Lecture 4:
ITER's Superconducting Magnets

(The largest set of superconducting magnets ever built)

AP 4990y Seminar
Columbia University
Spring, 2011

References

- Huguet. "Integrated design of the ITER magnets and their auxiliary systems." Nucl Fusion (1999) vol. **39** (11Y) pp. 2033-2041.
- Okuno et al. "Key features of the ITER-FEAT magnet system." Fusion Eng Des (2001) vol. **58-9** pp. 153-157
- Huguet. "Key engineering features of the ITER-FEAT magnet system and implications for the R&D programme." Nuclear fusion (2001) vol. **41** (10) pp. 1503
- Huguet. "The ITER magnets: preparation for full size construction based on the results of the model coil programme." Nucl Fusion (2003) vol. **43** (5) pp. 352-357
- Coatanea et al. "Quench Detection in the ITER Magnet System." Applied Superconductivity, IEEE Transactions on (2010) vol. **20** (3) pp. 427-430

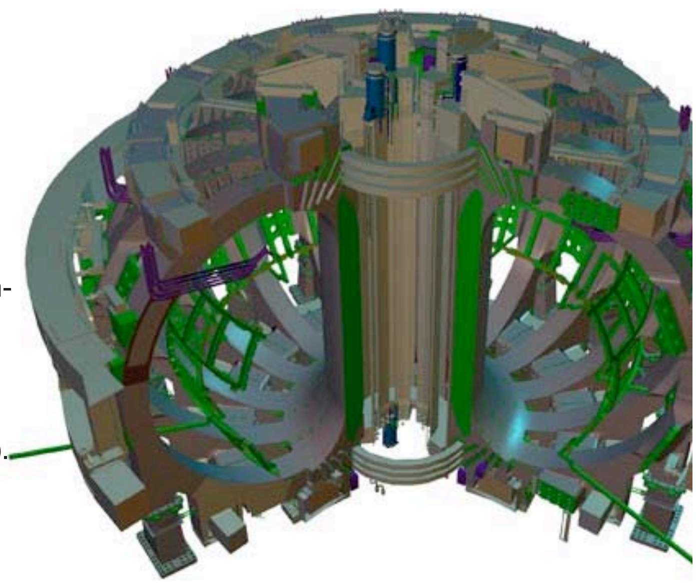
Description of ITER Magnets

(http://www.iter.org/mach/magnets)

•The superconducting material for both the Central Solenoid and the Toroidal Field coils is designed to achieve operation at high magnetic field (13 Tesla), and is a special alloy made of Niobium and Tin (Nb3Sn).

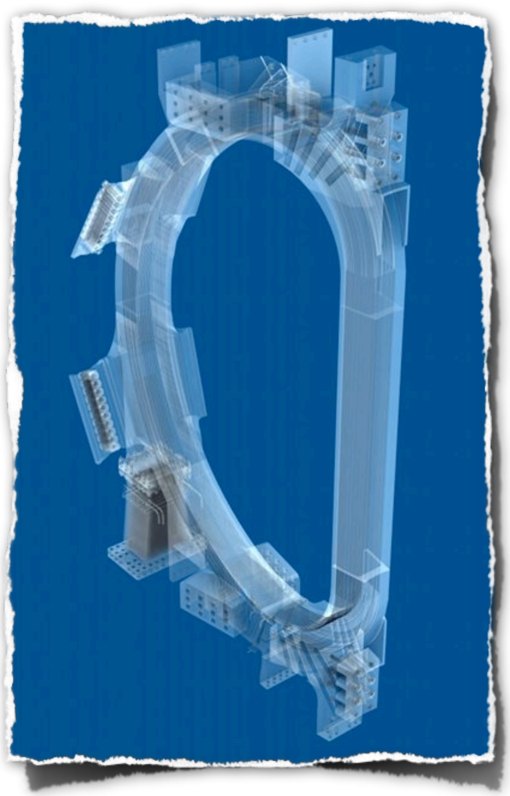
•The Poloidal Field coils and the Correction coils use a different, Niobium-Titanium (NbTi) alloy.

•In order to achieve superconductivity, all coils are cooled with supercritical Helium in the range of 4 Kelvin (-269°C).



TF

- The 18 Toroidal Field (TF) magnets produce a magnetic field around the torus, whose primary function is to confine the plasma particles.
- The ITER TF coils are designed to have a total magnetic energy of 41 gigajoules and a maximum magnetic field of 11.8 tesla.
- The coils will weigh 6540 tons total; besides the Vacuum Vessel, they are the biggest components of the ITER machine.
- The coils will be made of Cable-In-Conduit superconductors, in which a bundle of superconducting strands is cabled together and cooled by flowing Helium, and contained in a structural jacket.
- The strands necessary for the ITER TF coils have a total length of 150.000 kilometres and would span the earth more than three times

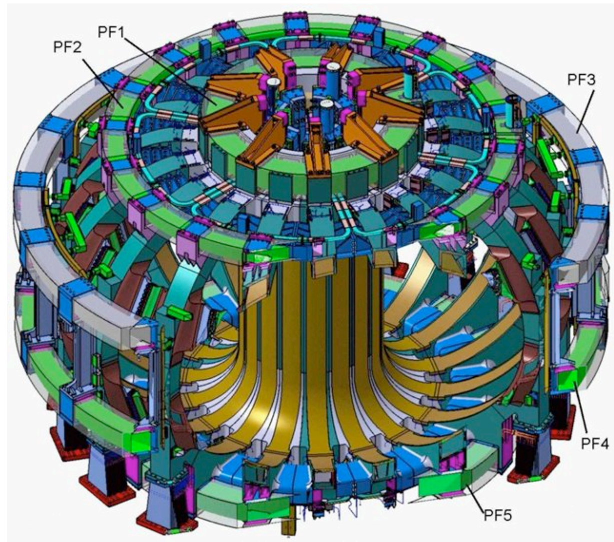


PF

- The Poloidal Field coil system consists of six horizontal coils placed outside the Toroidal Magnet structure.

- Due to their size, the actual winding of five of the six PF coils will take place in a dedicated, 250-metre long coil winding building on the ITER site in Cadarache. (The smallest of the PF coils will be manufactured offsite and delivered finished.)

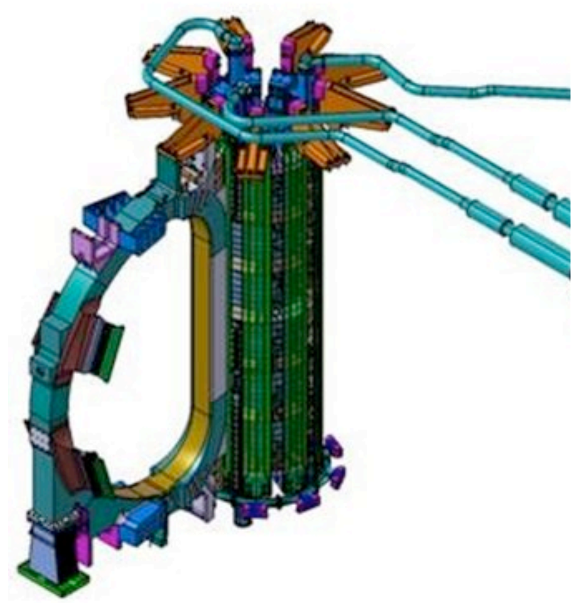
- The ITER PF coils are also made of Cable-in-Conduit conductors. Two different types of strands are used according to operating requirements, each displaying differences in high-current and high-temperature behavior.



CS

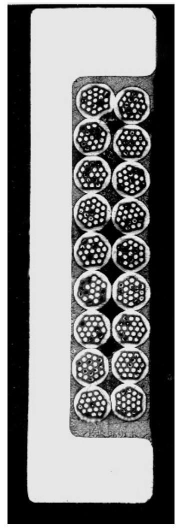
- The Central Solenoid is made of six independent coil packs that use a Niobium-Tin (Nb_3Sn) Cable-in-Conduit superconducting conductor, held together by a vertical precompression structure. This design enables ITER to access a wide operating window of plasma parameters, enabling the testing of different operating scenarios up to 17 MA and covering inductive and non-inductive operation.

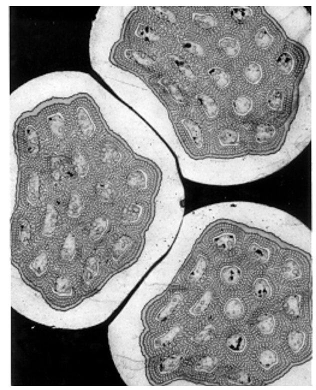
- Each coil is based on a stack of multiple pancake winding units that minimizes joints. A glass-polyimide electrical insulation, impregnated with epoxy resin, gives a high voltage operating capability, tested up to 29 kV. The conductor jacket material has to resist the large electromagnetic forces arising during operation and be able to demonstrate good fatigue behavior. The conductor will be produced in unit lengths up to 910 meters.



Nb_3Sn (Niobium Tin)

- Discovered in 1954
- Ceramic (brittle)
- In 1961, niobium-tin exhibits superconductivity at large currents and strong magnetic fields, becoming the first known material to support the high currents and fields necessary for high-field magnets
- Tc = 18.3 °K
- In April 2008 a record non-copper current density was achieved at 0.26 MA/cm^2 at 12 T and 4.2 K





LDX Conductor in Soldered Cable

NbTi (Niobium Titanium)

- First superconducting wire
- Relatively common place; large quantities produced
- Tc = 9.2 °K
- Used in Fermi Lab, the Large Hadron Collider (LHC), and the Large Helical Device (LHD)
- In LHC, 1,200 tons of NbTi are cooled to 1.9 °K for operation at 8.3 T with 10 GJ of stored energy (1/4 of ITER)

LDX C-coil

Fusion Superconducting Magnets

- T-7, T-15 (Russia)
- TRIAM (Japan)
- Tore Supra (France)
- IEA Large Coil Test
- LHD (Japan)
- LDX (US)
- W-7x (Germany)

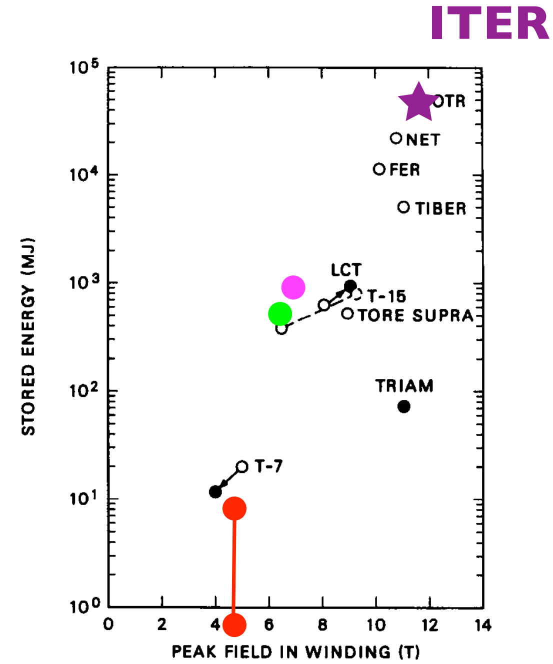


Fig. 10.1. Size and peak magnetic field in toroidal fusion magnets.

Large Coil Test (1977-87)

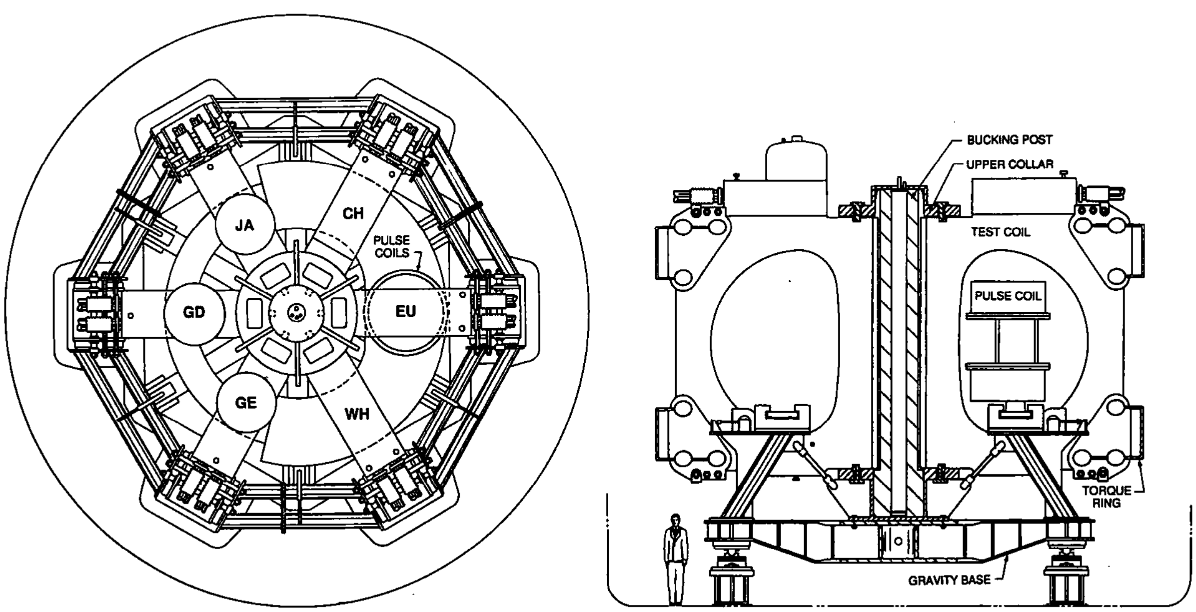
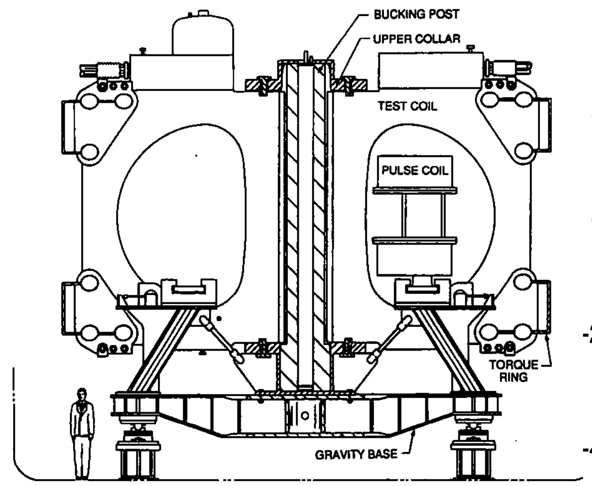


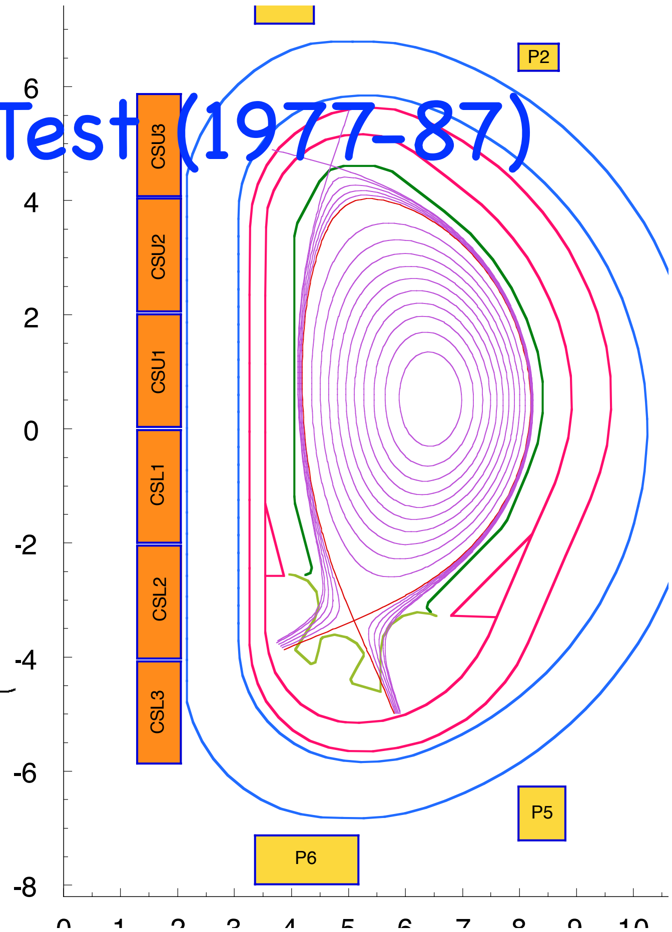
Fig. 2.1. Arrangement of coils in the IFSMTF test stand.

Large Coil Test (1977-87)



s in the IFSMTF test stand.

As compared with ITER…



"Results were gratifying"

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North-Holland, Amsterdam

3

EXECUTIVE SUMMARY

A multinational program of cooperative research, development, demonstrations, and exchanges of information on superconducting magnets for fusion was initiated in 1977 under an IEA agreement. The first major step in the development of TF magnets was called the Large Coil Task. Participants in LCT were the U.S. DOE, EURATOM, JAERI, and the Département Féderal de l'Intériéur of Switzerland.

The goals of LCT were to obtain experimental data, to demonstrate reliable operation of large superconducting coils, and to prove design principles and fabrication techniques being considered for the toroidal magnets of thermonuclear reactors. These goals were to be accomplished through coordinated but largely independent design, development, and construction of six test coils, followed by collaborative testing in a compact toroidal test array at fields of 8 T and higher.

Under the terms of the IEA Agreement, the United States built and operated the test facility at Oak Ridge and provided three test coils. The other participants provided one coil each. Information on design and manufacturing and all test data were shared by all. The LCT team of each participant included a government laboratory and industrial partners or contractors, as shown in fig. 1.

The last coil was completed in 1985, and the test assembly was completed in October of that year (see fig. 2). Over the next 23 months, the six-coil array was cooled down and extensive testing was performed. Results were gratifying, as tests achieved design-point performance and well beyond. (Each coil reached a peak field of 9 T.) Experiments elucidated coil behavior, delineated limits of operability, and demonstrated coil safety.

This special issue of *Fusion Engineering and Design* makes available to all potential users the LCT results and experience, which are described in detail sufficient for useful guidance of further work on toroidal superconducting magnets.

NbTi & Nb_3Sn (PB & FF)

Table 1
Distinctive features of the LCT test coils

Participant	United States	United States	United States	EURATOM	Japan	Switzerland
Direction	ORNL	ORNL	ORNL	KfK	JAERI	SIN
Coil	GD/C	GE/OR	WH	Siemens	Hitachi	BBC
Conductor	IGC/GD	IGC	Airco	Vacuumschmelze	Hitachi Cable Ltd.	SASM/BBC
Superconductor	NbTi	NbTi	Nb_3Sn	NbTi	NbTi	NbTi
Cooling mode[a]	PB	PB	FF	FF	PB	FF
Design current (A)	10,200	10,500	17,760	11,400	10,220	13,000
Current density, winding (MA/m^2)	27.4	24.7	20.1[c]	24.1	26.6	30.7
Current density, coil (MA/m^2)[b]	15.3	15.7	17.6	16.3	16.0	17.9
Design MA turns	6.40	6.53	7.53	6.70	6.73	5.95
Winding configuration	Edge wound, 14 layers	Flat wound, 6 double pancakes	In grooves in 24 plates	Flat wound, 7 double pancakes, impregnated	Edge wound, 20 double pancakes	Wound in 11 double pancakes, impregnated
Structure	Type 304L SS, welded case	Type 316LN SS, bolted and welded case	Al alloy 2219-T87, bolted plates	Type 316LN SS, bolted, sealed case	Type 304LN SS, bolted and welded case	Type 316L/316LN SS, bolted case

[a]PB: boiling at 0.1 MPa (4.2 K); FF: 1.2 MPa (3.8 K).
[b]Average over total cross section of winding and structure in nose region.
[c]Includes plate structure inside the outermost conductor boundary.

Conductor Configurations

Left: PB
Right: FF

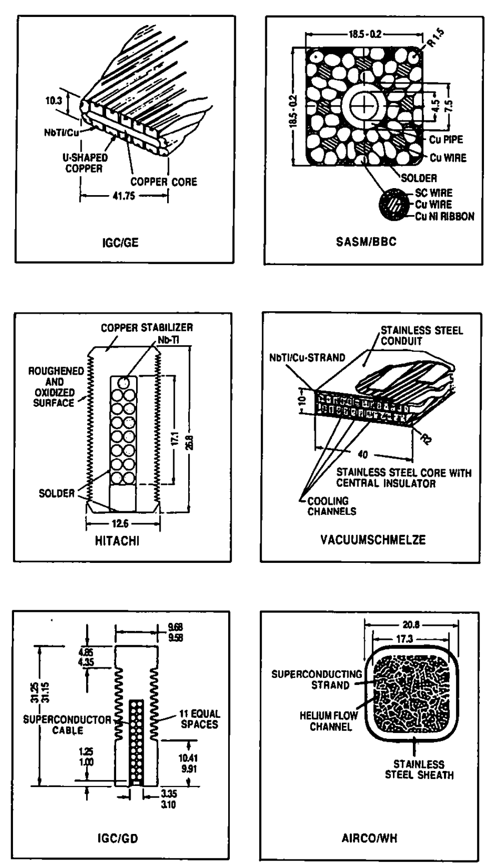

Fig. 3. Configurations of conductors in the LCT coils.

LCT Schedule

Table 2
Chronology of operation with full test array

Locate and repair leaks and evacuate tank	Nov.–Dec. 1985
Start cooldown and replace leaking heat exchanger	Dec.–Jan. 1986
Cool down and begin superconducting	Jan.–Feb. 1986
Single-coil checks and tests to full current	Mar.–June 1986
Test controls in six-coil operation	June–July 1986
Six-coil tests at rated current and 8 T	July–Nov. 1986
Tests with pulsed poloidal field	Dec.–Mar. 1987
Single-coil high-temperature and high-current tests	Mar.–June 1987
Six-coil higher-field tests	July–Aug. 1987
Five-coil tests with extreme out-of-plane loads	Aug. 1987
Coil safety tests	Aug. 1987
Highest-symmetric-field tests	Sept. 1987
Warmup	Sept. 1987

Table 3
Facility problems that delayed coil testing between November 1985 and September 1987

Problem area	Delays (d)
Air leakage into helium	124
Abnormal heat leaks	14
Leak in cooldown heat exchanger	29
Leak in oil cooler	17
Helium compressors	31
Other helium system components	3
Test coil current systems	5
Pulsed-field system	7
Data acquisition system	8
Planned services (air, water, and electricity)	22
	260

would be furthered by exchanging information on practical aspects of manufacturing conductors and coils. The Annex therefore required participants to arrange visits to coil fabrication facilities. This provision was

LCT Conclusions

The number of non-U.S. personnel stationed at the test facility averaged six at any time, with length of assignments typically ranging from one month to two years. Each participant was represented on a local committee that met at least weekly to review progress and decide near-term plans. There was complete integration of all participants in the coil testing and analysis activities but not in the facility operation, which was performed by specially trained crews of ORNL personnel.

At the conclusion of the test program, coils belonging to Japan, EURATOM, and Switzerland were returned to the country of origin. The final act of the collaboration in LCT was the evaluation of results, agreement on conclusions most relevant to future magnets for fusion programs, and joint preparation of a summary report.

Conclusions

- All 11 critical objectives identified at the outset of the LCT were accomplished.
- The goals of coil and facility designs were achieved.
- Limits of coil operability were explored and found

peratures much higher than would be tolerable for NbTi conductors.

- The practicality of both PB and FF cooling for TF coils of this size was demonstrated.
- Heat-removal capabilities commensurate with nuclear heating in the TF magnet of a tokamak reactor are practicable with either PB-cooled or FF-cooled coils.
- LCT tests demonstrated outstanding stability against thermal disturbances in the PB-cooled coils of this size.
- LCT tests demonstrated that thermal disturbances in FF-cooled coils can be minimized by the construction techniques possible with internally cooled conductors.
- Satisfactory stability of much larger tokamak magnets should be achievable through use of the design procedures tested in LCT.
- High-voltage insulation of conductors and the consequent feasibility of rapid discharge of FF-cooled coils in the event of a quench were demonstrated in LCT.
- Loss of flow in an FF-cooled coil need not demand extremely rapid action to prevent damage.
- Energy that is magnetically stored in TF coils of a tokamak can be harmlessly dissipated by practical means.
- Evaluation of LCT results, in conjunction with other magnet R&D findings, suggest that FF cooling is preferable for TF coils much larger than those in LCT. Further R&D is needed, however, on the stability of FF-cooled coils with much longer cooling channels.
- Solutions to the problem of detecting a normal zone in the LCT coils were demonstrated, but further development, including alternate methods, is desirable.
- There were enough similarities between operation of the LCT test stand and the operation of magnet systems in a tokamak to make the LCT experience

ITER Magnet Topics

- Huge forces
- Manufacturing
- Large energy & quench protection
- Cryogenics (mechanical support without thermal conduction) and cooling
- CS & PF fatigue
- Alignment, assembly, symmetry, thermal contraction, ...

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ITER SC Coil Types

TF: Nb_3Sn, 18 coils (11.8 T max)
CS: Nb_3Sn, 7 coils (13.5 T max)
PF: NbTi, 6 coils (6 T max)

- The whole magnet system is supported by flexible columns and pedestals, one under each TF coil.
- Each TF coil is electrically insulated from its support. The TF coil case also supports the vacuum vessel weight and operational loads.
- All TF coils, the CS and the upper and outer PF coils are designed for removal from the machine in the event of a major fault.
- The cryostat is designed so that the lower (trapped) PF coils can be rewound in situ under the machine. In addition, the PF coils have accessible joints (located at the outer diameter), so that individual double pancakes can be disconnected in-situ in the event of a fault.

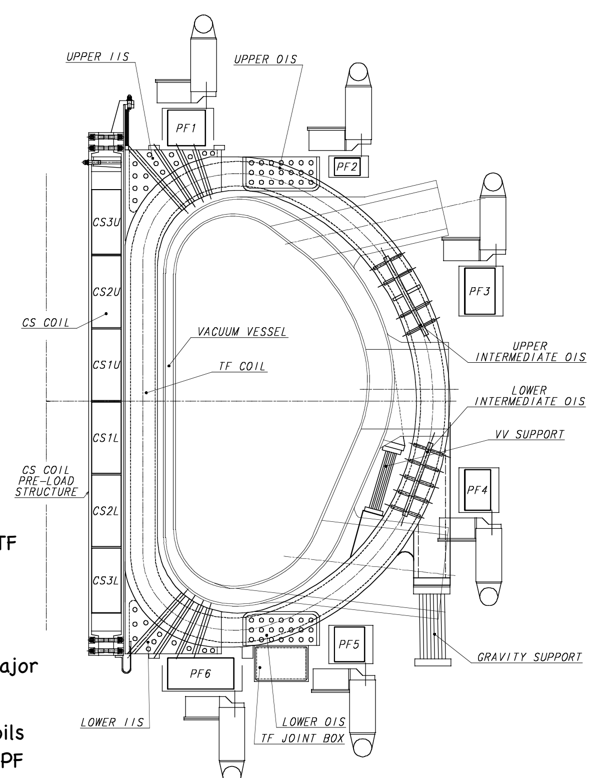

Figure II.1.3-1 Magnet Elevation View

All forces (including disruption forces) are supported from TF coil casing

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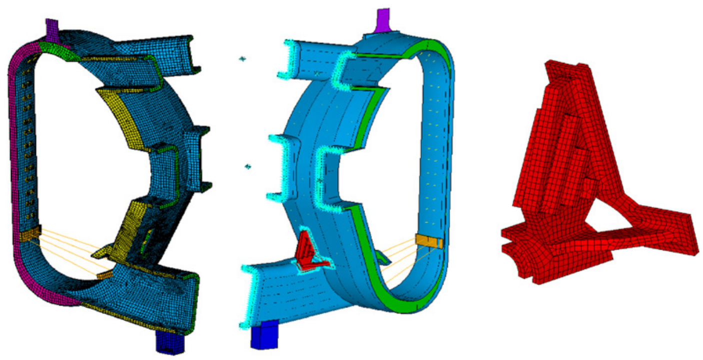

Figure 21. Reinforcement of the lower port connection to the main ITER vacuum vessel shell is shown in red. Only one of the two gussets is shown.

Key features of the ITER-FEAT magnet system

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Abstract

The design of the ITER magnet system is being finalized. The reference design of the winding pack of the TF coil is based on the use of circular conductors supported by radial plates. This design has been chosen for its high insulation reliability during operation. The overall TF coil structure includes pre-compression rings made of unidirectional fiber glass, which reduce the stress level in the outer intercoil structures and the coil case. The design of the central solenoid, including pre-load structure, has been developed. Two conductor jacket options are still under investigation for the CS and the final choice will be based on the results of on-going R&D.

K. Okuno et al. / Fusion Engineering and Design 58–59 (2001) 153–157

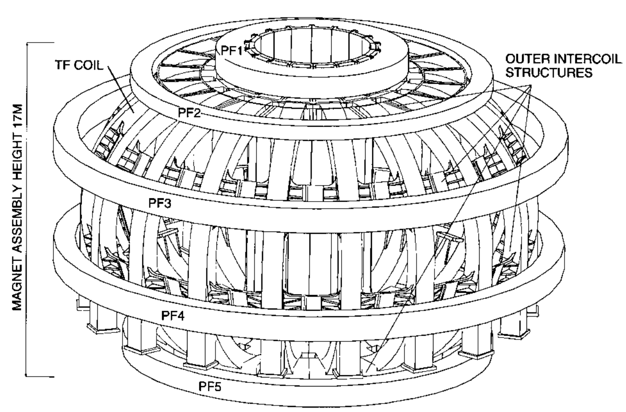

Fig. 1. ITER magnet system—isometric view. CS and correction coils are not shown.

Key engineering features of the ITER-FEAT magnet system and implications for the R&D programme

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Abstract. The magnet design of the new ITER-FEAT machine comprises 18 toroidal field (TF) coils, a central solenoid (CS), 6 poloidal field coils and correction coils. A key driver of this new design is the requirement to generate and control plasmas with a relatively high elongation ($\kappa_{95} = 1.7$) and a relatively high triangularity ($\delta_{95} = 0.35$). This has led to a design where the CS is vertically segmented and self-standing and the TF coils are wedged along their inboard legs. Another important design driver is the requirement to achieve a high operational reliability of the magnets, and this has resulted in several unconventional designs, and in particular the use of conductors supported in radial plates for the winding pack of the TF coils. A key mechanical issue is the cyclic loading of the TF coil cases due to the out-of-plane loads which result from the interaction of the TF coil current and the poloidal field. These loads are resisted by a combination of shear keys and 'pre-compression' rings able to provide a centripetal preload at assembly. The fatigue life of the CS conductor jacket is another issue, as it determines the CS performance in terms of the flux generation. Two jacket materials and designs are under study. Since 1993, the ITER magnet R&D programme has been focused on the manufacture and testing of a CS and a TF model coil. During its testing, the CS model coil has successfully achieved all its performance targets in DC and AC operations. The manufacture of the TF model coil is complete. The manufacture of segments of the full scale TF coil case is another important and successful part of this programme and is near completion. New R&D effort is now being initiated to cover specific aspects of the ITER-FEAT design.

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Critical issues for the manufacture of the ITER TF coil winding pack

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ABSTRACT

Research and trials by the Japan Atomic Energy Agency (JAEA) focus on the remaining technical issues in the ITER TF coil winding pack (WP) manufacturing process. Specific issues include the feasibility of automatically measuring conductor length during automatic winding with a high degree of accuracy (±0.02%) and a fabrication process to comply with the demanding tolerances (up to 1 mm distortion in flatness and 1.5 mm in-plane shrinkage) of the radial plate (RP) due to cover plate (CP) welding. The authors developed a new technique to measure conductor length very accurately by combining an ordinary encoder and a newly developed optical system. A simulation based on test results of CP welding using a RP mock-up indicates that a flatness of 1 mm is achievable, but the in-plane shrinkage of the RP is approximately 5 mm. One possible solution is to fabricate the RP larger than required to allow for in-plane shrinkage. Another solution is to reduce the thickness or length of the welding. The feasibility of these solutions to most of the major technical issues suggests that it is time for full qualification testing of the fabrication process in a dummy double-pancake trial.

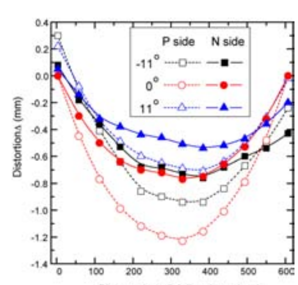

Fig. 6. Out-of-plane distortion of the RP mock-up after welding on the P and N sides. The out-of-plane distortion is measured at the ends (−11° and 11°) and in the center (0°) of the RP mock-up with the welded surface facing up. The P and N sides in the figure indicate the out-of-plane distortion measured after welding the P and N sides, respectively.

carried out to predict in-plane shrinkage precisely and to demonstrate the feasibility of achieving these highly demanding tolerance requirements.

4. Conclusion

A number of technical issues in the TFWP manufacturing process remain to be solved. Research and trials by JAEA focused on solving these issues, with the following results:

1) A new, highly accurate system for measuring conductor length to within ±0.02% is developed by combining an ordinary encoder and a newly developed optical system.
2) In-plane shrinkage of the DP due to CP welding is predicted to be about 5 mm, which is larger than the tolerance requirements. One possible solution is to fabricate the RP to dimensions larger than required in order to allow for in-plane shrinkage by welding. Another solution is to reduce welding thickness and/or length.

From these results, it may be concluded that most of the major technical issues have been solved and that it is time to move to final qualification testing via a dummy DP trial fabrication to completely demonstrate the feasibility of the TFWP manufacturing procedure.

Acknowledgment

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Cryostat and Thermal Shields/Supports

In all cases the thermal shields consist of stainless steel panels that are cooled by helium gas with 80K inlet temperature. The cooling lines remove the heat load intercepted from the warm surfaces. The cold structures, operating around 4K face the TS surfaces. The conductive heat loads from all thermal shields are limited to small losses through their supports.

- Initially, circular metallic bellows were considered to connect the interspace duct wall extensions of the VV ports with the cryostat port. Bellows are required to compensate for differential movements. However, due to the relatively large port sizes, these bellows would become so large that there would be insufficient space left between them for accessing (for repair operations) the region between the equatorial and divertor ports inside the cryostat. Two alternative designs have been proposed involving either metallic, circular bellows that are attached outside the interspace, or rectangular bellows made of reinforced elastomer materials. The latter leave maximum space for interventions inside the cryostat near the equatorial and divertor port regions and have the least impact on the building and component layout. **The use of rectangular, elastomer bellows is therefore the present reference configuration.**

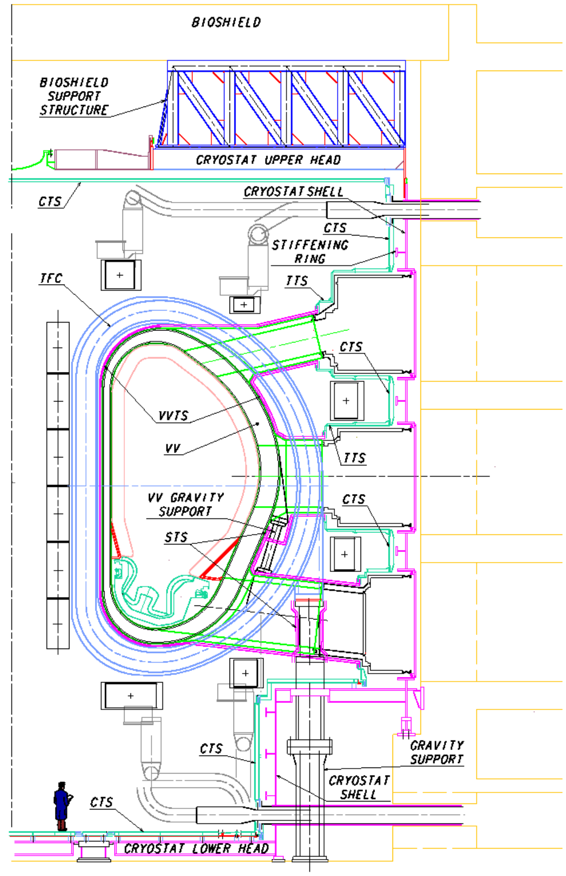

Figure II.3-1 Elevation View of Cryostat, Thermal Shields (cryostat (CTS), transition (TTS), vacuum vessel (VVTS) and support (STS)), and Gravity Supports

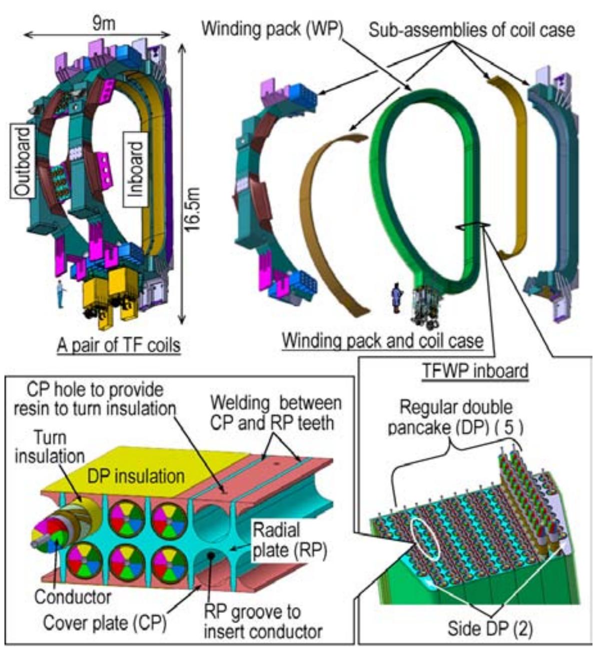

Fig. 1. ITER TF coil main design features.

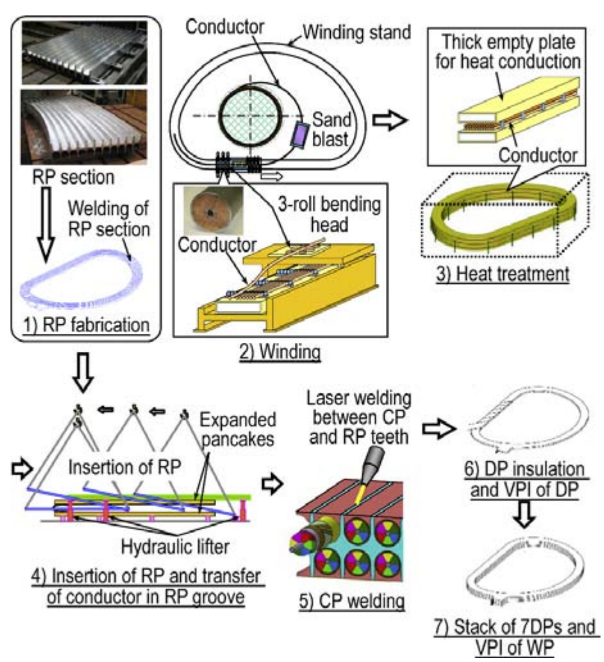

Fig. 2. Manufacturing process of the TF winding pack.

PF

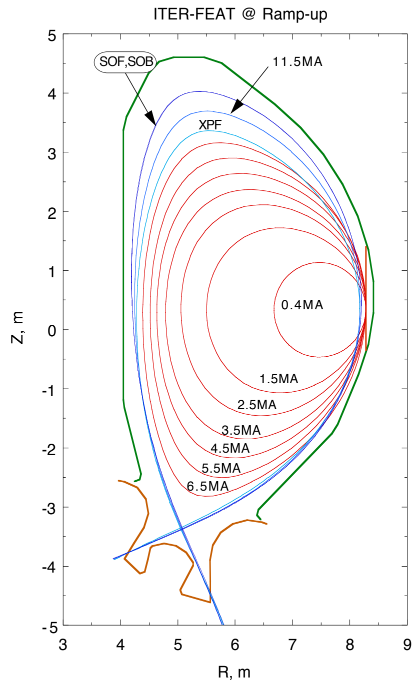

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Figure II.1.1-2 Evolution of the Plasma Boundary During the Current Ramp-up and Heating Phases

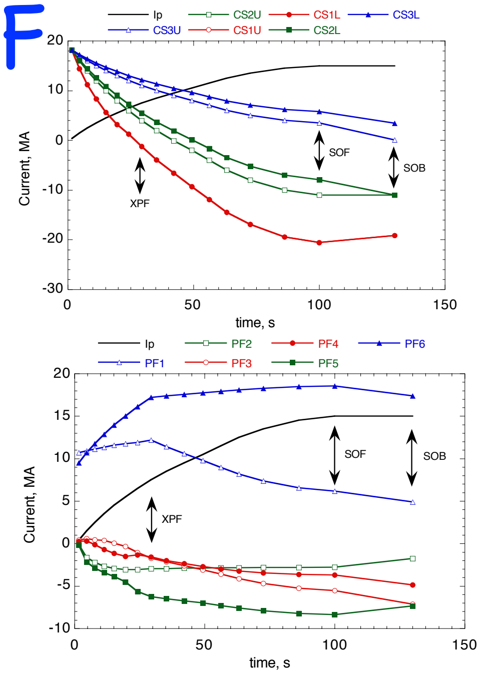

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Figure II.1.1-3 Waveforms of the Plasma and PF Coil Currents During the Current Ramp-up and Heating Phases

PF

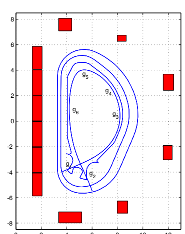

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Figure II.1.1-4 Location and Numbering of the Plasma-wall Gaps, which are Controlled

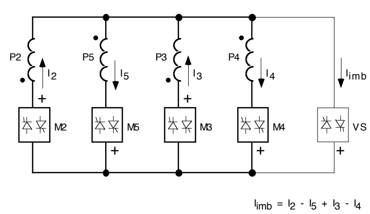

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Figure II.1.1-5 Vertical Stabilization circuit

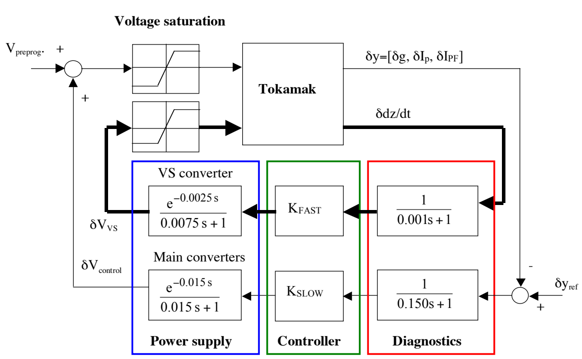

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Figure II.1.1-6 Two Loop Control Scheme Adopted (fast loop for vertical stabilization in bold arrows)

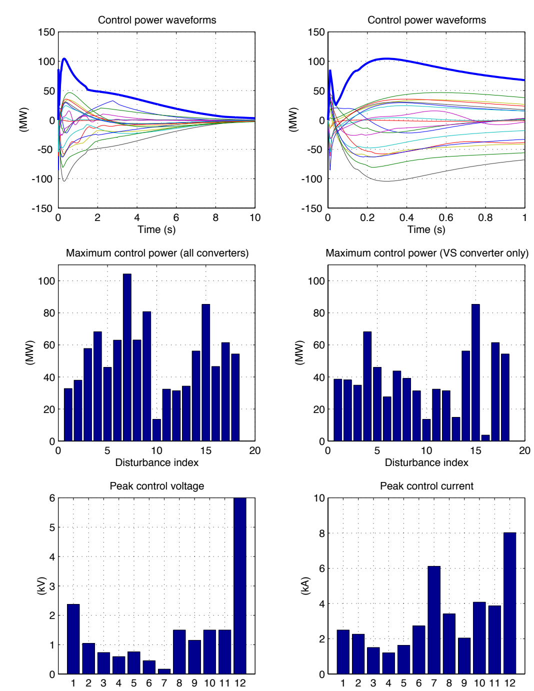

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Figure II.1.1-7 Control Power Waveforms (top) and Peaks (centre) Relative to the Disturbances Listed in Table II.1.1-5. The bottom frames show the peak control voltage (left) and current (right) on the main converters (1 through 11) and on the VS converter (index 12). The main converters are identified as follows: index 1 refers to the series CS1U&L, 2 to CS2U, 3 to CS2L, 4 to CS3U and 5 to CS3L. Indices 6 through 11 refer to PF1 through PF6 respectively

Cryo-plant ...

(like LHC, etc.)

Table II.1.4-1 Cooling Capacity of the LHe Plant

Liquefaction to cool the current leads	kg/s	0.066
Static heat load	kW	9.3
Averaged pulsed heat load [1]	kW	19
Heat loads of helium circulating pumps	kW	14.1
Heat load of cold compressors	kW	7.5
Torus cryopumps including liquefaction for fast cooldown during their regeneration [2]		4 kW + 0.06 kg/s
Small cryogen users	kW	1.0
Total		54.9 kW + 0.126 kg/s

[1] Pulsed heat loads are shown for the plasma scenario with a pulse repetition time of 1800 s and 500 s plasma burn phase.

[2] Initially 6 cryopumps will be installed. For steady state operation, 4 additional cryopumps will be added to the 6 cryopumps in order to allow continuous operation in such a way that at any time, 6 cryopumps are pumping and 4 cryopumps are under four different stages of regeneration.

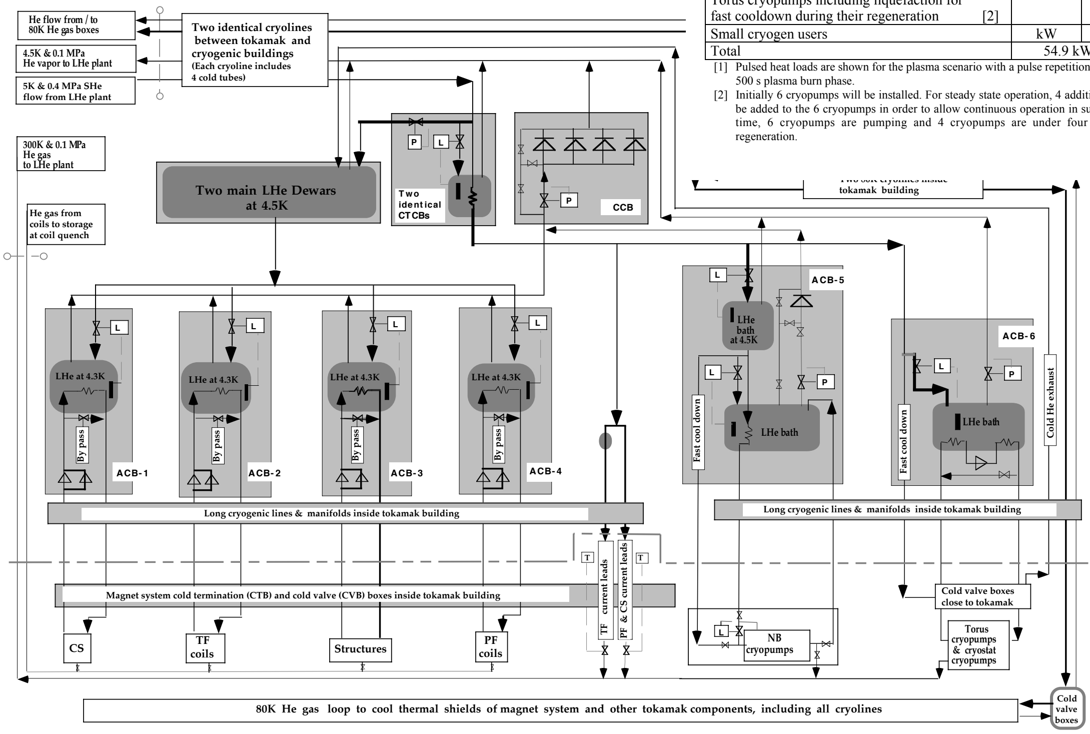

Figure II.1.4-1 Principal schematic of the cryodistribution system

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Summary

- ITER will be the largest superconducting magnet set ever built, and
- ITER is a major demonstration of superconducting magnet technology