SUPERCONDUCTING MAGNETS FOR FUSION REACTORS

The magnetic confinement of plasma is the most promising option to use controlled nuclear fusion as a power source for future generations. A number of different magnetic field configurations have been proposed to achieve plasma ignition, all requiring high field strength over a large volume. Most of the experimental machines use conventional, copper windings operated in pulsed mode, to investigate the plasma physics. The advanced plasma experiments, as well as the future fusion reactors, call for long confinement time and high magnetic field, which can be reasonably maintained only by superconducting coils.

Unlike other applications of superconductivity, for fusion magnets there is no "normal conducting" alternative: whenever a magnetic confinement fusion power plant will operate, it will have superconducting windings. For this reason, fusion magnets are an important, long-term factor in the market of superconducting technology. Today, for NbTi-based conductors, fusion is a nonnegligible share of the market, with over 50 t of strand recently used for the LHD and about 40 t committed for W7-X. For Nb_3Sn technology, two large devices, the T-15 tokamak and the ITER model coils, have used most of the conductor ever produced (each about 25 t of strand), being the driving input for the development of high performance Nb_3Sn strands.

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Table 1. Summary of Superconducting Magnet Systems for Fusion Devices

	Strand Weight (t)	Conductor/ Cooling[a]	Stored Energy (MJ)	Peak Field (T)	Operating Current (kA)
Tokamak T-7	1	NbTi/FF	20	5	6
Tokamak T-15	25	Nb_3Sn/FF	795	9.3[c]	5.6[c]
MFTF (all coils)	74	Nb_3Sn+NbTi/pool	1 000	2–12.75	1.5–5.9
TRIAM	2	Nb_3Sn/pool	76	11	6.2
Tore Supra	43	NbTi/pool 1.8 K	600	9	1.4
LHD-Helical (2 coils[b])	10	NbTi/pool 4.5(1.8)K	930 (1 650)	6.9 (9.2)	13 (17.3)
LHD-Poloidal (6 coils)	43	NbTi/FF	1 980	5–6.5	20.8–31.25
Wendelstein 7-X	37	NbTi/FF	600	6	16

[a] FF = forced flow.
[b] Operation at superfluid helium is planned at a later stage.
[c] Design values, achieved on single coil test.

The first use of superconducting coils in experimental fusion devices dates back to the mid-1970s. In the last twenty-five years, six sizable devices for magnetic plasma confinement have been built with superconducting coils (see Table 1): T-7 and T-15 in the former Soviet Union, MFTF in the United States, TRIAM and LHD in Japan, and Tore Supra in France. In Germany, Wendelstein 7-X is under construction. Moreover, a number of developmental and prototype coils have been tested in the scope of large international collaborations (large coil task, demonstration poloidal coils, ITER model coils).

The operating requirement for fusion magnets may vary over a broad range, depending on the kind of confinement and the size of the device (1), for example, from medium-field, pure dc mode in the helical coils of the stellarators, to the high-field, fast rate in the central solenoid of the tokamaks. There is no general recipe for the magnet design, but a few common issues can be identified. Long-term reliability calls for a conservative component design and generous operating margins. The maintenance by remote handling in a nuclear environment imposes strong restrictions to either repair or replacement of individual parts. Safety regulations are also a major issue for superconducting magnets in a fusion reactor: the design must account for any likely or less likely failure mode of the coil system and provide that it will not turn into a nuclear-grade accident. Last but not least, the cost of the magnets, which is a large fraction of the reactor cost, must be contained to be commercially competitive with other power sources.

Only low-temperature superconductors have been considered to date for use in fusion magnets at field amplitudes up to 13 T. A substantially higher field, which would make attractive the use of high-temperature superconductors, is not likely to be proposed as the electromagnetic loads, roughly proportional to the product of field, current and radius, already set a practical limit for structural materials. It may sound surprising that the actual superconducting material cross-section is mostly smaller than 5% of the overall coil cross-section. The choice between NbTi and Nb_3Sn conductors is dictated by the operating field. The upper critical field of NbTi conductors is $\approx$ 10 T at 4.5 K and $\approx$ 13 T at 1.8 K. According to the design current density and the temperature margin, the operating field is set at least 3 T to 4 T below the upper critical field. In the conservatively designed fusion magnets, the peak field for NbTi conductors is up to $\approx$ 9 T for coils cooled by a superfluid helium bath (e.g., Tore Supra and LHD helical coils) and up to $\approx$ 6 T for supercritical helium forced flow (e.g., W7-X and LHD poloidal coils). At a higher operating field, the choice of Nb_3Sn conductors is mandatory to obtain adequate temperature margins and high current density. The increasing confidence in Nb_3Sn technology, as well as its slowly decreasing cost, tends to move down the field threshold for the NbTi versus Nb_3Sn. Conductors based on Nb_3Al are in a developmental stage and may become an alternative to Nb_3Sn for selected high-field magnets (e.g., the D-shaped toroidal field coils), because of the better tolerance to bending strain.

The winding packs may be either potted in epoxy resin or laid out as a spaced matrix of noninsulated conductors in a liquid helium bath. This last option offers the advantages of constant operating temperature and potential high stability due to the bath-cooled conductor surface. The drawbacks are the poor stiffness of the winding and the limited operating voltages (the insulation relies on the helium as dielectric). The potted coils with forced-flow conductors have superior mechanical performance and may operate at higher voltage: as a rule of thumb, they become a mandatory option for stored energy in excess of 1 GJ to 2 GJ. The cable-in-conduit conductors became, in the last decade, the most popular option for forced-flow conductors because of the potential low ac loss and the good heat exchange due to the large wet surface. To withstand the mechanical and electromagnetic loads, the coils are fitted in thick-walled steel cases, either welded or bolted. More structural material may be added, if necessary, both in the conductor cross-section and in winding substructures, for example, plates and cowound strips.

The magnetic stored energy is very large, up to 130 GJ for the proposed magnet system of ITER. In case of a quench (local transition from superconducting to normal state), the stored energy must be dumped into an outer resistor to avoid an overheating and damage of the winding. A large operating current is needed to reduce the number of turns, that is, the winding inductance, and extract quickly the stored energy at a moderately high voltage (up to 10 kV to 20 kV). The operating current density in the superconducting cross-section (NbTi or Nb_3Sn filaments), J_{op}^{sc}, is selected according to the specific design criteria to be a fraction of the critical current density, J_c, at the highest operating field. Typically, J_{op}^{sc} is in the range of 200 A/mm^2 to 700 A/mm^2, and J_{op}^{sc}/J_c = 0.3

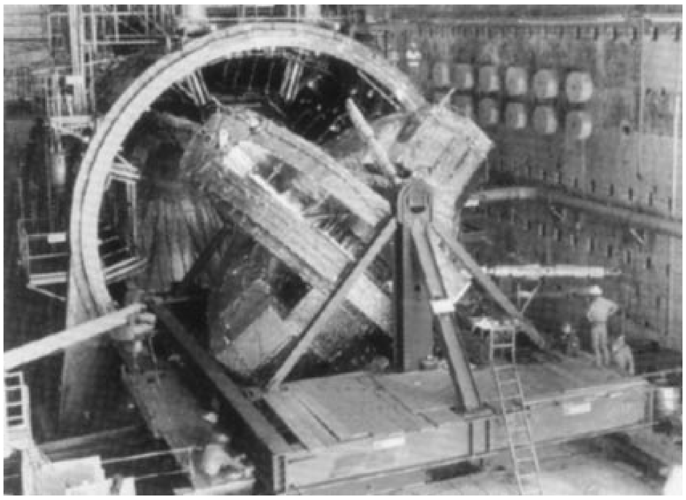

Figure 1. The Yin-Yang coils being assembled at one end the mirror fusion test facility (courtesy of C. H. Henning, Lawrence Livermore National Laboratory).

A/mm^2 to 0.6 A/mm^2. The current density over the coil cross-section is over one order of magnitude smaller.

In the nonsteady-state tokamak machines, the normal operating cycles and the occasional plasma disruption set additional, challenging requirements, in terms of mechanical fatigue of the structural materials and pulsed field loads on the superconductors.

REVIEW OF SUPERCONDUCTING COILS FOR FUSION

Mirror Devices

In the last decade interest in the magnetic confinement in linear machine (mirror fusion) has strongly declined, but they were very popular in the 1970s. At the Kurchatov Institute in Moscow, a plasma trap named LIN-5 was built in 1970 by a split solenoid system with 0.2 m inner diameter and 5.8 T peak field at 1 kA. In 1975, for the LIN-5B machine, a 5 tonne, bath-cooled baseball type coil was wound with 6 km of square, monolithic NbTi conductor, 6.2×6.2 mm^2 (2): at an operating current of 2 kA (75% of the short sample current), the peak field at the conductor is 5.6 T. The basic coil and conductor design is very similar to the US Baseball II-T winding (see below).

In the United States, a superconducting magnetic mirror apparatus (SUMMA) consisting of four coils with 0.9 m outer diameter, 8.8 T peak field, and 18 MJ stored energy was built by NASA in the early 1970s. At Livermore, the first superconducting baseball coil (Baseball II-T) was wound in 1971 with a square monolithic NbTi conductor, 6.35×6.35 mm^2: the peak field in the pool boiling cooled winding is 7.5 T at 2.4 kA, with a stored energy of 12 MJ.

The Mirror Fusion Test Facility (MFTF) (see Fig. 1) was assembled at Lawrence Livermore National Laboratory in 1985 (3). It is the largest set of superconducting magnets for fusion, with a total mass >1200 t and about 75 t of strand. It consists of 8 C-type coils, 12 low-field, and 4 high-field solenoids, including the A2 coils with a Nb_3Sn insert (see Fig. 2). All magnets are pool cooled at 4.5 K by natural convection, with two-phase coolant outlet (<5% gas). The coils are all wound in the coil case, which acts as a cryostat, with ground insulation applied before winding. In the Yin-Yang coils, the winding form is fitted into the thick case by a copper bladder filled with urethane. The turn insulation is provided by G10 spacers, the layer insulation by perforated G-10 plates. The conductors for the solenoids (two types of NbTi and one Nb_3Sn) consist of a thick multifilament composite soldered to the copper stabilizer (see Fig. 3). The joints are made by cold-welding of the composite. For the Yin-Yang coils, the conductor is a square NbTi monolith with a perforated Cu strip wrapped around and soldered to increase the wetted surface. The conductor for the pancake-wound Nb_3Sn insert, A21, is a flat multifilamentary composite soldered in the Cu housing after heat treatment (react and wind): the outer copper surface is oxidized to prevent solder wetting. All the conductors are designed to be cryostable. The fraction of I_{op}/I_c is always smaller than 2/3. After cool-down and successful commissioning of the magnet system, the project was discontinued in 1985. In 1990, the A2 coils were extracted and reassembled in the FENIX conductor test facility, which operated for three years at field levels as high as 13 T.

Tokamaks

The toroidal arrangement of plasma is the most promising confinement geometry, with the largest number of experimental devices, including the pinch and reverse pinch, the stellarator group of machines, and the tokamaks, where a toroidal plasma current is initiated and sustained by a pulsed ohmic heating coil (central solenoid). Most of the superconducting magnets for fusion belong to the tokamak family, including four plasma experiments (T-7, T-15, TRIAM, Tore Supra) and a number of sizable technology demonstration devices (LCT, TESPE, DPC, Polo, ITER Model Coils).

The very first superconducting tokamak, named T-7, was built at the Kurchatov Institute, Moscow in 1975–1976 and first cooled down in 1977 (see Fig. 4). The toroidal winding system (4) consists of 48 circular double pancakes, in aluminum case, with 60 turns in each coil. The average coil diameter is 1 m, the overall cold mass is 12 t and the stored energy 20 MJ at the nominal operation point. The torus was preas-

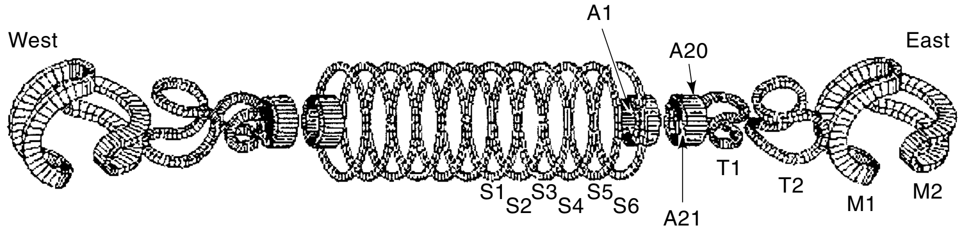

Figure 2. The Axicell coil configuration for the MFTF, with the Yin-Yang coils at the ends.

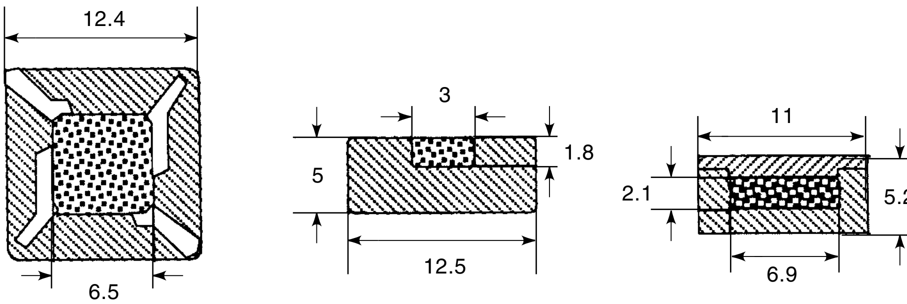

Figure 3. Conductors for the MFTF magnets, from left to right: NbTi square conductor with enhanced wet surface for the M, T, A1, and A20 coils, NbTi conductor for the S solenoids, react and wind Nb_3Sn conductor for the A21 insert. (See Fig. 2 for coil identification.)

sembled into eight segments, individually tested before final assembly. The forced-flow conductor (see Fig. 5), is made from a strip of nine copper pipes, 2 mm inner diameter, with 16 multifilamentary and 32 single-core NbTi strands sitting in the grooves between the pipes and bonded to them by electroplating a 0.6 mm copper layer up to the final size of 28 × 4.5 mm. The cooling is by two-phase helium at 4.5 K, with all the pancakes connected in parallel. A 15-mm-thick copper shell sorrounds each coil and acts as an eddy currents shield for the poloidal field variations. The strands are not transposed and the conductor suffered from severe flux jumps, triggering quenches, during ramp up and ramp down. However, 80% of the design current (6 kA at 5 T peak field) was achieved. Operation of T-7 was discontinued at the Kurchatov Institute in 1987. Later, T-7 was transferred to the Chinese Institute of Plasma Physics, where it has been operating since 1996.

The toroidal field coils of the T-15 tokamak, first operated in 1988 at the Kurchatov Institute in Moscow, are the largest worldwide application of Nb_3Sn conductors (see Fig. 6) (5). The 24 circular coils, with average diameter 2.4 m, consist each of 12 single pancakes, cooled in parallel, with the He inlet at the inner radius joints. The turn insulation is obtained by wet winding to balance the uneven conductor contour. Two stacks of six pancakes are vacuum impregnated in two-halves steel cases, eventually bolted together. The react and wind method was applied. The forced-flow conductor (see

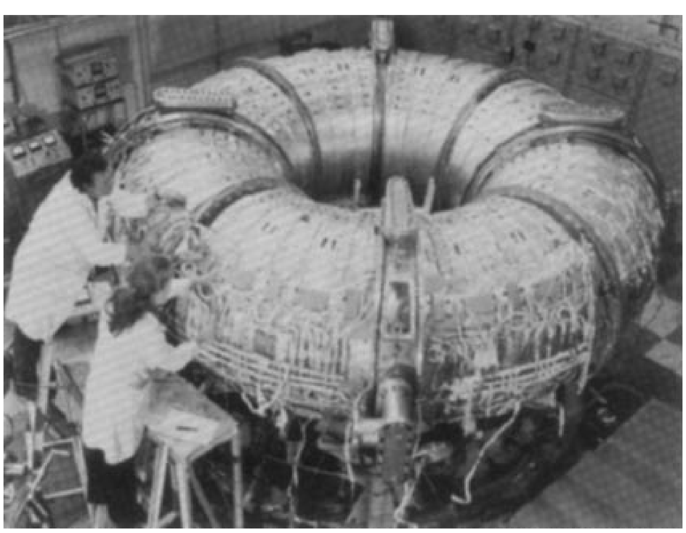

Figure 4. The 24 double pancakes of T-7 in the final assembly (courtesy of V. Keilin, Kurchatov Institute).

Fig. 5), is a flat cable of 11 nonstabilized Nb_3Sn strands, bonded after heat treatment to two copper pipes by an electroplated Cu layer, 1.2 mm thick. Over 100 km of conductor have been manufactured, in units of 200 m. Each coil was tested before assembly and achieved the specification, although a steady-state voltage was observed, by far larger than the joint voltage, in the range of 2.5 mV to 10 mV/coil. The design current in the tokamak is 5.6 kA at 9.3 T peak field. The limited size of the cryoplant (allowing a mass flow rate of only 0.38 g/s · conductor) and the large radiation loss limited the operating temperature to the range of 9 K to 10 K. The highest operation point was 3.9 kA at 6.5 T, 7 K to 8 K, in agreement with the single-coil test and in excess of the original strand specification. The field transients due to plasma disruption, up to 40 T/s, were withstood without quench, with an increase of the outlet temperature by 0.25 K. The design ground voltage is 1.5 kV. For fast discharge, 250 V was applied at the terminals, with a time constant of 104 s.

The TRIAM device at the University of Kyushu, Fukuoka (Japan), is a compact, high-field tokamak, first operated in 1986. The superconducting 16 D-shaped toroidal field coils, with ~3 m average perimeter, are cooled by a pressurized liquid helium bath at 4.5 K. The poloidal field coils, wound with normal conductor, are placed inside the TF coils. The magnet cold mass is 30 t. The toroidal field conductor (see Fig. 7), consists of a large Nb_3Sn bronze composite (10.5 × 3.3 mm for high grade, over a half million filaments) soldered after heat treatment in a copper housing with roughened side surfaces to improve the heat exchange. Beside the copper housing (RRR = 90), a Cu-clad high-purity Al profile (RRR = 3000) is used as a stabilizer. Each coil is a stack of six double pancakes, housed in a steel case. Three conductor grades, with the same width and decreasing height, are used with soldered joints for the double pancake. The conductor is designed to be cryostable and can withstand an energy input up to 7.8 J/cm^3 at the operating conditions: in case of plasma disruption, a normal zone may locally occur but is recovered within 0.3 s. At 6.2 kA, 11 T, the ratio I_{op}/I_c is 0.6. No quench event has been reported after three years of operation, including plasma disruptions (6).

The tokamak Tore Supra has been assembled at Cadarache, France, in 1987 (7). The poloidal coils are wound from copper conductors. The 18 pool cooled circular TF coils, wound from NbTi superconductor operating at superfluid helium, are the largest magnet mass, 160 ton, cooled at 1.8 K. Each coil is made out of 26 double pancake, with an average diameter of 2.6 m (see Fig. 8). A cowound prepreg tape, 0.15 mm thick, is used for the turn insulation. The pancake spacers, 2.2 mm, are built by a perforated prepreg thin plate with glued glass–

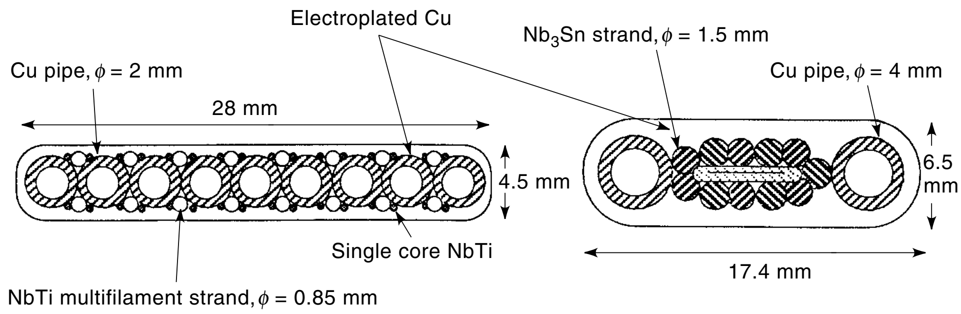

Figure 5. The forced flow conductors for T-7 (left) and T-15 tokamaks, bonded by copper electroplating.

epoxy bottoms. The ground insulation is obtained by overlapped prepreg plates. A 2-mm-thick steel case is shrink-fitted to the winding and contains the atmospheric, 1.8 K He bath. A thick steel case, with thermal insulation, is shrink-fitted and cooled at 4.2 K. The conductor is a rectangular NbTi/Cu/CuNi multifilament composite, 2.8 $\times$ 5.6 mm, wound on the short edge to minimize the ac loss from the poloidal field variation. The temperature margin is $\sim$2.5 K, with T_{cs} = 4.25 K at 1400 A, 9 T peak field. Little copper cross section is used in the conductor: for stability, the He bath enthalpy up to the λ point is available, due to the very high thermal conductivity of He II. The heat exchanger is placed underneath the coil case, open only at the bottom. In case of quench, a He gas pressure builds on the top of the case/cryostat, and siphons the whole He volume within 3 s through the bottom opening, providing a very fast quench propagation and limiting the hot spot temperature below 80 K. In 1988, about six months after first operation, an interpancake short occurred at one coil during a fast discharge, with 1.5 kV across the coil and $\sim$60 V across pancakes, well below the expected Paschen minimum for helium. The damaged coil was later replaced with a spare coil and the dump voltage was decreased to 500 V in order to limit the pancake voltage to $\sim$20 V, which was experimentally assessed as the safe threshold to avoid interpancake discharge. The poloidal field variations and the plasma disruption result in a temperature increase in the He II bath as small as 0.01 K.

Besides the four above described tokamaks, a number of prototype superconducting coils have been built under national or international auspices. The demonstration coils provide valuable opportunities to learn about magnet and conductor technology. In such projects, the pressure for a conservative design is less strong and the performance margins can be better explored than in a plasma experimental device. In the IEA Large Coil Task at the Oak Ridge National Laboratory (8), six large D-shaped magnets, 3 $\times$ 3.5 m bore, have been built to the same common specification using substantially different design approaches (see Table 2). The coils, assembled as a tokamak (see Fig. 9), operated (1984–1985) at the same design point (8 T peak field) with margins ranging from 120% to 140%. For the first time, a cable-in-conduit Nb_3Sn conductor (react and wind coil manufacture) was used in a large-scale application and, despite the broad resistive transition observed in selected coil sections (similar to T-15 behavior), the coil reached 8 T with T_{cs} = 8 K. The cryostable conductors for the bath-cooled coils (GD, GE, JA) could be easily graded (both layer and pancake windings) and, using soldered copper profiles as stabilizers, achieved impressive results in terms of effective use of strand: from 1.4 t of strand in the GE conductor to 8.2 t in the forced-flow conductors of EU (NbTi) and WH (Nb_3Sn), cooled by supercritical helium at 3.8 K, 10 bar to 15 bar. However, two out of three pool cooled coils could not be dumped to the design voltage of 1 kV. The nuclear heat load was simulated by heaters on the inner radius. The poloidal field coil variations were reproduced by a pulsed coil traveling inside the torus: the pulsed field test, with $\Delta B_{\parallel}$ = 0.1 T, $\Delta B_{\perp}$ = 0.14 T, t_o = 1 s, could be completed only for three coils (JA, CH, EU). Over 90% of the stored energy could be dumped in the external resistors, except for the WH coil, with short circuited radial plates.

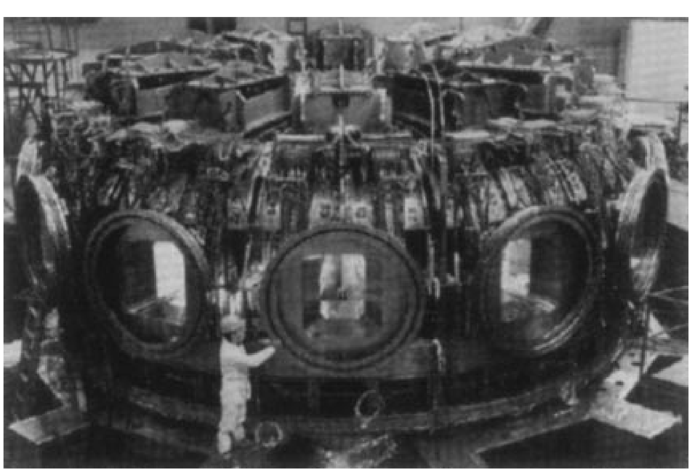

Figure 6. The 24 Nb_3Sn coils of T-15 assembled with the 12 horizontal ports (courtesy of V. Keilin, Kurchatov Institute).

At the same time of the LCT project, two small-size experiments with D-shaped coils were carried out at the Forschungszentrum Karlsruhe (Germany) and at Toshiba (Japan). The six toroidal coils of TESPE at Karlsruhe (9) have a 0.5 $\times$ 0.6 m bore and are pool-cooled at 4.2 K (8 t total cold mass). The coils are wound as double pancakes, shrink-fitted in steel housings insulated by glass epoxy laminate. The steel case is electron-beam welded and serves both as liquid helium container and mechanical reinforcement. The conductor is a soldered flat cable of 24 multifilament NbTi strands ϕ = 1.45 mm, operating at 7 kA, with a peak field of 7 T. The TESPE torus was first operated in 1984 with a test program focused on mechanical load and high-voltage safety issues. The double pancake built by Toshiba (10) in 1983 had a Nb_3Sn cable-in-conduit conductor, 18.3 $\times$ 15.7 mm, with 486 strands, encased into a 1-mm-thick 316L steel jacket. The D-shaped coil, 1.1 $\times$ 0.9 m, was wound after heat teatment and tested at 10 kA,

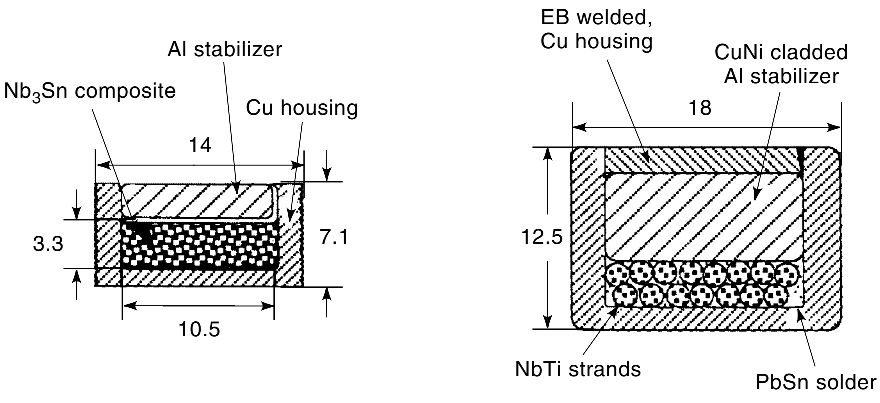

Figure 7. Soldered monolithic conductors, stabilized with high-purity aluminum profiles, for TRIAM (left) and the helical coils of LHD (right).

in a peak background field of 10 T, provided by a small split coil. The test included hydraulic friction factor and stability.

The most recent development project for tokamak's toroidal field coil is the ITER TF Model Coil, to be tested at the Forschungszentrum Karlsruhe by the end of 1999 in the background field of the EU-LCT coil (11). The race-track-shaped coil, with 2.5 × 1.4 bore, has an overall weight of 31 t and a stored energy of 60 MJ at 70 kA (compared to ≈400 t and ≈4 GJ in an individual full size ITER coil). The peak field is about 9 T, compared to 12.5 T in ITER. The main goal of the ITER TF Model Coil is to demonstrate the winding technique, which uses precision-machined steel radial plates where the pancake wound Nb_3Sn cable-in-conduit conductor is encased after the heat treatment (react and transfer method).

The Demonstration Poloidal Coil (DPC) project at JAERI, Naka (Japan), aimed at comparing conductor design options

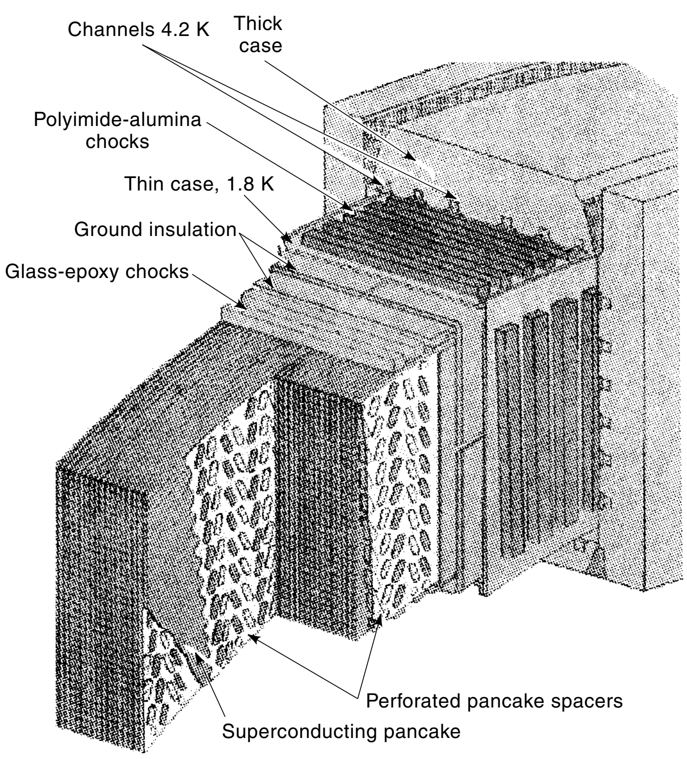

Figure 8. Winding pack layout for the pool-cooled, NbTi toroidal field coils of Tore Supra (courtesy of B. Turck, Tore Supra).

for pulsed-field coils. Three Nb_3Sn react and wind winding models with ϕ_{av} = 1.3 m, DPC-EX (12), US-DPC (13) and DPC-TJ (14) were tested in the DPC facility in pulsed mode up to about 7 T to 8 T, sandwiched between two pulsed NbTi solenoids (DPC-U), connected in series. The conductors are all forced-flow cable-in-conduit (see Fig. 10). The jacket material is Incoloy for the US-DPC (used for the first time). The DPC-TJ had a double jacket: the outer one is 3-D machined without bending and fitted by spot-welding to the conductor after the heat treatment. The strand surface is bare copper for the DPC-TJ, and the coupling loss is 1000 times larger compared to the US-DPC and DPC-EX conductors, with Cr plated strands (2 ms). Ramp rate limitation was observed in the US-DPC, probably due to transposition errors in the cable. The NbTi strands in the cable-in-conduit of the background coils were insulated to avoid interstrand coupling loss: the conductor turned to be unstable due to the inability to redistribute effectively the current among the strands.

The Polo coil, a NbTi circular winding with ϕ = 3 m, has been tested in 1994 at the Forschungszentrum Karlsruhe (Germany) (15). The cable-in-conduit conductor (see Fig. 10), has two separate hydraulic circuits: stagnant, supercritical He at 4 bar in the annular cable region and forced flow, 2 g/s, two-phase He at 4.5 K in the central pipe. This design allows a homogeneous temperature along the conductor with a small pressure drop. The strand is a NbTi/Cu/CuNi composite and the subcables have CuNi or insulating barriers, resulting in very low coupling loss, τ = 210 μs. Four stainless-steel corner profiles are laser-welded around the cable. Phase resolved partial discharge was first used at 4 K to assess the integrity of the glass-epoxy insulation. A midpoint electrical connection in the winding enables to create very-high-field transients in a half coil by a fast discharge of the other half coil. The coil has been tested up to 15 kA, 3.6 T. A degradation of I_c by 30% in dc operation has been observed compared to the strand performance. Polo does not have ramp rate limitation, the stability criterion being the only limiting criterion. Very-fast field transient, up to 1000 T/s are withstood without quench. High-voltage operation, up to 23 kV, has been demonstrated.

The most recent pulsed-coil development for tokamak is the ITER CS Model Coil (16), a layer-wound, two-in-hand solenoid to be operated at 46 kA, 13 T with 0.4 T/s field rate, scheduled to be tested at Naka (Japan) in 1999. The stored energy is 641 MJ, compared with 13 GJ in the full size central solenoid. The conductor is a Nb_3Sn cable-in-conduit with a

Table 2. Summary of the LCT Coil Characteristics

	GD	GE	WH	CH	EU	JA
Winding type	14 layers 3 grades	7 double pancake, 3 grades	4-in-hand, 12 double pancakes	22 pancakes	2-in-hand, 7 double pancakes	20 double pancake, 2 grades
Cooling method	Pool boiling	Pool boiling	Forced flow	Forced flow	Forced flow	Pool boiling
Conductor	Soldered flat cable, on edge	Divided flat cable	Nb_3Sn cable-in-conduit	Square, soldered cable	Divided, flat cable-in-conduit	Soldered flat cable, on edge
Non-Cu J_{op}	586 A/mm^2	525 A/mm^2	265 A/mm^2	302 A/mm^2	393 A/mm^2	327 A/mm^2
Winding J_{op}	27.4 A/mm^2	24.7 A/mm^2	20.1 A/mm^2	30.1 A/mm^2	25.7 A/mm^2	26.6 A/mm^2
He inventory	1320 1	1735 1	440 1	110 1	663 1	1425 1
Total weight	43.9 t	38.6 t	33.7 t	41.7 t	39 t	39 t
SC strand	2 t	1.4 t	8.2 t	3.5 t	8.2 t	2.6 t
Test voltage	2 kV	2.5 kV	9.2 kV	10 kV	12 kV	3 kV

thick-walled Incoloy 908 jacket, manufactured as extruded and drawn-down bars, assembled by butt-welding into a oversize pipe, up to 250 m long: the strand bundle is pulled into the jacket and eventually rolled to the final size. The conductor is insulated by prepreg glass fabric with interleaved kapton foil, applied after the heat treatment by controlled unspringing of the individual layers.

Stellarators

The plasma confinement can be achieved in stellarators by a number of winding configurations. The two large projects using superconducting coils are the Large Helical Device (LHD) at Toki, Japan, and the Wendelstein VII-X (W7-X), to be assembled at Greifswald, Germany. The superconducting magnet system of LHD operates in dc mode and consists of a two helical coils (17) wound around the toroidal vacuum vessel and three sets of circular poloidal coils (18). The helical coils are pool-cooled: initial operation will be at 4.2 K, 6.9 T, 13 kA and, at a later stage, at 1.8 K, 9.2 T, 17.3 kA. A precision tool with 13 numerically controlled driving axes has been used to wind in situ the two helical coils (see Fig. 11). The turn and layer insulation is provided by glass epoxy spacers graded across the winding pack to provide the best mechanical support in the stressed area and the largest wet conductor surface at the peak field (the highest field and the highest mechanical stress are not at the same winding location). The conductor (see Fig. 7), is a NbTi flat cable soldered to a CuNi-cladded Al stabilizer into a copper housing, eventually sealed by two electron beam welds. The conductor is designed to be cryostable, with $I_{op}/I_c = 0.55$. The forced-flow conductor for the poloidal coils (see Fig. 10), is a NbTi cable-in-conduit with 486 strands ($\phi = 0.76$ mm or 0.89 mm), 38% void fraction. To improve the current sharing among strands, the surface is not coated, with a coupling loss constant of 300 ms. The temperature margin is 1.2 K to 1.6 K, $I_{op}/I_c = 0.33$. The joints between the conductor sections are realized by filament joining, resulting in 0.14 nΩ resistance at full current. The OV coil, $\phi = 11.5$ m (see Fig. 12), has been wound on the site, with prepreg turn insulation. The first operation of the LHD was successfully started in March of 1998.

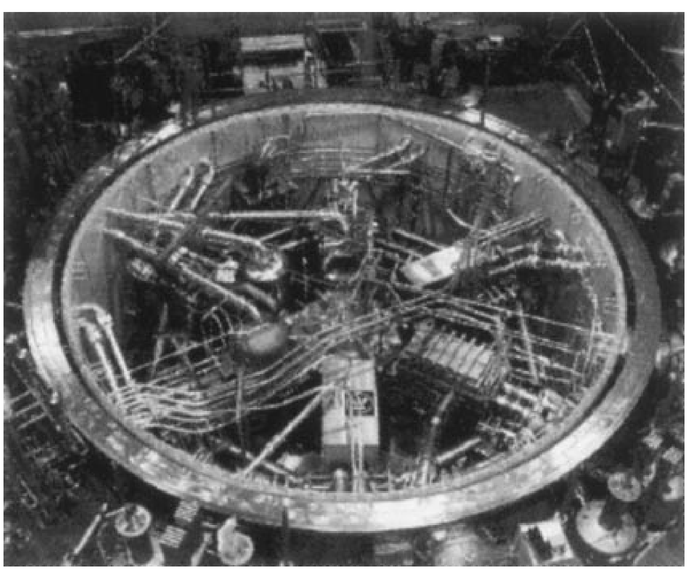

Figure 9. The six D-shaped LCT coils assembled as a torus in the vacuum tank (courtesy of M. Lubell, Oak Ridge National Laboratory).

The magnet system of the W7-X torus consists of 50 nonplanar and 20 planar coils assembled in five modular segments (19). A first nonplanar model coil was completed in 1997 and tested in 1998. The completion of the machine is scheduled by the year 2002. The forced-flow conductor to be used for all the coils (see Fig. 10), is a NbTi cable-in-conduit with 243 strands, $\phi = 0.57$ mm, 37% void fraction. The square jacket is made of a hardenable Al alloy, coextruded around the cable: it is soft after extrusion and during the winding process. After hardening at 170°C, it provides the required stiffness to the winding pack. The temperature margin is >1 K and $I_{op}/I_c = 0.5$.

CRUCIAL ISSUES FOR THE DESIGN OF FUSION MAGNETS

Electrical insulation

The very high level of neutron and gamma radiation at the vacuum vessel of a fusion reactor must be screened to limit the nuclear heat load and the radiation damage at the winding components. The size of the shield (up to 1 m thick in the ITER project) may have a substantial impact on the size and cost of the superconducting magnets. After screening, the radiation damage on the metallic components of a superconducting coil (steel, copper, NbTi and Nb_3Sn), is not critical and partly recovers (e.g., for copper) upon warming up at

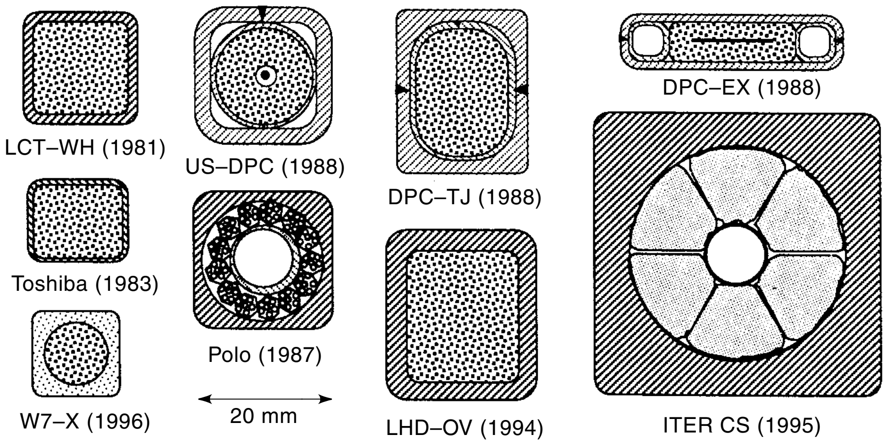

Figure 10. Selection of cable-in-conduit superconductors, drawn to the same scale. The strands of Polo, LHD-OV and W7-X are NbTi, all the other are Nb_3Sn strands. The jacket material is steel except for ITER (Incoloy) and W7-X (aluminum alloy).

room temperature. The actual weak link for radiation damage is the organic fraction of the electrical insulation. In potted windings, the glass-epoxy is broadly used, either as laminates or prepreg wraps or vacuum-impregnated fabrics, to bond together the winding turns and to provide the required dielectric strength. The neutron and gamma act on the long molecular chain of the resin, irreversibly breaking the atomic links. The mechanical strength of the composite, mostly the shear strength, is affected and macroscopic cracking may occur under operating loads, eventually leading to a short circuit between winding sections. To limit this risk, the magnets must be designed to have low stress in the insulation, that is, limit the risk of crack propagation. Another design approach is to separate the mechanical and electrical functions, for example, including a redundant electrical insulation layer, either interleaved or overlapped to the glass-epoxy, to stop the crack propagation in the resin. The free radicals originated from the broken organic polymers are chemically active and evolve into gaseous molecules. The most severe consequence of radiation induced chemical reactions at 4 K is the accumulation of frozen gas bubbles (mostly hydrogen). Upon warming-up of large windings, the internal pressure of the evolved gas increases dramatically due to the little permeation and may lead eventually to swelling in the insulation (20). A possible cure against postirradiation hazards of organic insulation is to reduce the resin volume fraction and select the resin composition to minimize the gas evolution rate. On the other hand, there is a broad reluctance to start an expensive and time-consuming task for the industrial development of innovative insulation systems, which will be actually needed only when a fusion reactor will work at full power on a time scale of several years. The full replacement of organic insulation systems by ceramic materials with adequate mechanical properties may be the ultimate, long-term goal to solve the issue of the electrical insulation in the heavily irradiated fusion magnets.

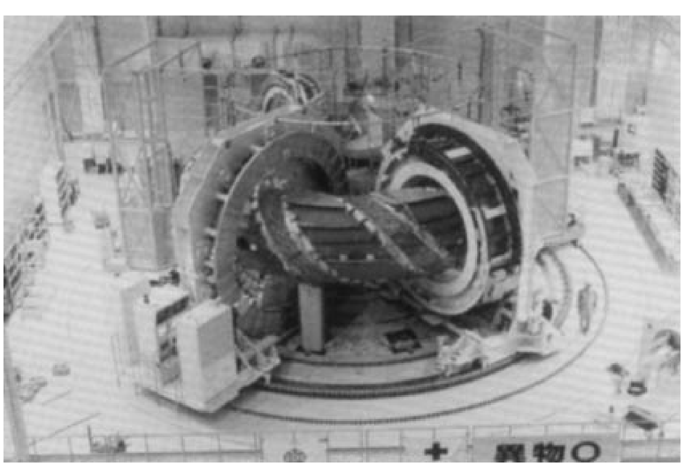

Figure 11. Winding tool with 13 numerically controlled axes for the helical coils of the LHD (courtesy of K, Takahata, NIFS).

Quench Protection

In case of quench, the huge amount of energy stored in a fusion magnet must be actively dumped in an outer resistor. If a quench fails to be detected, the ohmic power locally dissipated in the slowly expanding normal zone is sufficient within one minute or less to melt the conductor and start a chain of serious failures (vacuum break, electric arc, mechanical collapse). A number of quench detectors have been developed and are currently applied in superconducting magnets, from the easy ones (voltage balance of different winding sections, monitoring of outlet mass flow rate) to the most sophisticated, including the laser interference on optical fibers used as distributed thermometer, transmission, and reflection of super high-frequency waves in the coolant channel, acoustic emission, magnetization change at the normal zone (21). However, a redundant and intrusive instrumentation is not welcome in a fusion reactor, as it may increase the risk of leaks and insulation failure, due to the large number of feedthrough required. Whatever the quench detector is, the ultimate question always arises: What happens if the active quench protection fails? The design approach for an actual fusion magnet (i.e., not for an experimental device) will need to offer both a reliable and robust quench-protection system and a conductor/magnet layout that intrinsically limits the damage in case of failure of the protection system, for example, enhancing the quench propagation and the enthalpy at intermediate temperature.

Cost Optimization

In several applications of the superconducting magnets (e.g., accelerators, detectors, high-field magnets, prototypes), the achievement of the technical goal is the main care of the de-

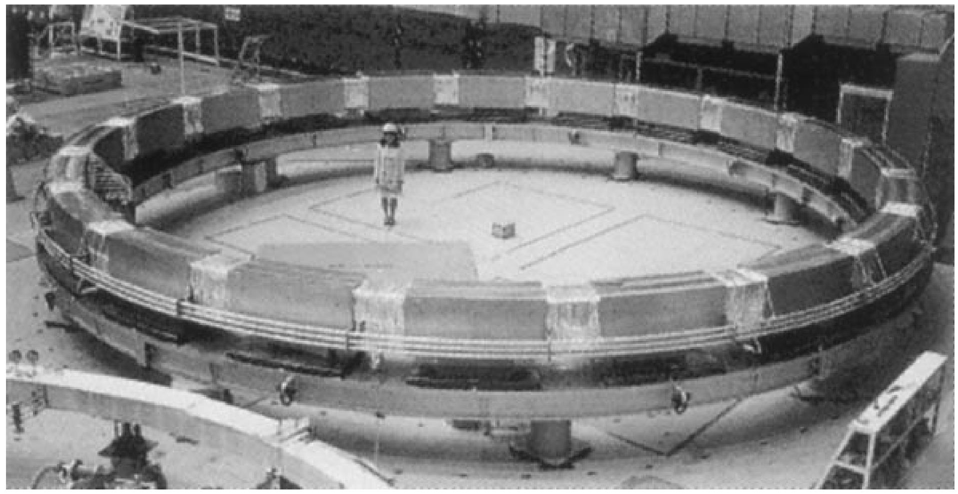

Figure 12. The OV poloidal field coil of the LHD (courtesy of K. Takahata, NIFS).

signer, while the cost of the device does not play a major role. However, after completion of the demonstration phase for the fusion magnets, the cost optimization will be a key issue for the commercial success of fusion. On one side, the behavior of the superconductor needs to be mastered by the designer (e.g., ac loss, stability, mechanical properties), in order to set the design margins at a safe but realistic level and make effective use of the expensive superconductors. On the other hand, the choice of the manufacturing methods and tooling may have a very strong impact on the cost of the coil and should be included as a driving factor in the design. Two examples are given to show how a design choice may affect the cost.

A high electrical conductivity material (stabilizer) needs to be added to the superconductor cross-section, to allow effective current-sharing and fast recovery for small thermal disturbances. The required stabilizer cross-section may be much larger than the superconductor. In cable-in-conduit conductors, the straight choice is to equally distribute the stabilizer cross-section in each superconducting strand, specifying a high Cu : non-Cu ratio. However, the cost of the Nb_3Sn strand is independent of the copper ratio. If the designer masters the mechanism of the current-sharing among strands and knows the operating values of the interstrand resistance, he or she may select a much smaller Cu : non-Cu ratio in the Nb_3Sn strand and add extra copper wires in the strand bundle. Keeping the same superconductor cross-section, that is, without affecting the operating margins, the amount of Nb_3Sn strand can be significantly reduced with a large cost saving.

A Nb_3Sn conductor needs a heat treatment at 650°C to 700°C to form the brittle intermetallic composite by solid-state diffusion. If the designer does not master heat resistant electrical insulation systems, he or she will conservatively choose to first heat-treat the conductor and then insulate it and wind in the final shape (e.g., react and wind or wind and react and transfer methods). As the Nb_3Sn after heat treatment is degraded for permanent deformation as large as 0.2% to 0.3%, the handling for post-heat treatment insulation and final assembly requires sophisticated tooling and continuous adjustment (e.g., shimming of each turn) to achieve the required tolerance with minimum strain on the conductor. If a reliable insulation system is selected, compatible with the heat treatment procedure, the coil can be wound in the final form and to the final tolerance before the heat treatment (wind and react method), saving the cost of a large number of tools and manufacturing steps and avoiding the risk associated to the post-heat-treatment handling.

Risk and Quality Assurance

Large superconducting magnets are usually unique items for which a thorough quality assurance program cannot be conveniently established in advance, as is the case for series production, due to the lack of iterative improvements in the manufacturing procedures. The global acceptance tests of the magnets tend to replace the quality assessment of the individual procedures, achievable only on the basis of a broad statistical database. However, an individual acceptance test of a fusion magnets cannot reproduce all the actual operating conditions, for example, mechanical load and peak field from other coils, nuclear radiation, mechanical and thermal cycling. Moreover, a global acceptance test should not be pushed to the failure limit, that is, the operating margin cannot be assessed. The lack of confidence may push the designer in a circle of overconservative choices, for example, assuming minimum performance for material properties, welds, assembly tolerances, which adversely affect the cost and the effectiveness of the design. To avoid this trend, the designer should identify the critical area for quality assurance and select a low-risk design and procedure. For example, the resistance of a joint between conductor sections cannot be checked during the manufacture. In this case, the designer should aim for a joint layout where the resistance performance is only marginally affected by incorrect assembly procedure.

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