THE FIRST FUSION REACTOR: ITER

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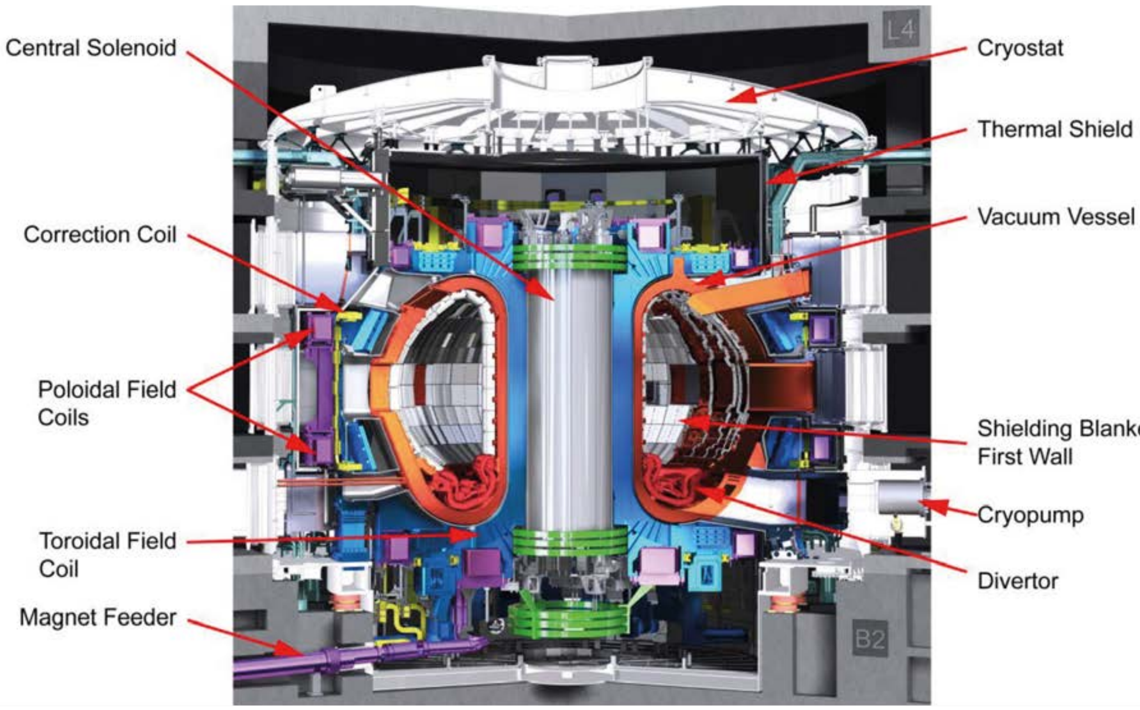
Established by the signature of the ITER Agreement in November 2006 and currently under construction at St Paul-lez-Durance in southern France, the ITER project [1,2] involves the European Union (including Switzerland), China, India, Japan, the Russian Federation, South Korea and the United States. ITER ('the way' in Latin) is a critical step in the development of fusion energy. Its role is to provide an integrated demonstration of the physics and technology required for a fusion power plant based on magnetic confinement.

In practical terms, the project's goal is to construct and operate a tokamak experiment which can confine a deuterium-tritium plasma in which the α-particle heating dominates all other forms of plasma heating. Formally, the primary mission of the ITER project is to demonstrate sustainment of a DT plasma producing ~500 MW of fusion power for durations of 300 - 500 s with a ratio of fusion output power to input heating power, *Q*, of at least 10. ITER is also designed to explore the physics basis for continuous operation of fusion power plants by investigating 'steady-state' plasma operation by means of non-inductive current drive for periods of up to several thousand seconds while maintaining a fusion gain, *Q*, of ~5. If plasma confinement characteristics are favourable, ITER would also be capable of exploring the 'controlled ignition' regime of tokamak operation (with $Q \sim 30$) in which power plant plasmas are expected to operate. The project's technical goals encompass significant technological demonstrations to prepare the design basis for a fusion power plant.

The unique nature of the ITER international collaboration is reflected in the scheme by which the components for the tokamak and auxiliary plant are being constructed. The ITER Organization (IO-CT) in France is responsible for design integration, procurement of components amounting to about 10% of the project's capital construction cost, management of the on-site installation of the tokamak and plant, and, ultimately,



▸ **FIG. 1:** Cutaway view of the ITER tokamak: the cryostat is about 29 m in diameter and 29 m high. © 2016, ITER Organization.

for the operation of the facility. The seven ITER partners have each established Domestic Agencies (IO-DAs) through which 90% of the facility's components are being procured 'in-kind' and supplied to the IO-CT for integration into the ITER facility.

ITER Design, Manufacturing and Construction

The engineering design for ITER has been developed around a long-pulse tokamak with an elongated plasma shape and a single-null poloidal divertor. The design has been validated by wide-ranging physics and engineering R&D: it is based on scientific understanding and extrapolations derived from extensive experimental studies in tokamaks in the international fusion research programme spanning several decades (*e.g.* [3]) and on the technical know-how flowing from the fusion technology R&D programmes in the ITER Members (*e.g.* [4]). A schematic of the ITER tokamak is shown in Fig. 1 and the principal parameters are listed in Table 1.

TABLE 1. MAIN PARAMETERS OF THE ITER TOKAMAK

Plasma current (I_p)	15 MA
Toroidal field (at $R = 6.2$ m)	5.3 T
Major/ minor radius (R / a)	6.2/ 2 m
Plasma elongation/ triangularity (κ / δ)	1.85/ 0.49
Installed auxiliary heating power	73 MW
Fusion power (at $Q = 10$)	500 MW
Pulse duration (at $Q = 10$)	~400 s

ITER is a superconducting device with several major magnet systems [5]: the 18 toroidal field (TF) coils and 6 central solenoid (CS) modules are fabricated from Nb_3Sn superconductor due to the high fields required, *e.g.* 13 T in the centre of the CS. The 6 poloidal field (PF) magnets use NbTi superconductor, as do the 18 correction coils (CC). The international collaboration formed around the production of superconducting magnets for the ITER tokamak has produced over 600 t of Nb_3Sn (increasing annual world production by approximately a factor of 10) and almost 250 t of NbTi superconducting strand. Over 80% of the superconductors required for the ITER magnets are now complete, and coil fabrication activities are underway in 6 of the 7 partners' factories (*e.g.*, Fig. 2(a)). Series production of the high temperature superconducting current leads will be launched during 2016. Operation of the magnet systems, which are cooled by supercritical helium, will be supported by the world's largest single-platform cryogenic plant.

Fabrication of the vacuum vessel, a double-walled stainless-steel toroidal chamber with an outside diameter of ~19.5 m and a height of ~11.5 m, is advancing, with structures being produced under the responsibility of four Domestic Agencies (*e.g.*, Fig. 2(b)). The first elements of the cryostat (~29 m diameter × ~29 m height – the largest stainless-steel high vacuum pressure vessel in existence when complete) have been delivered to the ITER site. In-vessel components such as the stainless-steel divertor cassettes (54 make up the entire divertor structure), stainless-steel shielding blanket modules (440 cover almost the entire first wall) and the associated (tungsten) divertor and (beryllium) first wall plasma facing components (PFCs) are undergoing prototyping and, in the case of the PFCs, high heat flux testing to their rated performance.

ITER will be equipped with a significant heating and current drive (H&CD) capability. This will consist initially of 33 MW of (negative ion based) neutral beam injection using 1 MeV deuterium, 20 MW of electron cyclotron resonance heating operating at 170 GHz, and 20 MW of ion cyclotron radiofrequency heating operating in the range 40-55 MHz. These systems are required for plasma initiation, heating of the plasma to temperatures at which fusion reactions can be initiated, controlling the fusion burn, provision of a substantial fraction of the non-inductive current drive for steady-state operation, control of the plasma current profile to avoid magnetohydrodynamic (MHD) instabilities and direct suppression of growing plasma instabilities. An extensive diagnostic capability consisting of about 50 large-scale systems will provide plasma measurements for control, investment protection and physics studies of burning plasmas, while a sophisticated control, data acquisition and command system will support all aspects of facility operation and protection.

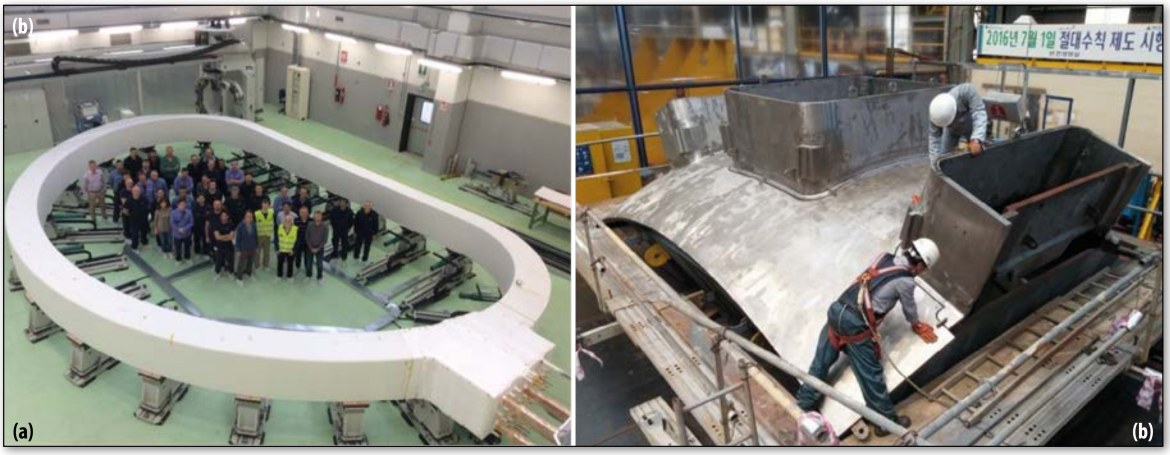
On-site construction of the ITER facility is advancing rapidly, as illustrated in Fig. 3.

Physics challenges for burning plasma studies in ITER

Successful operation of ITER will open new frontiers in fusion research involving the influence of a significant α-particle population on plasma heating, transport processes and stability. Moreover, to sustain high fusion power, it will be critical to control the exhaust of power and particles from the plasma to prevent overheating of plasma facing surfaces.

ITER operation will evolve through several stages (*e.g.* [6]): a period of hydrogen and helium operation will be used to commission all tokamak and auxiliary systems; this will be followed by a short period of deuterium operation to approach thermonuclear conditions more closely and to phase gradually into full DT operation, which will then provide access to significant levels of fusion power and α-particle heating.

Three 'design basis' scenarios have been assembled from the physics basis developed by the international fusion research community in recent decades. These reference scenarios provided a conceptual basis for the ITER design and form idealized targets for the various



▸ **FIG. 2: (a)** The first complete TF coil winding pack in the manufacturer's factory, fabricated from Nb_3Sn superconductor, with overall dimensions of 13.8 m × 8.7 m. © F4E, 2016; **(b)** An outer equatorial segment (40° toroidal, inner shell) of the ITER stainless-steel vacuum vessel with 3 port plug stubs under manufacture. © ITER Korea, 2016.

modes of plasma operation which will be explored in ITER. Their basic parameters are summarized in Table 2.

The inductive scenario is expected to provide the simplest route towards the achievement of high fusion power and to allow the first studies of substantial α-particle heating. The 'hybrid' scenario provides a relatively simple basis for technology testing under long-pulse stationary conditions.

Fully non-inductive operation is an altogether more complex plasma regime in which the total plasma current is driven by a combination of auxiliary heating systems and internal processes (bootstrap current). While the basic principles of operation in this regime have been understood for the past 20 years, and experimental demonstrations of candidate modes of operation have been made for periods of several seconds, considerable research is required, both in existing devices and, eventually, in ITER, to establish an operational mode in which all requirements of plasma confinement and stability are satisfied.

Like JET and all relevant fusion devices, ITER will also be equipped with a divertor, located in the lower region of the vacuum vessel. The material and geometry of the divertor surfaces are designed to handle high heat fluxes while allowing extraction of helium 'ash' produced by DT fusion reactions. The operational functions of the divertor are well-established in existing experiments, but the critical step that ITER will make is to integrate this power-handling strategy with a burning plasma core in such a way that the core and edge plasmas perform as intended, in benign coexistence.

Table 2. Key parameters of ITER reference scenarios

	Inductive	Hybrid	Non-inductive
Plasma current (MA)	15	13.8	9
Energy confinement time, τ_E (s)	3.4 [1]	2.7 [1]	3.1 [2]
Fusion power (MW)	500	400	360
Q	10	5.4	6
Burn duration (s)	300 – 500	>1000	~3000

[1] scaled from present experiments, [2] required to achieve $Q \geq 5$

A key aspect of this solution to the power and particle exhaust challenge is the choice of plasma facing materials. Two metals, beryllium and tungsten, have been chosen, with the former lining the first wall of the main plasma chamber and the latter covering the divertor surfaces. This material combination has been tested on JET in the frame of the ITER-like wall (see article by L. Horton).

Fusion Technology at ITER

The development and testing of key 'fusion' technologies required for construction of a fusion power plant is a principal mission goal of the ITER project. A significant element of this research is the Test Blanket Module (TBM) Programme [7], which will involve the construction and testing, by exposure to ITER plasmas, of 6 different concepts of tritium breeding module. The breeding of tritium, by reactions between neutrons emitted from the plasma and lithium contained in either ceramics or (Li-Pb) eutectics within blanket modules lining the reactor wall, is fundamental to the fuel cycle in a fusion power plant burning deuterium and tritium. While ITER can be fuelled by tritium from external sources in the fission programme, it is designed to conduct the first tests of concepts for tritium breeding which could be applied in a DEMO reactor.

The primary research goal will be to confirm the rate at which tritium can be produced: in DEMO, the 'tritium breeding ratio', defined as the ratio at which tritium is bred against the tritium burn rate, must certainly exceed unity. ITER tests will allow the first studies of the tritium production and extraction rates which can be achieved in a practical design.

Once ITER makes the transition to routine DT operation, the fusion power level, burn duration and duty cycle required will necessitate real-time reprocessing of the tokamak exhaust gas stream to provide DT fuel at an adequate rate to sustain the planned experimental programme. While a significant quantity of tritium can be stored on the ITER site, this inventory will be recycled, resulting in as much as 25 times this amount of tritium being reprocessed annually to maintain the ITER experimental programme at the required performance level. This will require a tritium processing plant of unprecedented scale, and its operation will establish the technical basis for tritium reprocessing in fusion power plants.

The development and application of remote handling technology for ITER will also provide a substantial basis for the future application of this technology in the fusion environment. Soon after the transition to DT operation, activation of the ITER tokamak due to the interaction of 14 MeV neutrons with the reactor structure will require that all maintenance, repair and upgrade work in the tokamak core be carried out using remote handling methods.

A final significant facet of the ITER nuclear R&D programme, and a key ITER mission, will be the demonstration of the environmental and safety advantages of fusion energy. After the submission of the formal application documents and an extensive interaction between the ITER Organization and the French nuclear regulatory authorities, the French Government granted the Decree of Authorization of a nuclear facility to the ITER Organization in November 2012. ITER is now established as Basic Nuclear Installation 174 (INB-174) within the French regulatory framework.

Towards ITER Operation

Construction of the ITER facility is now moving forward rapidly and the ITER partners have recently agreed to work together towards a First Plasma date of December 2025. DT operation is expected to begin about 10 years later. The research programme under development will establish the major lines of research within the ITER experimental plan in order to optimize the fusion performance of the device and to exploit the opportunities which ITER offers for studies in burning plasma research at the reactor scale. ■

Acknowledgments

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About the Author

David Campbell spent 14 years at JET, Europe's major fusion experiment, followed by 10 years leading the EU's physics and plasma engineering R&D activities for ITER. He joined the ITER Organization in 2007 and is currently Director of the Science and Operations Department, which is responsible for developing the ITER facility's central control systems, for conducting the project's fusion physics research and for preparing the framework for ITER operations.

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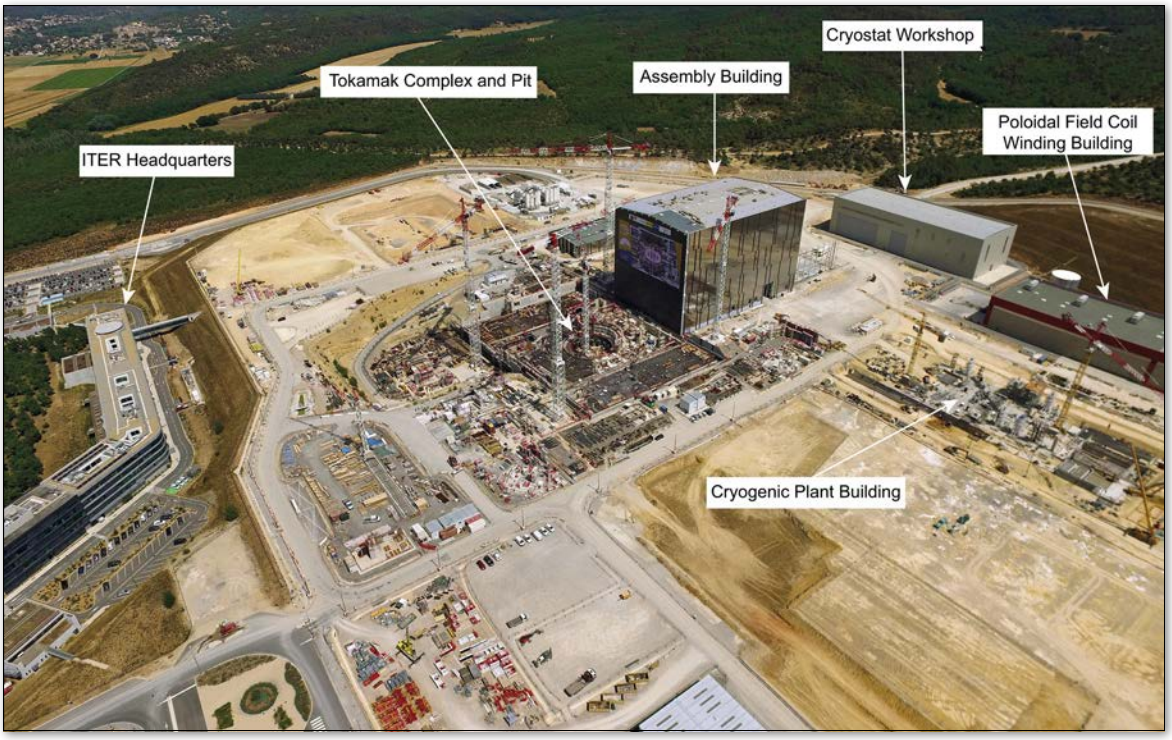

◂ **FIG. 3:** Aerial view of the ITER site showing the current status of facility construction. The Tokamak Pit, defined by the cylindrical bioshield (inner diameter ~30 m) visible in the centre of the Tokamak Complex, hosts the ITER tokamak shown in Fig. 1. © E. Riche / ITER Organization, July 2016.