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Status and Progress of ITER Project – Plasma Performance and Research Plan

David Campbell

ITER Organization
Route de Vinon-sur-Verdon, CS 90 046, F-13067 St Paul-lez-Durance Cedex, France

ITER is a critical step in the development of fusion energy: its role is to confirm the feasibility of exploiting magnetic confinement fusion for the production of energy for peaceful purposes by providing an integrated demonstration of the physics and technology required for a fusion power plant. On-site construction is advancing at St Paul-lez-Durance in Provence and manufacturing activities for major tokamak systems are underway in factories across the world. The device is designed to achieve a significant DT fusion power amplification factor of at least 10 with a fusion power output of around 500 MW for durations of several hundred seconds. In addition, the project aims to demonstrate fully non-inductive 'steady-state' operation with a fusion power amplification factor of at least 5. The projections for fusion performance and the experimental programme required to reach these goals will be discussed.

1. Introduction

The ITER Project [1], now under construction at St Paul-lez-Durance in southern France, is a partnership among seven of the world's major economic and research communities: China, the European Union (including Switzerland), India, Japan, the Russian Federation, South Korea and the United States. The ITER tokamak confines ~800 m^3 of high temperature plasma carrying a plasma current of up to 15 MA within a toroidal magnetic field of 5.3 T. It is designed to sustain 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. A cutaway view of the tokamak is shown in Fig. 1, while Table I lists the major parameters of the device.

Components of the ITER facility are predominantly (90%) constructed under the responsibility of the Domestic Agencies established by the ITER partners, while the ITER Organization, located at the ITER facility, is responsible for the overall design, assembly and operation of ITER. Manufacturing activities in the ITER Members' industries are advancing and many large scale prototypes have been fabricated to test major technologies. A significant project milestone was achieved in December 2012 when the French Prime Minister authorized the creation of the 'Basic Nuclear Installation ITER' following an extensive nuclear licensing procedure and a public enquiry. This represents a significant step both for ITER and for the development of fusion energy.

2. ITER Tokamak – Design and Technology

ITER is a superconducting device with several major magnet systems [2]: the 18 toroidal field (TF) coils and 6 central solenoid (CS) modules are fabricated from Nb_3Sn superconductor due to the high fields required, eg 13 T in the centre of the CS. Significant progress has been made in the production of superconducting strand for these magnets, with almost all of the ~500 t of Nb_3Sn strand required for the TF magnets now available. The 6 poloidal field (PF) magnets use NbTi superconductor, as do the 18 correction coils (CC) used to trim the axisymmetry of the overall magnetic field configuration.

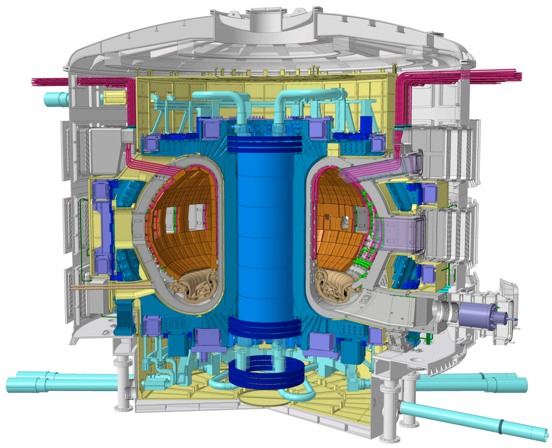

Fig.1. Cutaway view of the ITER tokamak: the cryostat is about 30 m in diameter and 29 m high.

Table I. Key parameters of ITER

Plasma current	15 MA
Toroidal field	5.3 T
Major/ minor radius	6.2/ 2 m
Plasma elongation/ triangularity	1.85/ 0.49
Pulse duration (at Q = 10)	~400 s

Other major mechanical components such as the double-walled stainless steel vacuum vessel, the cryostat, which encloses the entire tokamak core, and the (in-cryostat) thermal shield are being fabricated at factories around the world. In-vessel components such as the divertor cassettes (54 make up the entire divertor structure), shielding blanket modules, of which there are 440 covering 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.

ITER will be equipped with a significant heating and current drive (H&CD) capability to support its experimental programme. 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 In addition, an extensive plasma measurement capability consisting of over 40 large-scale diagnostic systems will be installed to support plasma control, investment protection and physics studies of burning plasmas. A sophisticated control, data acquisition and command system will support all aspects of plant operation and protection, as well as plasma control.

Testing of tritium breeding technology relevant to a fusion reactor will be carried out for the first time within the ITER project [3]. A series of test blanket modules (TBMs) constructed by the ITER Members will be exposed to ITER plasmas to test their capacity for tritium breeding and the extraction of high grade heat, which would allow electricity generation in a fusion power plant.

3. Plasma Performance

ITER is designed to exploit a range of plasma scenarios in attaining its primary physics goals. The demonstration of high power operation with $Q \geq 10$ will be developed around an ELMy H-mode plasma scenario at 15 MA, which should allow burn durations in the range of 300 - 500 s to be sustained, while fully non-inductive steady-state operation will likely be developed at a plasma current around 9 MA. In this latter mode of operation, the ITER facility is designed to allow pulse lengths of up to 3000 s. In addition, the so-called 'hybrid' mode of operation, sometimes referred to as an 'advanced inductive' mode, should also be accessible, potentially allowing pulse lengths beyond 1000 s with Q values in the range of 5 - 10.

The implementation of a sophisticated plasma control system will be a critical element of the ITER operational framework in order to ensure reliable operation at high fusion power [4]. This system will not only provide feedback control of key plasma parameters, but also control of the plasma burn, active control of a range of MHD instabilities (eg sawteeth, neoclassical tearing modes, edge localized modes, resistive wall modes) and disruption mitigation, and will also incorporate the capability of responding to changes in plasma state or the operating environment so as to sustain a well-defined plasma regime or to provide a controlled plasma shutdown.

4. ITER Research Plan

To elucidate the strategy for a rapid transition to DT burning plasma experiments during the Operations Phase, an ITER Research Plan (IRP) has been developed which details the logic of the experimental programme, defines key milestones and evaluates the major physics risks [5]. The IRP defines several phases of ITER operation, leading through non-active plasmas in hydrogen and helium, during which all ITER subsystems will be commissioned with plasma and the required plasma control capability will be developed, deuterium operation, which will probably allow the first experiments with hydrogenic H-modes, and DT operation, where the highest priority goal will be the rapid demonstration of burning plasma operation at several hundred MW of fusion power with $Q \geq 10$. The key elements of this plan will be presented and the major physics R&D issues which must be addressed in the current fusion programme to support the preparations for ITER operation will be discussed.

Acknowledgments

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The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

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