

## Review of European Activities in Superconductivity for Thermonuclear Fusion, in the Light of ITER

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*Abstract* --- Fusion research in the European Union (EU) is organized within the EURATOM treaty as a joint activity since its beginning in the 1960s. In it, superconducting magnet technology has been recognized as an important technological goal already in the 1970s. Since that time a substantial amount of effort on projects with increasing difficulty has been carried out in the different “associated laboratories”. In most cases, project work was done together with European industry and also in international collaborations beyond Europe. Fusion-related activities started with the “Large Coil Task” (LCT) in the frame of a International Energy Agency program, and with the design and construction of toroidal magnet system for the French tokamak TORE SUPRA. A development project for poloidal field coils for very fast field changes followed. In parallel the design of a “Next European Torus” (NET) was started, already with emphasis on the superconducting magnet system and the development of relevant Nb<sub>3</sub>Sn conductors for high current and high field. For their testing, facilities were constructed for investigating at high fields and under mechanical load of up to full size samples - with currents of about 40 kA. With the establishment of the ITER design activity, the NET project and the NET team were terminated. The effort concentrated on ITER and the work was coordinated by EFDA, the newly created “European Fusion Development Agreement”. Major activities concentrated on the development of conductors and the toroidal field coils. A big subtask for EURATOM was the fabrication and testing of an ITER toroidal field model coil (TFMC). For its testing, the coil test facility TOSKA, constructed at Research Centre Karlsruhe (FZK) already for the LCT, was upgraded and used. For conductor testing, beside facilities for strand and sub-size conductor measurements, the full-size conductor test facility SULTAN is playing an important role; it will be complemented with a dipole facility, which is being manufactured. After the decision in June 2006 to proceed with the construction of ITER, Europe decided to create the “Fusion for Energy (F4E) Joint Undertaking” to support the activity of the ITER construction. Beside the work for ITER, the superconductivity program contains also work for the magnet system of the modular stellarator W7-X and for the exploration of the use of HTS in fusion systems after ITER.

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### I. INTRODUCTION

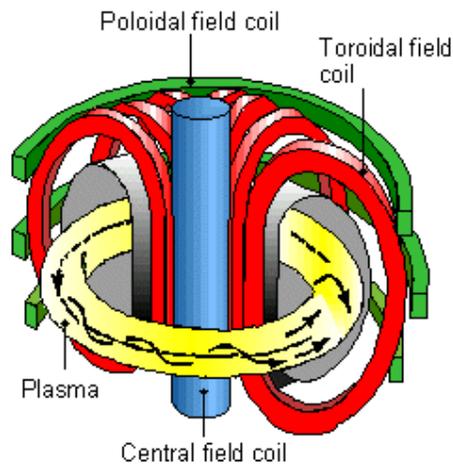
We present an overview the evolution of European activities in superconductivity for nuclear magnetically confined fusion, without offering any extended tutorial introduction to controlled thermonuclear fusion in general, and to fusion magnets and those of ITER (International Thermonuclear Experimental Reactor) in particular. Ample literature exists on these subjects (see, for example, [1-3]) and an overview of ITER magnets was recently published [4].

For the purpose of this article, it suffices to remind the readers that thermonuclear fusion reaction is the source of energy of the sun and the universe. The fusion of nuclei of deuterium and tritium generates energy, which is carried by the alpha particle and the neutron generated by the reaction. The neutron will be slowed down in the blanket surrounding the plasma and it will generate the tritium, which is needed to sustain the reaction. The alpha particle will heat the plasma to maintain the very high temperature needed for the nuclear reaction to occur and continue.

Many approaches have been tried to achieve on Earth and in very confined dimensions what nature produces at large scale. The currently most successful approach is the tokamak. The first tokamak has been built in Soviet Union, about 40 years ago; also the present ITER project is based on this tokamak concept of magnetic plasma confinement. In the tokamak fusion reactor, three main types magnet coils are needed:

- Toroidal field (TF) magnet coils for the confinement and stabilization of plasma, generating nearly static fields, but subject to ac losses in plasma operation.
- Poloidal field (PF) coils for the positioning and shaping of plasma, capable of very fast field changes, and
- A central solenoid (CS) coil for inducing plasma current, also capable of fast field changes.

These three coil types are showed schematically in Figure 1. In practice, additional correction coils (not shown) may also be needed.



**Fig. 1.** Schematic 3-dimensional view of the three main magnet coil types needed to confine and shape the plasma, as well as to induce the plasma current. Red – D-shaped TF coils, green – PF coils, blue CS field coil.

The need for superconducting magnets in magnetically confined fusion for efficient energy generation stems from their ability to produce very intense magnetic fields in very large volumes with only minimal energy consumption. To take the example of ITER, which should demonstrate generation of 500 MW of fusion power with a power amplification factor of 10, its magnetic field system is all superconducting. The coils with the maximum field on the winding of about 6 tesla and below are made of niobium titanium (NbTi) superconducting strands; for higher fields the Nb<sub>3</sub>Sn strands are used. Superconducting cables, used in ITER coils carry very high current, on the order of several tens of kiloamperes. The nuclear and eddy-current-generated heat must be removed from the coils by appropriate cooling scheme permitting to keep their effective temperature in the 4 kelvin range of liquid helium. As will be discussed below, the cable-in-conduit forced-flow concept has been proved to be the most

favourable for cooling the magnet coils of a tokamak. An extensive research and development programme has been carried out to demonstrate the ITER coils feasibility. Critical issues, which have been addressed, are: strain sensitivity of Nb<sub>3</sub>Sn strands, manufacturing of strands with high current and limited hysteretic losses, current distribution in the cable, thermal hydraulic analysis of the cable, cable stability and structural material/support for the cable jacket.

Overall, superconductivity-related activities in fusion magnet development, and specifically for ITER encompass:

- Suitable composite low-loss conductors operating in very strong fields with very high critical current and capable to withstand very high mechanical loads.
- High-field magnets and magnet clusters with attending high-current leads.
- Supporting cryogenics (see Forum review [CR4](#)): all ITER coils are cooled by forced supercritical helium flow maintained by cryogenic circulation pumps.

## **II. HISTORICAL EVOLUTION OF SUPERCONDUCTIVITY-RELATED EUROPEAN ACTIVITIES FOR THERMONUCLEAR FUSION**

In European Union, all fusion-related research and development activities were from their inception in the 1960s conducted jointly within the EURATOM treaty. The superconducting magnet technology for fusion has been recognized as an important technological objective already in the 1970s.

Responding to an invitation from the United States Department of Energy (US-DOE), EURATOM and Switzerland, which in the 1970s was not yet associated with EURATOM, joined in 1976 the so called “Large Coil Task” (LCT) project formulated by US-DOE. The task was to develop a torus for realistic tokamak magnet testing with six superconducting D-shaped coils of about 2.5 m x 3.5 m bore. Three coils had to be developed in USA, one in Japan (which was also invited as a further partner) and one in EURATOM and Switzerland each. The work was formally organized through an “Implementing Agreement” of the International Energy Agency (IEA). All but one LCT magnets (all outside non-US) were constructed using NbTi conductors, which was sufficient to attain the specifications. Nevertheless it was noted that the one LCT coil constructed using Nb<sub>3</sub>Sn conductor readily exceeded the specifications, in spite of the then unproven conductor design approach.

Already during the execution of the LCT task, EURATOM started the definition of a NET (Next European Torus) machine with the development of several key technological components, especially the superconducting magnets, based on the experience achieved with LCT, but for high field capabilities offered by the use of Nb<sub>3</sub>Sn-conductor strands. Also in the late 1970s, CEA (Centre d’Études Atomiques) of France started with the design of the TORE SUPRA tokamak, where superconducting circular toroidal field coils with a mean diameter of 2.6 m, were cooled with superfluid helium to increase the magnetic field intensity while still using NbTi.

As a follow-up of LCT and as part of the NET programme at FZK (Research Centre Karlsruhe, Germany), the development and testing of a poloidal field coil, suitable for fast field changes, was successfully conducted.

With the establishment of the ITER “Engineering Design Activity “ (EDA), Europe decided to create a new organisation called “European Fusion Development Agreement” (EFDA) to organize and to coordinate all activities supporting ITER in Europe. The large subtasks for EURATOM were the fabrication of a toroidal field model coil (TFMC), its testing in the upgraded coil test facility TOSKA at FZK, and the supply of strands and conductors for the CSMC (Central Solenoid Model Coil) facility at Naka, Japan.

After the decision in June 2006 to proceed with the construction of ITER, Europe decided to create the “Fusion for Energy (F4E) Joint Undertaking” to support the activity of the ITER construction. The international partners are participating with funds and contributions in kind to the ITER construction. EU should manufacture 10 TF coils and all PF coils but PF1. This F4E is the EU “domestic agency” of the ITER Organisation (IO). Therefore it shall provide personnel and resources for the design and R&D, as well as it shall procure all the components foreseen as contributions in kind. In order to carry out its duties, F4E can contract work to European research institutions and industries.

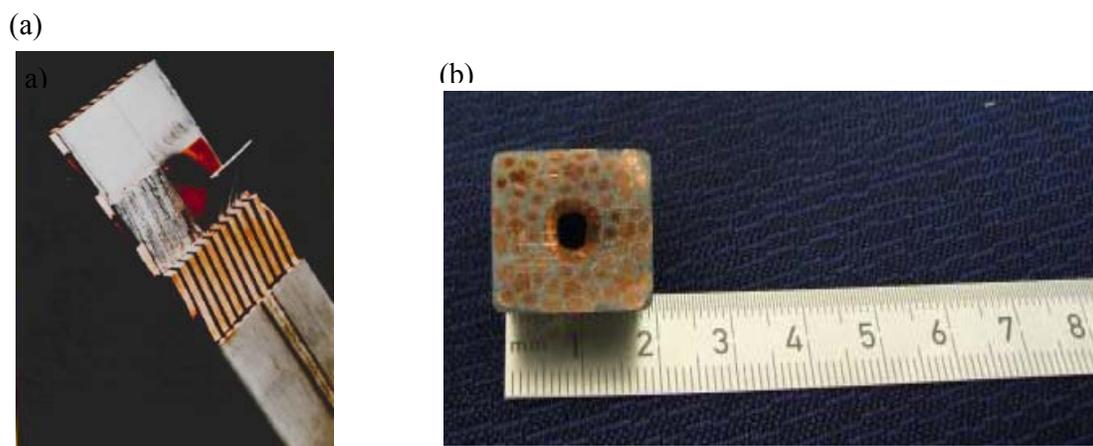
The EURATOM fusion technology programme looks also beyond the ITER time frame. Since several years within the physics programme the construction of the stellarator “Wendelstein 7-X” with large non-planar superconducting coils is in progress and as a long term goal the use of high temperature superconductors in fusion magnets started to be investigated.

In the following sections, we give short technical surveys of the different projects mentioned above.

### III. THE LARGE COIL TASK (LCT) AND ADDITIONAL PROGRAMME

In the 1970s, the LCT project was a big challenge to the partners. Both teams in Europe decided to use advanced forced flow cooling design for the conductors and to provide cable structures which reduce ac losses [5]\*. Figure 2 shows the cross sections of the two conductor types. In comparison to the standard design, which at that time was pool boiling for large coils, the advantages of forced flow within a strong conduit were the optimal force transmission to the coil case, a well-defined heat transfer and fully predictable voltage properties. In 1986/87, the European coils were very successfully operated, together with the other four coils, in the test facility built up for that purpose at ORNL, USA,. In the extended test condition the EURATOM coil reached with 9.1 T at 3.6 K, a record value.

In conjunction with the LCT project a smaller test facility TOSKA (a cryostat with 4.5 m bore and 8 m height was built at FZK initially for testing the EURATOM coil as a single coil before its shipment to the US, but also having in mind future upgrades for using this facility in subsequent fusion coil projects in EUROPE, which turned out to be very wise, as will be seen below. Figure 3 shows the installation of the EURATOM coil in TOSKA.



**Fig. 2.** Cross section of the two European LCT conductors with internal forced flow supercritical He-cooling  
(a) EURATOM conductor, b) Swiss conductor)

\* The US coil with Nb<sub>3</sub>Sn strands also used forced flow cooling.



**Fig. 3.** Installation of the EURATOM LCT coil in TOSKA.



**Fig. 4.** The mechanically reinforced EURATOM LCT coil for its re-installation into TOSKA.

After its operation at ORNL the EURATOM coil was shipped back to Karlsruhe safely. As an additional programme task the investigation of its performance at reduced temperatures (down to 1.8 K) was performed, to demonstrate the possible higher field operation of such NbTi coils [6]. For that purpose the coil had to be reinforced by external thick steel bandages to withstand the huge mechanical forces in a single coil operation. The mechanically reinforced coil is shown in Figure 4. Again, the tests were performed very successfully in the TOSKA facility. While the highest operational current at ORNL was  $I = 16$  kA at 3.6 K, achieving 9.1 T at the winding, the coil was now operated up to 19 kA at 1.8 K, and achieved 11 T. The coil quenched only slightly above that point, close to the critical current load line. This result made it possible to use the coil later on as background field coil for the testing of the ITER TF Model Coil and the Wendelstein 7-X Demo Coil, to be described below.

#### IV. THE TORE SUPRA MAGNET SYSTEM

In the late 1970's CEA started the design of a medium-size tokamak (plasma major radius R: 2.25 m) with the aim to study long plasma pulses in the highest magnetic fields technically feasible at that time [7]. To generate the long plasma pulses, 18 round (circular) toroidal field coils were used. Each coil mean diameter was 2.6 meters. Due to the rather early state of the art of Nb<sub>3</sub>Sn conductors at that time, the designers decided to use NbTi with superfluid He-cooling at 1.8 K to achieve a maximum field of 9 T. To limit ac losses during plasma operation, mainly due to possible plasma disruptions, the rectangular conductor (2.8 x 5.6 mm) was made with a mixed matrix of Cu/CuNi. It included 11000 filaments of 23  $\mu$ m diameter each, for a total rated current of 1400 A. Figure 5 shows a sketch of the coil design. The windings are enclosed in thick stainless steel casings, cooled by circulating supercritical He, while the windings proper are cooled by stagnant He II at atmospheric pressure, a technique developed at CEA during those years.

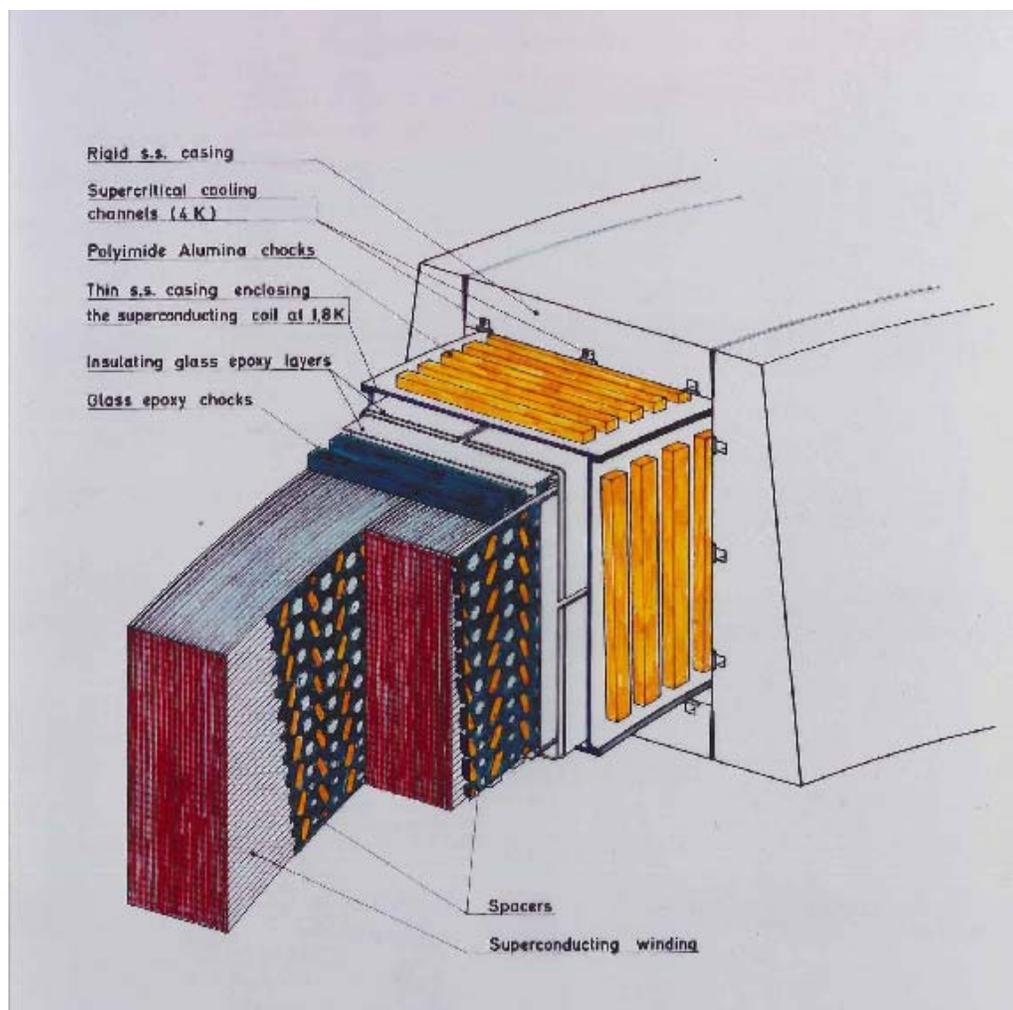


Fig. 5. Sketch of the winding built up of TORE SUPRA (drawing by CEA).

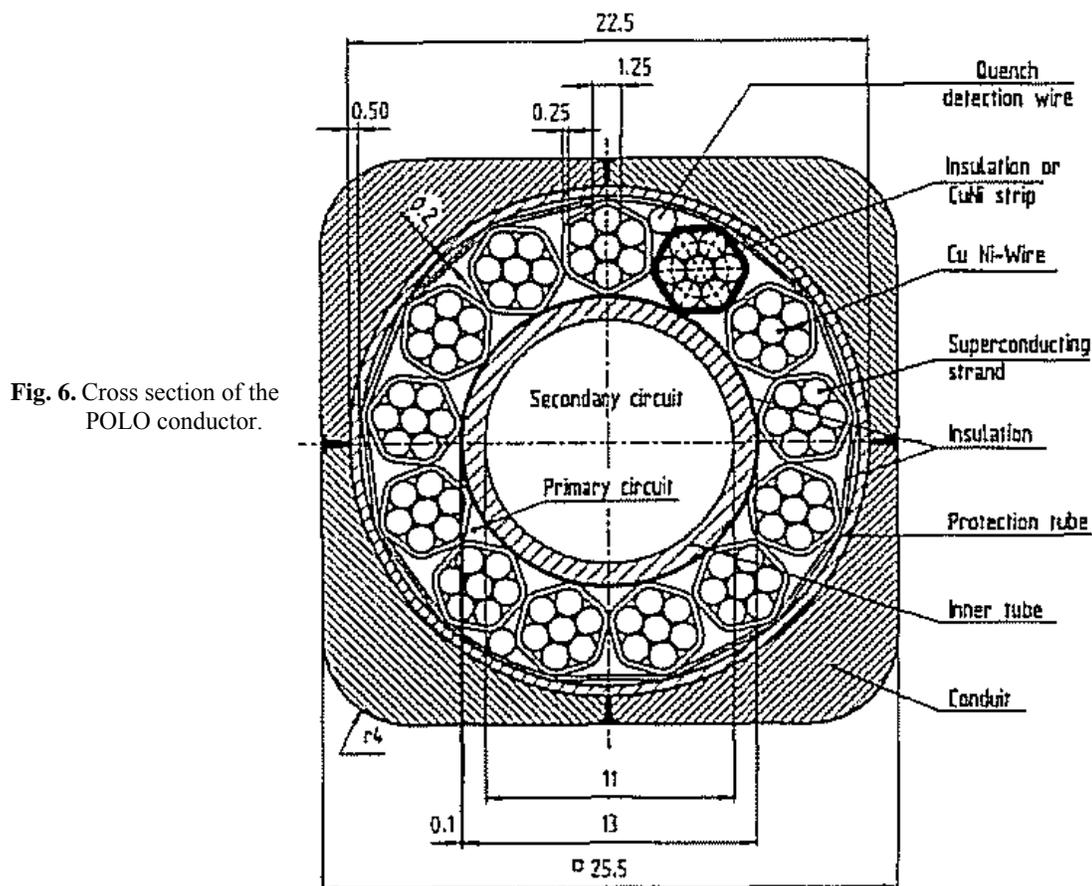
After the successful operation of a prototype coil, magnet construction took place, with acceptance tests following in 1988. Thereby one disadvantage of pool boiling design showed up, namely the limited high voltage properties. A short circuit in one coil occurred, such that this coil had to be replaced. Operation of the tokamak with 90 % of the rated current as the routine value started in 1989. An obviously less important electric short was observed in another coil, not limiting the operation and disappearing after a fast safety discharge in 1995. TORE SUPRA is in operation now for more than 18 years, without further problems and with many plasma discharges up to more than 2 min each. This example demonstrates the reliability of large superconducting magnet systems.

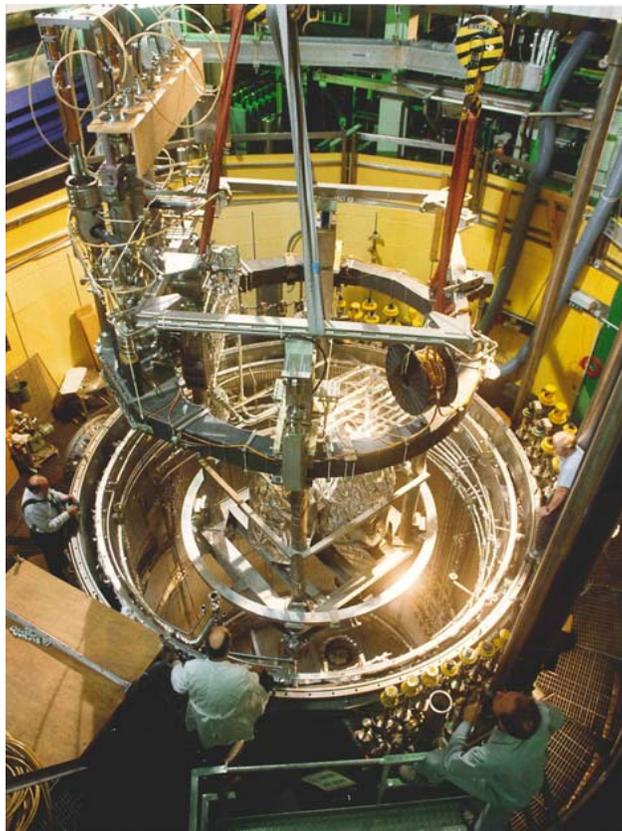
## V. THE POLO PROJECT

In discussion with the TORE SUPRA team, the option for replacing some of the copper poloidal coils by superconducting ones, also in view of the upcoming NET, FZK designed

and build a superconducting poloidal field coil for very fast field changes [8]. Major emphasis in the development has been given to the conductor performance and the high-voltage capability. The coil had an average winding diameter of 3 m and a winding cross section of 216x246 mm. While the field was relatively low (1.6 T), the discharge within less than 10 ms without a quench (!) was a challenge to the conductor and required a rated voltage up to 23 kV. Thus, current leads (rated current 15 kA) and measuring feedthroughs for this high voltage had to be developed. The experience gained in this project had been of great value for later large projects as the ITER model coil.

The superconductor cable consists of mixed matrix strands (NbTi in Cu/CuNi) cooled by a dual cooling system: two-phase forced flow of He for heat removal at constant temperature in central hole (secondary circuit) shown in Figure 6 and stagnant He in direct contact with the superconducting strands (primary circuit) for good transient stability. The cable is enclosed in a thick-walled stainless steel conduit to resist the high mechanical stresses during operation. The fabrication posed quite a challenge to the conductor manufacturer (EAS, former VAC), but turned out successful. The fabricated coil was then tested in the TOSKA facility (Figure 7). A special switching circuit of a high discharge power up to 700 MW (30 kA, 23 kV) was constructed and used for generating the typical positive and negative magnetic field transients. All results were in agreement with the calculations and fulfilled the specifications. One highlight for this conductor type was that the stability against transient field changes measured was in line with the expected stability margin of the critical boundary computed from the strand characteristics.





**Fig. 7.** Installation of the POLO coil into TOSKA.

## VI. THE NET DEVELOPMENT

### A. Establishment of the NET Design and the NET Team

Following the strategy towards fusion energy demonstration in the 70's and 80's, the main objective of the European fusion programme, in the medium term, was to develop the scientific and technical basis necessary to construct NET (the Next European Torus). This was done by carrying out a substantial research programme in tokamak physics at the Joint Europe Torus (JET), inaugurated in 1983, which does not use any superconducting magnets, and in a EU accompanying programme. The (NET) should have followed JET to provide the basis for designing a Demonstration Reactor (DEMO).

In 1983, the NET team was assembled in Garching and started the definition phase for the design of the NET device aimed at demonstrating the feasibility of useful energy production from deuterium and tritium fusion reactions. The definition of the objectives of NET resulted from analysis of different elements such as the technical requirements of DEMO, available knowledge and anticipated rate of progress, resources required, and maintaining of effectiveness in the European Fusion Programme.

The analysis indicated that a device aiming at the demonstration of reactor relevant plasma performance (*i.e.*, long burn), will require large resources, new technologies and improved physics data base which will be obtained by exploiting JET and the other European tokamaks. Therefore, the construction of NET could have been launched in the 1990s when this data base was expected to be available. According to this plan, in 1991 JET operated with a mixture D (90%)-T (10%) and produced about 1MW of fusion power. The general

objectives of NET were defined as the demonstration of the feasibility of fusion energy production in a plant which integrates the essential technologies of a reactor; superconducting magnets represent one such key technology.

In 1986, the NET activity entered the pre-design phase [9] which was overlapping with the ITER Conceptual design Activity (1988-1990). The resources for the NET definition phase amounted to about 200 professional person-year and more than 200 professionals/year were engaged in the associated EURATOM laboratories and industries to carry out R&D.

### *B. The NET Conductor Development*

The conductor R&D before NET was focussed on NbTi conductors and low-current Nb<sub>3</sub>Sn conductors to manufacture, react and wind coils. During the NET definition phase the design team realised very early that the toroidal field coils and the central solenoid should be manufactured using a high-current Nb<sub>3</sub>Sn conductor to attain the high magnetic fields and high conductor current (more than 40 kA) also required to reduce the fast discharge voltage. It was also recognised that for some poloidal field coils the NbTi conductor suffices. The conductor R&D had to be redirected to develop high-current Nb<sub>3</sub>Sn conductors to manufacture coils by “wind and react” technique. The cable-in-conduit conductor was the obvious choice in the light of past experience, and industrial R&D contracts were placed, which lead to the validation of conductors operating at ~ 40kA/ 12.5 T at 4.2 K. A photo of such conductor sections is shown Figure 8.



**Fig 8.** The rectangular NET conductors (CS and TF)



**Fig. 9.** The Sultan Facility

### C. Conductor Test Facilities

Important for the conductor development was, of course, the availability of conductor test facilities. To take into account the strain sensitivity of the Nb<sub>3</sub>Sn conductors, such facilities have to be able to measure the critical current versus field dependence under relevant mechanical load. To be able to do so, several facilities have been built. A few ones for strand measurements in different laboratories, one for sub-cable measurements up to 10 kA/12 T/10 ton (FBI – a facility at FZK) and one for full-size conductor measurements (> 40 kA/12 T) at CRPP in Villigen, Switzerland. This is the SULTAN facility, photo of which is shown in Figure 9. These facilities have been used and also partially upgraded for the ITER conductor development and qualification [10]. Due to the large number of tests needed for ITER construction, EU has decided to build a new dipole facility (EDIPO) to supplement the SULTAN facility. EDIPO is a dipole with a rectangular testing channel of 140 mm x 90 mm, 1200 mm long maintained at 12.5 T [11] and is under construction.

## VII. THE ITER ENGINEERING DESIGN ACTIVITY (EDA)

Following the political decision to go ahead with the ITER Engineering Design Activity (EDA), a worldwide project, European Union decided in 1999 to terminate the NET activity and create a new organisation called “European Fusion Development Agreement (EFDA)” to coordinate all activities in support of ITER EDA. In reality that means that existing teams and expertise were transferred under the new umbrella and the tasks were reformulated accordingly.

The design of the ITER magnet system has been carried out by the central team with the collaboration of the home teams via specific tasks and workshops, where the design progress was reviewed. A description of the design can be found in [8].

Once the design of ITER was well advanced, 2 major magnet R&D projects were launched with the collaboration of the four partners, namely Euratom, Japan, France and USA. Teams of all four partners produced strands and cables for the Central Solenoid Model Coil (CSMC) [13], the EU team made the jacketing of all the conductors. Cross-sections of CS and TF conductors in the jackets are shown in Figure 10. The USA team fabricated the inner CSMC module, while the Japanese build the outer CSMC module and a CS conductor insert. The Russian Federation team made the TF conductor insert, and EU the complete TF Model Coil (TMFC). In year 2000, the CSMC was successfully tested at JAERI 2000 and the TMFC test at FZK followed in 2002 (Figure 10) [14]. In addition to the model coils, special attention was devoted in Europe to the feasibility demonstration of the radial plates and the casing for the TF coils. About 100 tons of forged and cast parts were produced to achieve this objective. The material AISI 316 LN with high nitrogen content was selected for the big forged pieces. Parts of radial plates were produced at full scale to demonstrate the feasibility.



**Fig.10.** The ITER conductors (CS on left, TF on right side)



**Fig. 11.** Installation of the ITER TF Model Coil in TOSKA

## VIII. THE ITER CONSTRUCTION PHASE AND SPECIFICATIONS FOR MAGNET COILS

### *A. Organizational Structure in the Construction Phase*

After the decision in June 2006 to proceed with the construction of ITER, Europe has decided to create the “Fusion for Energy Joint Undertaking” (F4E) to support the activity of ITER. The partners participate with funds and contribution in kind to the ITER construction. The F4E is operational in Barcelona, Spain, and is the EU domestic agency of the ITER Organisation (IO). Therefore, it shall provide personnel and resources for the design and R&D, as well as procure all the components foreseen as contribution in kind. To carry out its duties, F4E may contract work to European Research institutions and industries. The EFDA continues to coordinate the EU physics of the fusion reactor with the related R&D; it is located in Garching near Muenchen, Germany.

The EU shall manufacture 10 toroidal field coils and all poloidal field coils except PF1. This is the only PF coil of radius small enough to be transportable to the ITER site. In the frame of ITER, a bilateral agreement between Japan and EU was signed, which mandates also the construction in EU of the TF coils of the Japanese superconducting tokamak JT60 SA.

### *B. General Specifications for ITER Coils to Be Constructed*

To give the reader a feeling of the magnitude of the undertaking, we present here excerpts of general specifications for the ITER magnet coils. The data of Table I include coils with  $\text{Nb}_3\text{Sn}$  windings, *i.e.*, the TF and CS coils. Table II characterizes the six PF coils, while Table III provides information on the correction coils (CC). A 3D drawing depicting the shapes of correction coils is shown in Figure 12. All PF and CC coils are to be wound with NbTi conductors. The magnets are to be combined in an integrated overall assembly, a schematic section of which is presented in Figure 13. This assembly should simplify the equilibration of electromagnetic loads. When energised, the TF coils will press together along their straight inner sections to form a vault. The coils will be individually encased to aid their support, and to transfer loads across keys between the cases. The poloidal field crossing the TF coils will create overturning moments and circumferential torques on each TF coil. A shell-like structure between the coils, to the extent port penetrations allow, permits these forces to be

reacted within the magnet structure and provides a strong support for the poloidal field coils [15].

**Table I.** General Specifications for ITER Magnet Coils with Nb<sub>3</sub>Sn Conductor

Coil Type	Number of Coils	Dimensions <sup>1</sup> meter	Weight ton	Supercond / Stabilizer	$B_{\max}$ <sup>1</sup> at winding tesla	Design Approach
TF	18	14 H 8 W	290	Nb <sub>3</sub> Sn/Cu	11.8	Welded case w. enclosed grooved plates; each conductor in separate groove
CS	1 6 modules	12 H 4 Ø	840	Nb <sub>3</sub> Sn/Cu	13.5	Independent pancake-wound modules

<sup>1</sup>Dimensions: H – height, W – width, Ø – diameter,  $B_{\max}$  is the maximum field intensity.

In addition to the data of Table I, we should note that the CS coil should produce an inductive swing of 30 webers for plasma burn of 15 megamperes, with a total inductive swing of 280 Wb. The maximum design voltage between terminals is 10 kV and the coil's design capacity is 140 megaampereturns. According to the design, the TF coil should produce field intensity at the plasma axis of 5.3 tesla.

**Table II.** Specifications for PF magnet coils wound using NbTi.

	PF1	PF2	PF3	PF4	PF5	PF6
Winding weight (t)	145	129	385	353	255	263
Max. coil current capacity (MA)	11.34	4.39	8.46	7.74	9.90	19.26
Number of turns per coil:						
Radial	16	11	12	11	14	27
Vertical	16	10	16	16	16	16
Total	252	107	118	172	220	428
Conductor unit length (m) (double pancake, two-in-hand)	392	560	884	807	727	723
Maximum Turn voltage (V)*	625	1750	875	1000	875	714
Ground/Terminal voltage (kV)*	5/10	7/14	7/14	7/14	7/14	5/10

\* Turn voltages are for a two-in hand winding configuration. For PF2 to PF5, a vertical stabilisation voltage of 9 kV (on-load voltage) is assumed, allowing 3 kV above the present 6 kV converter design for possible upgrades.

**Table III.** Specifications for CC coils wound using NbTi.

Parameters	Top coil	Side coil	Bottom coil
Max. operating current (kA)	10.0	10.0	10.0
Max. current capacity per coil (kA)	140	200	180
Max. total field (T)	< 6	< 6	< 6

A three-dimensional sketch illustrating the shapes of correction coils is shown on the next page (Fig. 12).

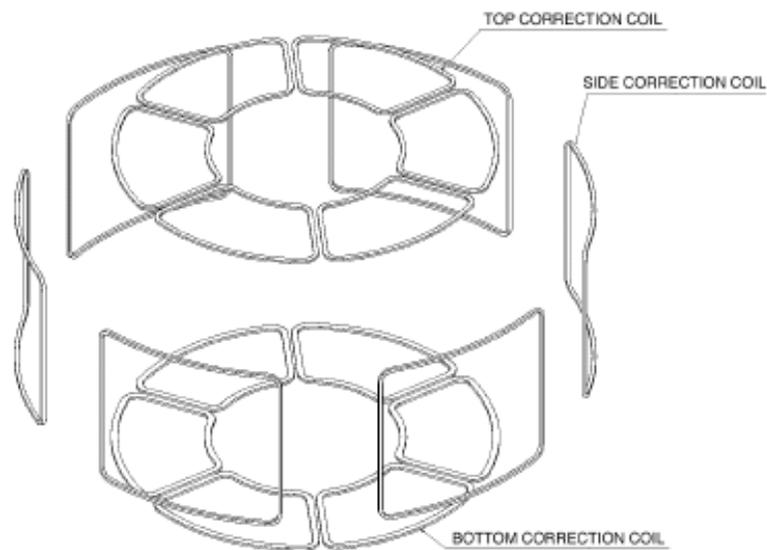


Fig. 12. Three-dimensional sketch of correction coils.

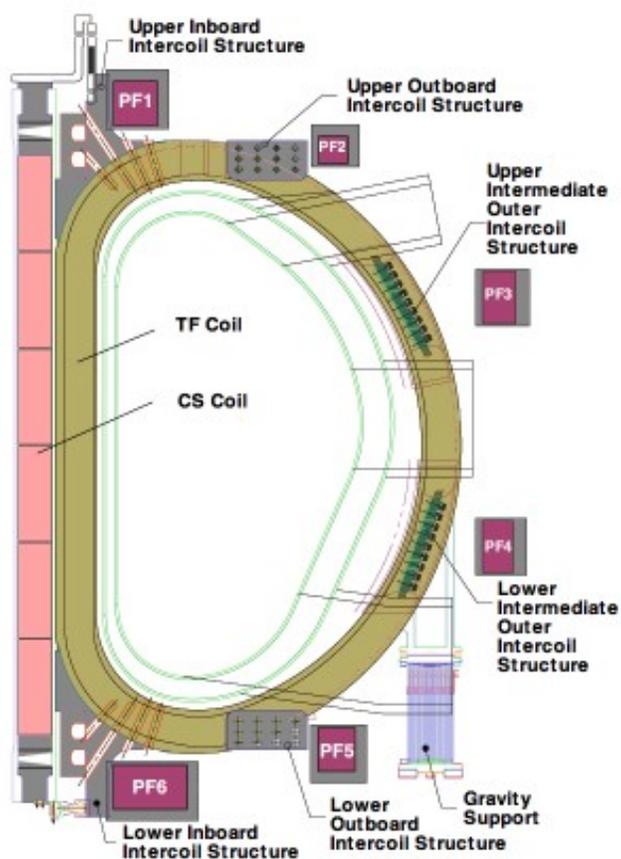


Fig. 13. Schematic section in the ITER coil assembly [15].

The whole magnet assembly requires 60 current leads, for a total current of more than 2.5 MA (see Section X).

## IX. SUPERCONDUCTING MAGNETS FOR THE STELLARATOR W7-X

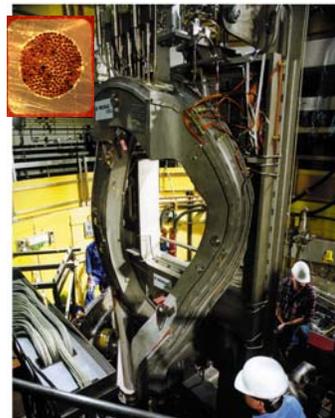
In a purely toroidal magnetic field, the charged particles drift vertically due to the magnetic field radial gradient. To confine these charged particles, toroidal current flow must be induced in the tokamak plasma, which is the function of the CS coil. In an alternative concept, the stellarator, the confinement of charged particles is attained by rotation of the toroidal magnetic field created with electrical current flowing in helical coils, and plasma current is no longer needed.

The stellarator “Wendelstein 7-X” (W7-X) is under construction at the Greifswald branch of the Max Planck Institute for Plasma Physics (IPP) [16]. This modular stellarator experiment of about JET-size is a follow up of the first modular system W7-AS successfully operated at IPP in Garching. The set of non-planar coils provide both the toroidal and poloidal field components to achieve the necessary twist of the magnetic field lines. The size of the machine and the potential of the stellarator concept for steady-state operation justify a superconducting confinement magnet system. In the 1990s, following an extended development effort, the suitable superconductor, the coil manufacture and the complicated cryostat have been developed. The ultimate goal then became the construction and testing of a full size DEMO-coil. This was completed in 1999 by its successful testing in the TOSKA facility at FZK (Fig. 13).

The ultimate W7-X magnet system with the plasma major radius of 5.5 m should produce a field of up to 6.7 T at the winding for attaining the central field at the plasma axis of 3 T. As one can see indicated in Fig. 13, a set of five different types of non-planar coils are periodically arranged in the torus. In addition, 20 planar coils (also superconducting) are placed to allow the variation of the rotational transform between the values  $5/6$  and  $5/4$  (by modifying the main  $B$ -field). Each non-planar coil has a height of about 3.3 m, a width of 2.5 m and a depth of 1.5 m.

The superconductor is of the cable-in-conduit type with NbTi-strands. Due to the need to bend the superconductor over two axes the matrix material is a special developed aluminium alloy (6063) which is soft enough after co-extrusion with the bundle of strands to enable the winding work, but will be hardened afterward during the curing of the winding pack with its resin at about  $100^{\circ}\text{C}$  to achieve sufficient stiffness. The cross-section of the conductor can be seen in the upper left corner inset of Figure 13. The conduit’s dimensions are 16x16mm with 243 strands of 0.58 mm each. The Helium is flowing in the voids between the strands. The rated current is 17.6 kA.

The construction status is well advanced, all coils are manufactured, most of them have already passed the cold acceptance test and their installation is proceeding.



**Fig. 13.** Installation of the W7-X DEMO coil for testing in TOSKA

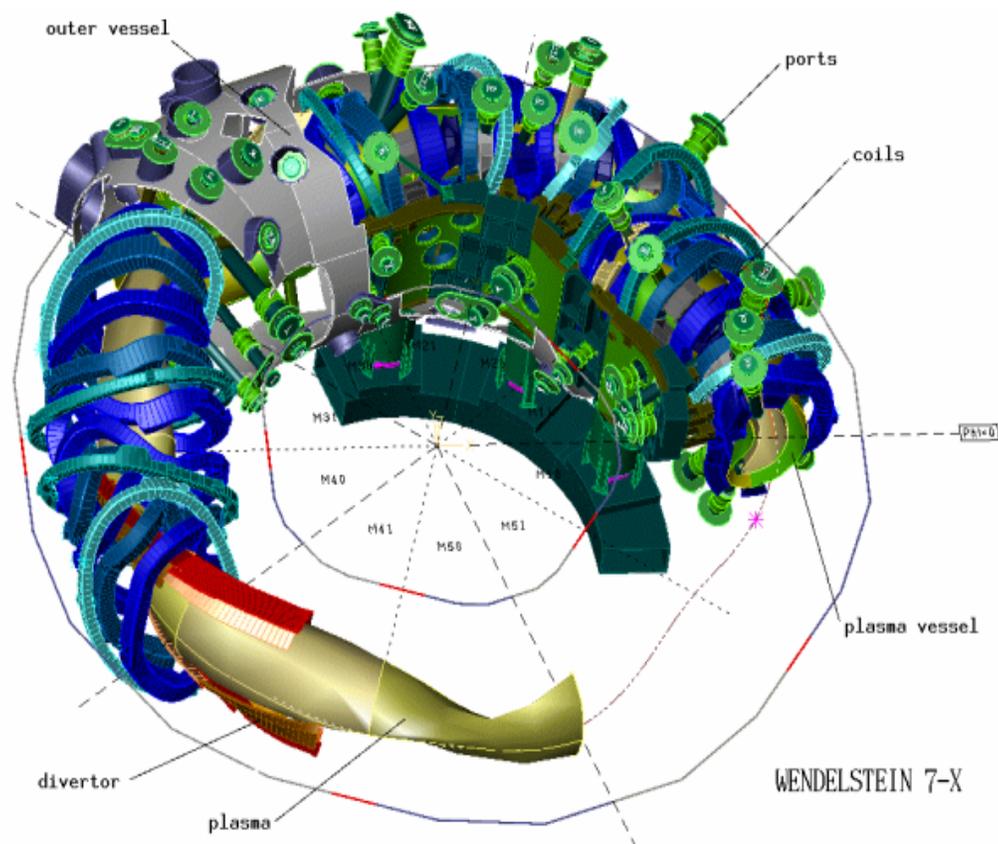


Fig. 14. The layout of W7-X stellarator.

## X. HTS CURRENT LEADS FOR FUSION DEVICES

The 60 current leads (CL) of ITER magnet system should carry a total current of more than 2.5 MA, as already mentioned in Section VIII. To reduce the resulting large refrigeration load at 4.5 K, high-temperature superconductor current leads could be used. Therefore, EFDA had launched a development program for a 70 kA HTS CL demonstrator. This task was executed by FZK together with CRPP in Villingen [17]. The demonstrator leads used Bi (2223) multifilament tape conductors stacked up and embedded in grooves of a non-metallic cylinder.

The constructed HTS CL was installed and tested in the TOSKA facility at FZK. The rated 70 kA current could be carried even at the highest temperature of 85 K. The CL showed a large safety margin in the case of loss of coolant; auxiliary cooling with LN<sub>2</sub> has been also demonstrated. Based on these results, the ITER team decided to use HTS leads thus saving about one refrigerator with cooling power of 20 kW at 4.5 K. In the distribution of working packages among the ITER partners, the leads will be supplied by the Chinese partners, thus further work in Europe on the ITER leads has been stopped. However, the experience gained is used at FZK for the construction of the 16 kA leads for W7-X and the leads for the Japanese tokamak JT 60 Super-Upgrade. Superconducting current leads are also being designed by CRPP for EDIPO. More details can be found in [17] and the Forum paper [ST12](#).

## XI. TOWARDS THE POSSIBLE USE OF HTS MAGNETS IN LATER FUSION REACTORS

With a magnet system operating at 20 K, a fusion reactor would be more efficient with respect to electric power consumption for cryogenics. Even better would be a machine with a magnet system operating above 65 K, which could use liquid nitrogen as a coolant with much reduced volume of thermal insulation. High magnetic fields required for fusion, would limit the use of the already commercially available BSCCO tape conductors of the first generation (1G) to about 20 K. However, 2<sup>nd</sup> generation (2G) YBCO coated conductors (CC) on metallic tape substrates have made impressive progress recently, and could make possible the operation above 65 K, even in high fields. A big drawback thus far is that such conductors are only available in the form of thin tapes with high aspect ratio. This makes the tailoring of high-current conductors and attaining low ac-losses very difficult. Therefore, possible cabling and bundling techniques to fabricate Roebel bar type conductors capable of alleviating these problems are in progress worldwide, as documented by Forum papers [ST15](#) and [ST26](#). At FZK, both the transfer of cabling method to industry and the development of medium-size model coils with Roebel conductors are planned, and could be anticipated for the next approximately five years.

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