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SUPERCONDUCTING MAGNETS FOR TOROIDAL FUSION REACTORS*

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Summary

Fusion reactors will soon be employing superconducting magnets to confine plasma in which deuterium and tritium (D-T) are fused to produce usable energy. At present there is one small confinement experiment with superconducting toroidal field (TF) coils: Tokamak 7 (T-7), in the USSR, which operates at 4 T. By 1983, six different 2.5×3.5 -m D-shaped coils from six manufacturers in four countries will be assembled in a toroidal array in the Large Coil Test Facility (LCTF) at Oak Ridge National Laboratory (ORNL) for testing at fields up to 8 T. Soon afterwards ELMO Bumpy Torus (EBT-P) will begin operation at Oak Ridge with superconducting TF coils. At the same time there will be tokamaks with superconducting TF coils 2 to 3 m in diameter in the USSR and France. Toroidal field strength in these machines will range from 6-9 T. NbTi and Nb₃Sn, both cooling and forced flow, cryostable and metastable - various designs are being tried in this period when this new application of superconductivity is growing and maturing.

Introduction

Nuclear fusion has aptly been called "starfire" because this reaction is the source of the energy being radiated from our sun and every other star. Deuterium and tritium, colliding at velocities above 500,000 m/s, fuse together, and part of their mass is converted into energy (about 18 meV per event). In the sun, forces of gravity many times greater than that on the earth hold the reacting mass together. On earth, the chief hope for practical "man-made solar energy" depends upon the use of strong magnetic fields to effect the confinement. Magnetic fusion reactors must confine plasma (deuterium and tritium ions and free electrons) at temperatures around 100,000,000°C and pressures in the megapascal range for substantial periods of time. This feat requires magnetic coils that have openings several meters across and that produce steady fields of 8-12 T. If fusion energy is to become a practical source of electrical power, these magnets must be superconducting, they must be economical, and they must be extremely reliable.

In a half-dozen major countries around the world, intensive development of superconducting magnets for fusion reactors has been under way for the past 5-10 years. Substantial progress has been made, supporting confidence that superconducting coils will be reliable components of the fusion power reactors that could begin operation within another 10-12 years.

The fusion reactor concept that is generally regarded as most advanced at this time is based on the tokamak scheme of confinement.² The other leading concept in the US is the tandem mirror.³ An alternate concept being pursued at a significant level in the US and Japan is the EBT.⁴ This paper describes development of coils for the toroidal machines (tokamak and EBT). Paper HA-1 will discuss magnets for mirror machines.

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Tokamak Coil Systems

In a tokamak, plasma confinement is achieved by a combination of three different magnetic fields: the toroidal field, the poloidal field, and the equilibrium field.⁵ The major component is the toroidal field, produced by a set of coils surrounding a toroidal vacuum chamber (Fig. 1). In operation, the chamber is evacuated, a controlled amount of gas is admitted and then ionized, and a current is induced in the ring of plasma thus created. The poloidal field encircling the plasma ring is produced by the plasma current. The equilibrium field, produced by currents in windings that parallel the plasma, resists the tendency for the major diameter of the plasma ring to grow.

Fig. 1. Tokamak magnet systems.

Some tokamaks include windings near the plasma to shape the current profile. Divertors, which skim off plasma from the surface to control impurities, require special coils, also near the plasma.

The tokamak is basically a pulsed device in which the plasma current must be induced by making the plasma the one-turn secondary of a transformer. In early tokamaks the primary induction coils surrounded an iron core that ran through the center of the torus, but in some tokamaks the iron is omitted (as shown in Fig. 1). The capability of the induction system, measured in volt-seconds, is the integral of the voltage around the plasma ring and its duration.

In a tokamak, magnetohydrodynamic (MHD) instabilities in the plasma particle trajectories and fields cause loss of confinement when the ratio of the plasma pressure to the equivalent pressure of the toroidal field at the plasma exceeds a particular value that depends on machine design and operating conditions. This ratio, usually represented as beta (β), is obviously an extremely important parameter, because the toroidal magnetic field needed to confine plasma at the temperature and density required for D-T ignition varies inversely with the square root of β . It appears that in tokamak power reactors, β values in the range of 4-6% can be assured, requiring the field at the plasma to be in the range of 5-6 T. The toroidal field (which falls off inversely with major radius) must be 10-12 T at the TF coil windings to produce this field strength at the plasma centerline.

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The toroidal field can be steady (with resistive TF coils it is pulsed to reduce cooling and power supply requirements). The induction and equilibrium field windings, on the other hand, must be pulsed. This will produce field transients at the TF coil windings on the order of 0.1-1.0 T in 1-10 s, depending upon the location of the windings and the mode of plasma startup. Disruption of the plasma will produce a more rapid field transient at the TF windings. The magnitude and rate depend upon machine configuration; a typical value for a tokamak reactor may be about 0.5 T with a time constant of 0.1 s.

Another significant aspect of the interaction of the equilibrium fields and the TF coils is the over-turning moment on the TF coils; this requires intercoil structure and careful design of the winding to prevent conductor motion.

Superconducting Tokamaks

To date, with but one exception, all tokamaks have used copper coils cooled either by water or liquid nitrogen. The only tokamak so far to use superconducting coils is T-7 in Moscow.⁶ Another, also in the USSR, is in the early stages of construction. In France, construction of Torus II Supra is expected to start soon. The first superconducting tokamak to be built in the US will most likely be a much larger, D-T burning tokamak, which is now in the preconceptual design stage.

T-7 and T-15

T-7 was designed and built at the Kurchatov Institute, primarily as a confinement experiment rather than as a magnet technology test. It uses superconducting TF coils to permit a higher pulse repetition rate than could be achieved in the copper coil device (T-4) that it replaced. In the TF magnet there are 48 coils, having an 0.85-m circular bore. The TF coil conductor consists of a flat array of 9 copper tubes, each with 2-mm ID, with multifilamentary NbTi-Cu strands laid between tubes; the whole is covered and joined by electrodeposited copper. Cooling is by a forced flow of helium at the saturation point so that heat input results in vapor formation rather than temperature rise. Toroidal copper shells inside and outside the TF coils prevent field transients due to the pulsed poloidal field (PF) coils from producing significant heating in the TF coils. The PF coils are water-cooled copper.

The TF coil set for T-7 was tested in 1978 at currents up to 4800 A and a peak field of 4 T. It appears that this performance is considered adequate to meet the needs of the plasma confinement experiments now under way, and there are no immediate plans to try to reach the nominal design point of 6000 A, 5 T.

Tokamak-15 (T-15) (which appeared first in an earlier 5-year plan as "reconstructed T-10") will be built at the Kurchatov Institute. Magnet systems design is by a team at the Efremov Institute, with criteria and conductor development input from scientists at the Kurchatov. After considering 2.1×3.3 -m D-shaped TF coils using either NbTi or Nb₃Sn conductor, T-15 designers finally settled on a set of 24 circular coils, with a 2.2-m bore diameter, using Nb₃Sn conductor at a current of ~8000 A and a peak field of 6 T. As in T-7, the conductor is cooled by forced flow of two-phase helium. In the T-15 conductor, however, there are only two tubes, one on either edge of a rectangular copper-superconductor composite. Nb₃Sn was chosen over NbTi, even though the peak field is only 6 T, because of its greater tolerance for temperature excursions due to ac losses resulting from regular pulsed operation of

the PF coils or occasional, unintentional but anticipated, sudden disruption of the plasma current. Preparations are being made to test T-15 coils in clusters of four as they are completed.

Early conceptual designs for T-15 had some of the poloidal windings threading the TF coil bores and operating at liquid helium temperature. Now, however, it appears that all will be water-cooled copper and outside the TF coils.

The schedule for T-15 is uncertain, but manufacture of the TF coils is expected to be completed in the next 3-5 years.

Torus II Supra

In 1977 in France, conceptual design of a compact tokamak with superconducting TF coils focussed serious attention on TF coil problems due to time-varying poloidal fields. The designers of Torus II Supra,⁷ aiming for a peak field of 9 T with NbTi conductor, sought to provide adequate tolerance for ac heating by cooling with pressurized superfluid helium (1.8 K and atmospheric pressure). Tests on heat transfer at Grenoble in 1978-79 showed the practicality of this approach. The Torus II project is now proceeding into detailed design under the Association EURATOM-CEA.

The TF magnet in the Torus II design now has 18 circular coils, in six modules of 3 coils each. Bore diameter will be somewhat larger than 1.8 m. The conductor is monolithic, rectangular in cross section, wound in pancakes on the narrow edge. A design criterion is that the conductor not go normal when subjected to the field transients anticipated in normal operation and as a result of plasma disruption. Because of the relatively small size of Torus II and the high field desired, the current density in the TF windings is necessarily rather high — 5000 A/cm² — but the excellent cooling by the superfluid helium results in cryostability. Joint barriers are incorporated in the conductor to help limit ac losses.

ETF and INICR

Over the past 6-8 years various groups in the US and abroad have considered how the tokamak concept could be embodied in a practical, electricity-producing power plant. Dimensions and details of the preconceptual designs have changed with the state of knowledge of plasma confinement and the various technologies required, but the principal features are similar: a tokamak with D-T plasma that is heated to ignition by neutral beams and microwaves; a cooled toroidal vacuum vessel surrounded by a blanket and shield to absorb particles and energy leaving the plasma; and D-shaped, superconducting TF coils. Depending upon the designers, the PF coils may be inside or out, normal or superconducting. Presently the goal of the US tokamak program, after completing the large experimental confinement machines now under construction, is to build the Engineering Test Facility (ETF).⁸ A national ETF Design Center has been established at Oak Ridge, with participants from all the government laboratories that are active in the tokamak field and several major industrial firms. Preconceptual design began in 1979 and the present schedule is to begin detailed design in 1983 and have the ETF in operation around 1990.

The TF magnet in the ETF reference design, depicted in Fig. 2, presently has 10 coils, each with an 8 × 11-m D-shaped bore. To obtain a field at the plasma centerline of 5.5 T, the peak field at the conductor is 11.4 T. Each coil contains about 15 mA turns of conductor carrying a current of 15 kA. No choice has yet been made between NbTi and Nb₃Sn, nor has the

cooling mode (bath or forced flow) been selected. The induction windings comprise a central solenoid, using NbTi conductor, swung from +7 to -3.5 T in 6 s to induce a current of 5-6 MA in the plasma.

Fig. 2. Reference Design, Engineering Test Facility Fusion Reactor.

A tokamak reactor with dimensions very close to those of ETF is presently the subject of a scoping study by an international team, meeting semiannually in Vienna under the auspices of the International Atomic Energy Agency (IAEA). This concept is called INTOR (for International Tokamak Reactor).⁹ Representatives of EURATOM, Japan, the US, and the USSR have agreed that INTOR should have 12 TF coils, with a 7.5 x 10.5-m D-shaped bore and a peak field of 10.5 T, using either NbTi or Nb₃Sn. The nature of the poloidal windings is presently undefined.

The toroidal magnet for either ETF or INTOR will be huge in comparison with superconducting coils built to date. This is illustrated in Fig. 3, which plots stored energy against peak field. Shown for comparison are the two largest superconducting magnets in the world, the bubble chamber magnets at the National Accelerator Laboratory (NAL) and in Switzerland at CERN. Also shown are two superconducting mirror machines, the Mirror Fusion Test Facility (MFTF) and its modification MFTF-3, that will be discussed in Paper HA-1 by Henning.

IEA Large Coil Task

Organization and Objectives

In 1974 the US Fusion Program decided to build the Tokamak Fusion Test Reactor (TFTR) with water-cooled copper TF coils and concurrently to develop superconducting TF coils for use in subsequent, larger tokamaks. Late in 1975 a decision was reached to proceed with several different large TF coils to be designed and built in US industry for test and evaluation. ORNL was charged with the responsibility for managing this activity, known as the Large Coil Program (LCP).¹⁰ Several coils were needed to achieve the 8-T fields required for testing, and there was inadequate basis for settling on one set of design features (bath cooled or force flow cooled, for example). It was decided, therefore, that specifications should define interfaces, envelope dimensions, and performance requirements but leave internal design to the industrial firms.

Fig. 3. Scaleup of superconducting magnets for fusion: stored energy and field strength.

Consideration of various testing schemes led to selection of a compact, 6-coil toroidal array. It was also decided to award contracts to three different industrial teams for one coil each, with the option of later buying one additional coil from each. In 1977 these contracts were competitively awarded to General Electric, General Dynamics Convair Division, and Westinghouse Electric. Late that year a proposal to build the LCTF at Oak Ridge was prepared by ORNL and accepted by the Energy Research and Development Administration.

At the same time that the US was initiating the LCP, other nations were considering how best to scale up their fusion magnet development. Under the auspices of the International Energy Agency (IEA), member countries and agencies began studying modes of cooperation centered around the LCTF. This led to an agreement under which EURATOM, Japan, and Switzerland are each designing and building one coil to be tested along with the three US coils.¹¹ Design data, certain manufacturing information, and test results are being shared among the participants.

LCTF

The LCTF¹² consists of a test stand, housed in an 11-m-diam vacuum tank, with supporting cryogenic, power, control, and data systems. Test coils are supported from a central column mounted on a 6-legged base, as shown in Fig. 4. A pair of solenoids mounted in the bores of the TF coils can be pulsed to test the capabilities of the different coils. Heat conduction is minimized by supporting the base through rollers on nitrogen-cooled tracks. Radiant heat transfer is blocked by a nitrogen-cooled shroud lining the tank, with superinsulation on either side.

Construction of the LCTF began in 1978. Shakedown and preliminary tests with a partial array (of batch-cooled coils) is scheduled for 1982. Subsequently provisions for testing 6 coils, including those with forced flow, will be completed in time for full testing in 1983.

Test Coil Descriptions

The LCP test coil specifications were prepared by ORNL with the intent of ensuring relevance of the fabrication and testing experience to tokamak magnet requirements. Highlights are listed in Table 1. The size of the test coils was set at 2.5×3.5 -m bore on the basis of the gap in size to be bridged and considerations of technology and cost. Cryogenic stabilization was adopted as a criterion in view of the importance of continuous magnet system operation in a reactor. The specifications place the responsibility on the LCP coil designer to define credible events within his coil and to ensure its stability. As a minimum, US coils are required to have the capability of recovering from any half-turn normal. (The non-US coils are exempted from this latter requirement and others that are not essential to performance as background coils.)

Table 1. Features of test coil specifications

- Cryogenic stabilization
- Either NbTi or Nb₃Sn superconductor
- Produce 8-T peak field in a six-coil torus with five similar coils at 80% of design current
- Operate stably at this peak field, with a superimposed pulsed field at 0.14 T in 1.0 s, with simulated radiation heating
- Survive grossly unequal current distribution among coils of the torus and other specified extended operations
- Conductor current in the range of 10-18 kA
- Stay within specified helium refrigeration limits
- Force transmission, helium supply, and power leads through precisely defined interfaces

A peak field of 8 T was specified to be within the practical range of NbTi at 4.2 K, but use of Nb₃Sn was allowed in order to encourage advances in design and manufacture of coils with the higher-field, but less well-developed, material. The pulse field magnitude is approximately that of the transients to be experienced during normal startup and shutdown of the plasma current in a large tokamak reactor that has some poloidal coils threading the TF coils. The conductor currents are in the range anticipated for reactor coils, so that nearly identical conductors can be used in next-generation coil designs. The limitation on refrigeration power and the interface specifications ensure compatibility with the test stand and supporting systems.

The coil structure is required to withstand operation under extended test conditions, partly to permit a range of testing conditions to allow for exploration of the limits of superconducting operability and partly to accommodate large inequalities of TF coil currents imposed either in tests with a partial array or unintentionally.

Fig. 4. Test stand in LCTF.

Each of the six test coils must meet the specifications to ensure reactor-relevant performance and compatibility with the LCTF. Otherwise, the specifications allow the coil design teams freedom to come up with what they consider to be the best solution to tokamak reactor magnet design and fabrication problems. The LCP coils represent a considerable advance beyond previous experience, and although several design options appear to be feasible, none could be chosen *a priori* as the most promising. As a result, the designs that have evolved for the six test coils are quite different (Table 2).¹³⁻¹⁸

In five of the six coils, the conductor is wound in a heavy stainless steel case. Figure 5 shows a typical coil case. The Westinghouse coil is quite different in that the conductor is mounted in grooves in aluminum plates that are bolted together, as shown in Fig. 6.

High Field Conductors

In consideration of the promising results in development of large coils for 6-9-T peak fields and in view of the incentives for higher fields in tokamaks, fusion programs in several countries are pursuing development of conductors suitable for use in tokamak TF coils at fields up to 10-12 T. In the US up to four different coils (1 m OD) will be tested sequentially in the High Field Test Facility (HFTF)¹⁹ at Livermore at

Table 2. LCP Test Coil Features

	GD/CONVAIR	GENERAL ELECTRIC	WESTINGHOUSE	EURATOM	JAPAN	SWITZERLAND
Core bore (specified)	2.5 × 3.5 m	2.5 × 3.5 m	2.5 × 3.5 m	2.5 × 3.5 m	2.5 × 3.5 m	2.5 × 3.5 m
Peak field (specified)	8.0 T	8.0 T	8.0 T	9.0 T	8.0 T	8.0 T
Ampere-turns	6.65×10^6	6.98×10^6	7.36×10^6	6.62×10^6	6.76×10^6	6.6×10^6
Conductor current	10,200 A	10,450 A	17,800 A	11,000 A	10,210 A	15,000 A
Conductor material	NbTi	NbTi	Nb ₃ Sn	NbTi	NbTi	NbTi
Conductor configuration	Flattened cable in extended-surface copper bar	16 subelements, spiraled around copper core	Cable (insulated strands) in square conduit	22 subelements, spiraled around CrNi core, in rectangular conduit	Flattened cable in rough-surface copper bar	Solder-filled cable surrounded by He-filled copper cable in square conduit
Helium conditions	Pool boiling (4.2K, 1 atm)	Pool boiling (4.2K, 1 atm)	Supercritical, forced flow	Supercritical, forced flow	Pool boiling (4.2K, 1 atm)	Supercritical, forced flow
Winding configuration	Edge wound in layers (14)	Flat wound in pancakes (7)	Laid in spiral grooves in 26 structural plates	Flat wound in pancakes (7)	Edge wound in pancakes (20)	Pancakes (12)
Structural material	304L stainless steel	316 LN stainless steel	2219-T87 plates A286 bolts	Stainless steel similar to 316 LN	304 L stainless steel	Stainless steel similar to 316 LN
Structure configuration	Fully welded case	Welded case with bolted closure	Grooved flat plates, bolted	Welded case, bolted or welded closure	Welded case, bolted side plate closure	Bolted case

Fig. 5. Exploded view of LCP coil by General Dynamics Convair.

Fig. 6. Exploded view of LCP coil by Westinghouse.

fields up to 12 T. Results are to be combined with the information on large coils from the LCP to provide the basis for decisions on ETF coils. Three of the concepts, advanced by the Massachusetts Institute of Technology (MIT), General Dynamics, and General Electric, would use Nb₃Sn. The fourth, being developed by General Atomic, employs a NbTiTa alloy, operating at 3 K. The MIT coil will use forced flow of supercritical helium for cooling; the others, bath cooling.

Tokamak Poloidal Coil Development

For the pulsed induction windings of tokamaks, development is aimed at conductors carrying currents up to 50 kA, with a field swing from -7 to +7 T in a few seconds. In the US this development is centered at Los Alamos Scientific Laboratory (LASL), where a program is under way that will include testing a +7-T, 50-kA, 20-MJ coil²⁰ being built by Westinghouse. Conductor development is also being done at Argonne National Laboratory (ANL). This work is based upon substantial progress both at LASL and ANL in development of energy storage coils that can be very rapidly discharged and recharged.

The other poloidal windings of a tokamak face less demanding requirements than the induction coil, with one notable exception. Because of maintenance considerations, demountable joints are considered to be essential in any poloidal winding that threads the bores of the TF coils. Some windings will almost certainly be required inside to shape and control the plasma. Unless practical joints can be developed for superconducting coils, these will have to be resistive.

EBT

A bumpy torus is a toroidal confinement device consisting of linked mirror cells formed between adjacent toroidal field coils. In the EBT, application of electron cyclotron heating produces annuli of high energy electrons in each cell that modify the magnetic field so as to prevent MHD instability of the plasma. EBT-P is a proof-of-principle experiment to be designed and built at Oak Ridge in the 1980-84 time frame. McDonnell Douglas is the principal contractor, with oversight, physics support, and magnet development by ORNL.

In the EBT-P reference design, the toroidal magnet consists of 36 circular coils, each with a winding bore of 0.48 m. Peak field is 7.4 T. Because of constraints imposed by machine size, field strength, and mirror ratio, a current density in the windings of at least 10,000 A/cm² is necessary. After considering both NbTi and Nb₃Sn and various helium conditions, EBT-P designers chose a metastable design using NbTi conductor cooled by boiling helium at atmospheric pressure, in preference to a cryostable Nb₃Sn design. The reference design²¹ is being verified by construction of two prototype coils at Oak Ridge with 2-kA conductor being produced by Airco.

Conclusions

Development of superconducting TF coils for toroidal fusion machines is proceeding vigorously in many countries. Although this application of superconductivity has some challenging dimensions, its practicality is not in doubt. Several distinctly different design options are being pursued, each promising certain advantages. Within the next three years, fabrication and testing of large coils are expected to provide an adequate basis for confidently choosing a TF coil design for a fusion reactor that will produce useful amounts of electrical energy.

The induction windings of tokamak reactors will also be superconducting. Development for these coils, although on a smaller scale than for TF coils, is making good progress.

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