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ABSTRACT

Mirror experiments have led the way in applying superconductivity to fusion research because of unique requirements for high and steady magnetic fields. The first significant applications were Baseball II at LLNL[1] and IMP at ORNL.[2] More recently, the MFTF-B yin-yang coil[3] was successfully tested[4] and the entire tandem configuration is nearing completion. Tokamak magnets have also enjoyed recent success with the large coil project tests at ORNL,[5] preceded by single coil tests in Japan and Germany. In the USSR, the T-7 Tokamak has been operational for many years and the T-15 Tokamak is under construction, with the TF coils nearing completion. Also the Tore Supra is being built in France.

RECENT PROGRESS

During 1982, the 375-ton yin-yang magnet for MFTF-B was tested to the full field of 7.7 T, as shown in Table I, and a complete tandem mirror MFTF-B facility is nearly complete.[6] This MFTF-B magnet system will provide the

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field environment necessary for tandem mirror plasma physics investigations with thermal barriers. There will be superconducting coils consisting of 2 sets of stretched, yin-yang pairs; 12 central cell solenoids; 2 transition coils between the solenoid and yin-yang coils; and 2 axicell coils at each end of the coil system. The overall length of the magnet system will be about 52 m, the mean diameter of the solenoids being 5 m, and the total stored magnetic energy about 2000 MJ.

The conductor for the yin-yang and transition coils consists of a 6.5 mm-square, copper-stabilized, niobium-titanium composite wrapped in an embossed and perforated sheath of high purity copper. The core-to-sheath bond had to be improved by replacing the original 90/10 Pb-Sn solder with 50/50 Pb-Sn solder for improved wetability at lower bonding temperature. Once initial manufacturing difficulties were resolved, both quality and production efficiency were good to meet the parameters in Table II.

Further progress has been made for tokamak magnets with the IAEA large coil task in which the U.S., Euratom, Japan, and Switzerland agreed to cooperate in the development of superconducting tokamak magnets. Oak Ridge National Laboratory prepared specifications for test coils[7] and contracted with three U.S. industries to construct coils of their own design. In 1978 Euratom, Japan, and Switzerland officially joined the collaboration. Each contributor used a different conductor and coil design to meet the 8-T peak field, with tolerance for pulsed external fields up to 0.4 T, as summarized in Table III. Each coil has a D-shape of 2.5 x 3.5 meter bore and contains 7 MA-turns of conductor. Successful single coil tests have been performed by Japan first and then at KfK by Euratom. The Japan and U.S. General Dynamics coil were then successfully tested in the Large Coil Test Facility at ORNL and a complete six coil test array is nearly ready.

FUTURE DEVELOPMENTS

To date, no superconducting fusion magnets have had to withstand the damage of 14 MeV neutrons and other restraints imposed on a commercial reactor. However, as a strategic goal, the Mirror Advanced Reactor Study (MARS) was performed to conceptualize (Fig. 1) a commercial fusion reactor producing electricity and synthetic fuels.[8] The MARS central cell is ignited by alpha heating, while electron-cyclotron resonance heating and negative ion beams maintain electrostatic plasma confining potentials in the end plugs. Very high (24 T) choke coil magnets are placed between the central solenoid and end cell yin-yang magnets. (Plug injection power is reduced almost linearly with increasing choke coil fields). The anchor and plug coils listed in Table IV are similar in design to the MFTF-B yin-yangs but are larger and have a higher peak field on the conductor. Niobium titanium was used because of its ductility for the difficult winding geometry, but the coolant is superfluid LHe II to allow high current density at 10 T.

The greatest extrapolation from present technology is in the high-field choke coil. To create a 24-T field on axis, a superconducting background coil operating at 15.8 T is with a normal conducting insert that provides an additional 8 T on axis and operates at 24.8 T. The high field is, of course, accompanied by high stresses. These were accommodated using a Nitrónic-40 steel-reinforced conductor that limits strain to 0.35%. To obtain sufficient current density in the superconductor at the high fields, superfluid LHe II at 1.8 K is used as the coolant. This also allows an increase in the heat flux, which results in about a factor of 2 increase in bulk current density.

The copper insert coil must be designed to minimize power consumption (both ohmic heating and coolant pumping) while producing the required field. It must have long life in the high radiation field, and be easily replaced. The baseline design is a double pancake that is machined from a single plate of cold-worked copper-allow MZC, which has a working stress of 331 MPa and an acceptably low resistivity. Spinel ($MgAl_2O_4$) is the insulator because of its low rate of isotropic, neutron-induced swelling.

To enhance credibility, relatively conservative design criteria were used in MARS. However, this lead to larger magnet sizes than necessary, such that magnet was 41% of the total reactor plant equipment cost.

Reduction of the the yin-yang coil mass by a factor of 3 is possible by designing radiation-hardened magnets, as discussed in the Fusion Power Demonstration (FPD) report.[9] The minor radius of the yin-yang coil strongly controls the coil sizes and field efficiency. Thus, by reducing the neutron shielding in FPD to 35 cm, corresponding to a coil minor radius of 0.8 m (compared to 1.1 m in MARS and 35 cm, corresponding to a coil minor radius of 01.8 m (compared to 1.1 m in MARS and 0. 75 m in MFTF-B), we were to decrease the coil mass and cost substantially. Of course, traditional design limits on epoxy glass insulators must be raised from 10^9 rads to approach 10^{11} rads with polyimide binders and woven glass cloths. Similarly, the damage to the copper stabilizer must be accommodated by deviating from traditional full-cryogenic stability. This may not be too serious, since upon initial charging the magnet would be fully cryostable. Then as neutron damage increased the copper resistivity, the stability would make a transition to cold-end recovery criteria and perhaps become metastable. Eventually, after a few years the magnets would be warmed to room temperature, at which time most of the neutron damage would be annealed out prior to subsequent cooling and recharging.

An alternative technology using internally cooled conductors is available as well. Here stability depends mostly of the heat capacity of the helium rather than the resistivity of the copper, so that stability is not lost during extended neutron irradiation. However, it may be difficult to remove nuclear heating in the coil except with short channels, forced flow, and possibly niobium-tin superconductors. Ultimately, the nuclear heating in the coil winding should be the determining factor for neutron shielding (assuming technology advances have solved the insulation and copper resistance problems).

Similar radiation hardened advantages can be found in tokamaks. Recent studies of the Tokamak Fusion Coil Experiment (TFCX) have helped to demonstrate that much smaller tokamak sizes are available when departing from more conventional designs. The major radius can be readily reduced from the 4 to 6-m size to 3.6 m. As part of a continuing effort by the U.S. DOE Office of Fusion Energy to define an ignition experiment, a superconducting tokamak has been designed with thin neutron shielding and aggressive magnet and plasma parameters. The Tokamak Ignition/Burn Experimental Research device (TIBER) has been designed to represent the smallest superconducting ignition experiment consistent with the DOE/OFE Mission II physics and engineering objective.[10] In order to reduce the size and cost of the tokamak, more aggressive design assumptions were necessary. Plasma shaping is used to achieve higher beta, and neutron shielding is minimized. However, the superconducting TF magnets must be sufficiently shielded to reduce neutron heat load to the coolant, neutron fluence to the superconductor and stabilizer, and gamma doses to the insulator. In particular, the insulation must retain adequate strength and electrical properties after irradiation to end-of-life fluences above 10^{19} n/cm^2 and gamma dosages above 10^{10} rads, especially in those portions of the magnet adjacent to shield-penetrations for diagnostics on plasma-heating systems.

Non-interlinking TF and PF coils are used in TIBER for easy maintenance. Also, the entire tokamak is enclosed in a single vacuum vessel, similar to the practice for the Mirror Fusion Test Facility. In this way internal dimensions and complications are minimized, lending to easily serviced external vacuum joints for rapid disassembly.

A point design of the TIBER device is summarized in Table V. The design depends upon radiation-hardened magnets with thin neutron shielding and shaped high-beta plasmas. The peak nuclear heating rate in the superconducting TF coil is 8 milliwatts per cubic centimeter, resulting in total system heat load of 50 kilowatts. Both the TF and PF coils would be constructed of force-cooled, niobium tin, to achieve high-current density with acceptable nuclear heat removal and neutron damage limits.

A high-field (15-T) pusher coil is used in the normal position for the ohmic heating coil. By effecting a modest plasma indentation, it achieves a beta of 10 percent. All inner-leg components of the tokamak must be kept as small as possible so that the plasma center line can be minimized and plasma shaping maximized. Accordingly, magnet current densities of 4 KA/cm^2 and an integrated structural design are necessary. The neutron shielding of the inner leg is only 45 cm.

It is envisioned that fast-wave, lower-hybrid heating would be used to induce a plasma current above 10 million amperes at low-plasma density around 0.4×10^{14} ions/cm^3. Then a laser-driven pellet fuel injector would raise the density to 3×10^{14}, with continued fast and slow-wave lower hybrid heating to achieve ignition. Because the current drive is not efficient at high-plasma density, the current would decay with a time constant of 500 seconds. During this period all plasma heating would be turned off in order to study pure ignition physics. As the plasma current decays to 7 million amp and density drops to 10^{14} ions/cm^3 the slow-wave, lower-hybrid would again be used to sustain a steady-state plasma with a wall loading of 0.8 MW/m^2 and a plasma Q of about 10. Thus, the TIBER device can not only study ignition physics, but can also operate in a steady-state, current-driven mode to study plasma/wall interactions, helium ash removal, and numerous neutron damage effects.

It is apparent that not only has superconductivity for fusion research advanced markedly over the past decade, but aggressive superconducting magnet design will result in more affordable future steps in the fusion program.

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Table I. MFTF yin-yang magnet parameters as tested in 1982.

Maximum field, T	7.68
Central field, T	2.0
Mirror ratio	2.1:1
Mirror-to-mirror length, m	3.6
Major radius (mean), m	2.5
Minor radius (mean), m	0.75
Current, A	5775
Turns	1392
Stored energy, MJ	409
Conductor current density, $A \cdot cm^{-2}$	3729
Coil current density, $A \cdot cm^{-2}$	0.19
Conductor length, km	50
Total weight, kg	341,000

Table II. MFTF conductor specifications for yin yang and transition coils.

Conductor parameters	Value
Superconductor:	
Critical current, kA at 7.5 T, 4.2 K	10
Copper to superconductor	1.7:1
Number of filaments	480
Filament diameter, mm	0.20
Twist pitch, mm	180
Conductor-resistance ratio	150:1
Core size, mm	6.5 x 6.5
Stabilized conductor:	
Maximum conductor field, T	7.68
Maximum conductor current, A	5775
Conductor operating temperature, K	4.5
Overall: copper to superconductor ratio	6.7:1
Stabilizer copper-resistance ratio	220:1
Copper resistance (at 7.68 T, 4.5 K), n$\Omega \cdot$cm	46
Helium-cooled surface area, $cm^2 \cdot cm^{-1}$	8.17
Required heat transfer rate, $W \cdot cm^{-2}$	0.19
Overall size, mm	12.4 x 12.4

Table III. Coil parameters in the large coil task.[5]

Designer/ manufacturer	General Dynamics	General Electric	Westinghouse Electric	Euratom (Siemens)	Japan (Hitachi)	Switzerland (Brown Boveri)
Superconductor	NbTi	NbTi	Nb_3Sn	NbTi	NbTi	NbTi
Configuration	Flat cable in finned bar	Subelements around core	Cable in conduit	Subelements around core, in conduit	Flat cable in roughened bar	Solder-filled cable with central tube
Current	10,200 A	10,400 A	17,800 A	11,000 A	10,200 A	13,000 A
Helium	Boiling pool 4.2 K, 1 atm	Boiling pool 4.2 K, 1 atm	Forced flow, 3.7 K, 15 atm	Forced flow, 3.7 K, 15 atm	Boiling pool 4.2 K, 1 atm	Forced flow, 3.7 K, 15 atm
Winding	Edge wound in layers	Flat wound in spirals	In spiral grooves	Flat wound in spirals, potted	Edge wound in spirals	Wound in spirals, potted
Structure	Type 304L SS welded case	Type 316LN SS bolted, welded case	3319-T87 Al plates, bolted	Type 316LN SS bolted, welded case	Type 304L SS bolted, welded case	Type 316LN SS bolted case

Table IV. MARS magnet dimensions and fields.

Coil	Mean diameter (m)	Central field (T)	Peak field inner radius (T) Major	Minor	Superconductor
Central cell	5.42	4.7	7.2		NbTi
Choke-background	3.8	24	16.1		Nb_3Sn:Ti, Nb_3Sn, NbTi
Choke-insert	1.2	24	24		Copper (normal-conducting)
Transition	2.2, 6.5	4.1/7	9.5	2.5	NbTi
Anchor	2.2, 5.0	7.3/3.9/ 7.3	9.9 9.7	10.5 10.7	NbTi NbTi
Plug	2.2, 6.5	7.2/3.0/ 7.5	9.2 9.7	9.8 9.8	NbTi NbTi
Recircularizer					
C coil	2.2, 5.0	7.5	9.8	9.5	NbTi
Solenoid	2.2	--	6.5		NbTi

Table V. TIBER Point design parameters.

Major plasma radius	2.60 meters
Minor plasma radius	0.73 meters
Elongation	1.94
Triangularity	0.60
Indentation	0.05
Average beta	10 percent
Safety factor	3.80
Plasma current	6.6×10^6 amps
TF coil (Nb-Ti)	10 T
PF coil (Nb_3Sn)	15 T
Nuclear heating rate	5 mW/cc
Fusion power	194 megawatts
Current drive power	19 megawatts
Plasma Q	10

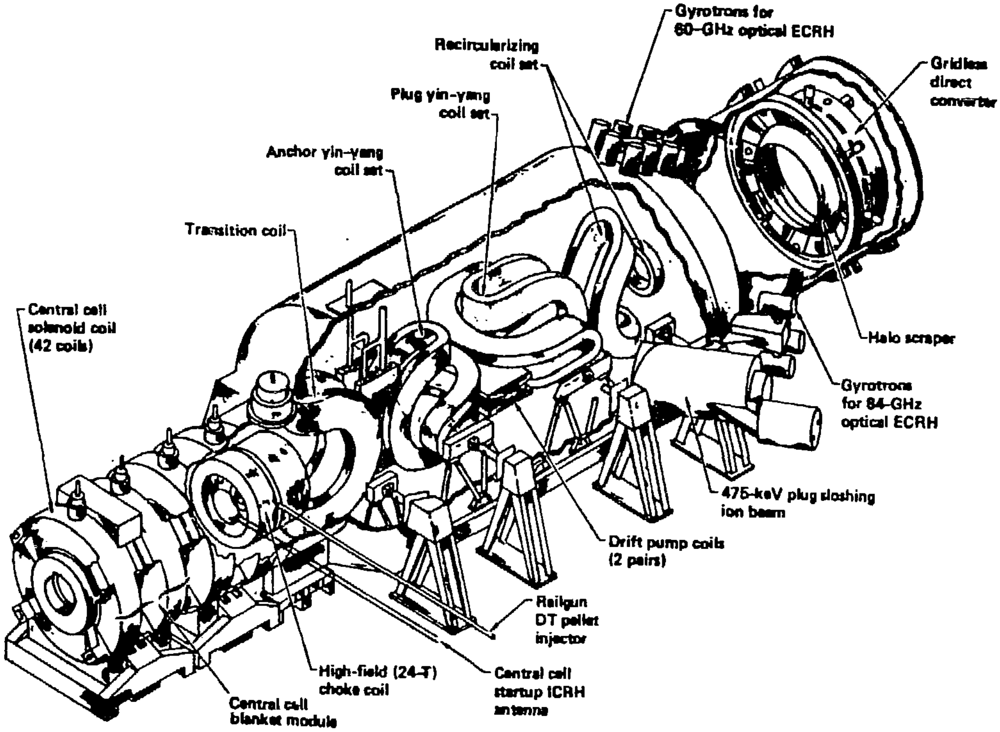

Fig. 1. View of one end of the MARS Reactor.