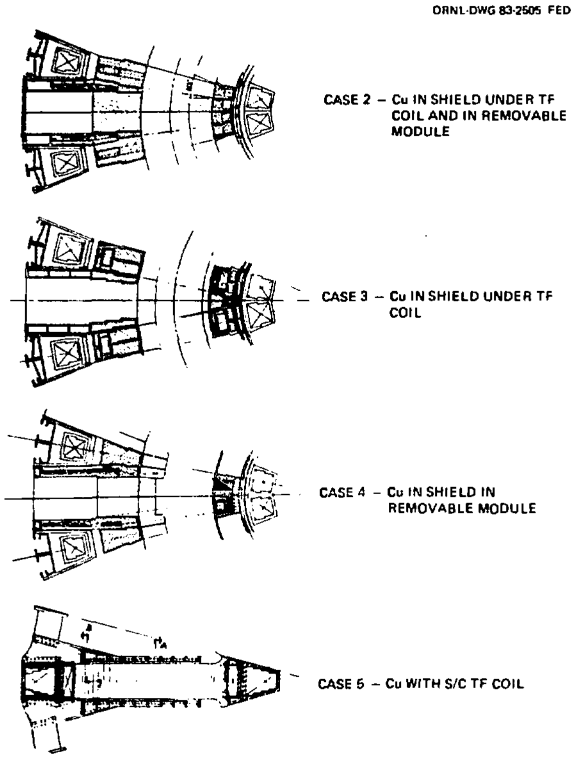

Fig. 1. TFCD hybrid configuration options.

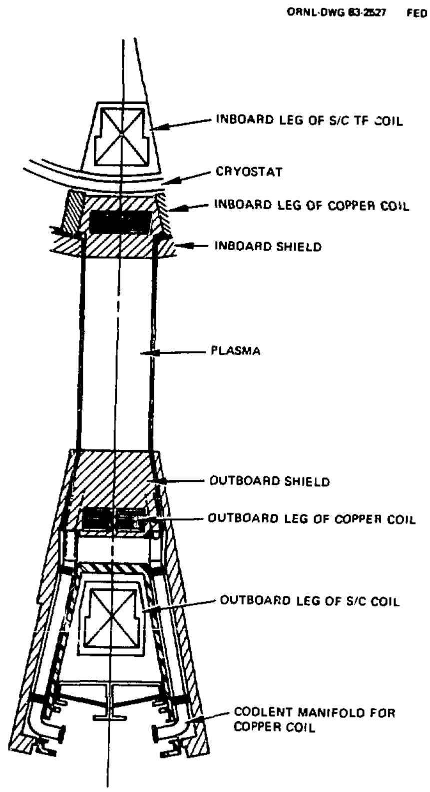

Fig. 2. TFCD hybrid Option 3 radial arrangement of coils.

option has been on Options 3 and 4. Feasible approaches have been identified to incorporate the copper coils into the shield, to locate the coolant and electrical leads, to accommodate the magnetically induced loads, and to provide for access and maintenance. Figure 2 shows some of the details of the radial build arrangement for Option 3.

Other configuration features incorporated into the design are those developed in recent reactor studies, such as FED and INTOR. These features include:

- Relatively few superconducting TF coils (designs with 12 to 16 coils are being assessed; since the major radius of these designs is smaller than for the FED/INTOR class, the edge ripple considerations tend toward a slightly larger number of TF coils).

- A torus consisting of modular sectors radially inserted between the outboard legs of the TF coils.

- All PF coils located external to the TF coil bore and accessible.

- A single combined vacuum boundary between the plasma chamber and the superconducting TF coils.

- Horizontal access at the midplane of the device.

- Simplified gravity support for the torus shield and vacuum system.

TF Coils

A maximum field at the superconductor of up to 12 T is being considered. However, system trade studies suggest that the minimum cost device may be obtained for a maximum field at about 10 T. The following three candidate coil technologies, each requiring development, are being considered:

- NbTi pool-boiled and superfluid-cooled to 1.8 K.
- NbTi forced-flow subcooled to 3 K.
- Nb_3Sn/NbTi combination cooled to 4.2 K.

A design using the second option was adopted in the earlier FED Baseline design. Each option is being examined in the present design effort. The winding pack current densities appropriate in these three options are 1600, 2000, and 2400 A/cm^2 (in the Nb_3Sn region), respectively. The present design studies are employing a configuration with 16 TF coils each with a clear bore of about 3.6 by 5.2 m.

The copper insert coils (in Options 3 and 4) are located in the shield region. They are water cooled with a clear bore of about 3.0 by 3.9 m. A current density of about 1400 A/cm^2 is employed. The copper resistive power is on the order of 150 MW. Design challenges associated with these coils include (1) incorporation of the coils into the shield to accommodate separate cooling circuits, (2) providing manifolding to get the coolant lines in and out, (3) providing electrical lead paths, and (4) designing the total torus system to react the centering loads and overturning moments imposed on the torus sectors. Viable solutions have been obtained in these challenging areas for Options 3 and 4.

PF Coils

The PF coil system consists of a superconducting (NbTi) solenoid complemented with superconducting (NbTi) ring coils located outside the bore of the TF coils. The coils will be located and sized to permit operation with a plasma elongation of 1.6 and a triangularity of 0.3. The design features of both the OH solenoid and the equilibrium field (EF) coils will be patterned after the designs used in FED.

Torus

The torus consists of removable torus sectors inserted radially between the outboard legs of the TF coil in a fashion similar to that used in FED and INTOR. Because the overall machine is significantly smaller than FED, it is not possible to have the torus composed of 16 equal-size sectors (equal to the number of TF coils). In the options being considered, the torus design requires additional sectors located in the plane of the superconducting TF coils. A viable structure design appears feasible for each option, even though copper coils are located in some of the torus sectors in the hybrid options.

The inboard bulk shield of the torus will be sized to limit the instantaneous nuclear heat to the first layer of superconductor to about 0.3 MW/cm^3. For the hybrid options, which has a nominal neutron wall loading of about 2.2 MW/m^2, the required inboard shield will be about 80 cm. The total outboard shielding required will be about 120 cm and is determined by the limiting dose rate for contact maintenance (~2.5 mrem/h 24 h after shutdown).

Impurity Control

A pumped limiter is employed for impurity control. It forms a toroidal belt at the bottom of the plasma chamber. Details of the design remain to be worked out. However, the design approach will be that used for FED, namely, a single-edge, flat plate design. The replaceable protective surface materials under consideration include Be, BeO, and SiC with the possibility of Ta or W used at the leading edge. The limiter will be divided into 32 removable segments, two in each torus sector of the device. The limiter segments will be removable independent from the torus sectors. The design configuration also has the flexibility to accommodate the use of a poloidal divertor without major changes in the overall configuration.

Remaining Systems

Design of the remaining systems and components for TFCD will follow approaches used in FED/INTOR. No difficulty in the design of these systems is anticipated, and this design effort will be performed after the selection of a TF coil option (superconducting, superconducting-copper hybrid, or copper).

CONCLUSIONS AND PLANS

Given that a primary objective of the magnetic fusion program is to achieve successful operation of an ETR during the late 1990s, it is clear that there are many issues to be resolved in the early 1990s, issues that will not be resolved by TFTR, Doublet III Upgrade, or their contemporaries. Examination of the issues indicates most are associated with ignition and long-pulse burn. The necessary, logical step on the path toward a ETR is then a tokamak capable of ignition and long-pulse burn. Three options for such a tokamak are delineated by the type of TF magnet: all-superconductor, all-copper, or a copper-superconductor combination. The all-copper option is least relevant, while the combination hybrid copper-superconducting version may offer an attractive combination of cost and relevance.

Preconceptual design work on the three options is now in progress. The bulk of the design is being done at the FEDC, while coordinated activity is under way at PPPL, the Massachusetts Insitute of Technology, General Atomic Technologies, and Argonne National Laboratory. In the spring of 1984, a choice between the competing options can be made and a conceptual design effort initiated.