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The present status and future prospects of the ITER project

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Abstract

ITER will be the first magnetic fusion experiment to operate under reactor-relevant plasma conditions and to demonstrate the key technologies needed for a practical source of electrical power. Regarding components in high radiation fields, ITER will reach a 14 MeV neutron power load on the first wall averaging 0.5 $\mathrm{MW\,m^{-2}}$ and will accumulate a lifetime average fluence of >0.3 $\mathrm{MW\,a\,m^{-2}}$. The ITER design was completed in July 2001 to the extent necessary to obtain a realistic cost estimate. Since then, technical work has focused on developing detailed specifications for the most urgent procurements. Meanwhile, negotiations on the joint construction of ITER should lead in early 2004 to signature/ratification of a Joint Implementation Agreement. The ITER International Fusion Energy Organisation (IIFEO) can then begin work during 2004, leading to the first plasma in 2014. This paper highlights ITER materials issues, and gives the present status of technical preparations and negotiations.

1. Introduction

Nuclear fusion offers a safe, long term source of energy with abundant resources and major environmental advantages. During operation, there would be almost no contributions to greenhouse gases or acidic emissions. Also, magnetic fusion has a number of favourable safety characteristics, including:

- the nuclear fusion reaction is self-limiting in terms of the pressure of the burning plasma;
- the fusion power density and radioactive decay heat densities are moderate, and thus fast-acting emergency cooling systems are not required;
- the ultimate assured performance of confinement barriers in accidents involving tritium release for a fusion reactor needs to be about one order of magnitude reduction – by comparison, barriers providing six to seven orders of magnitude reduction are required for iodine and rare gases in fission power reactors;
- although activation products are produced by the interaction of fusion neutrons with structural materials surrounding the plasma, with the successful development of appropriate materials, tailored to minimize induced radioactivity, the wastes from fusion power would not need to be isolated from the environment for more than one hundred years.

With a successful demonstration of key fusion technologies, and further optimization of the fusion power-plant concept, fusion is expected to have costs in the same range as other long term energy sources. Thus, fusion power plants could eventually be deployed to provide a significant fraction of world electricity needs. By maintaining adequate fusion R&D, and demonstrating feasibility early in the 21st century, a power plant producing electricity could be introduced before 2050. Later in the 21st century, fusion can provide an attractive centralised energy production option to society. Equally important, fusion could begin to be deployed at a time when climate issues are likely to have become more critical than today, and when oil and gas are in short supply.

In order to demonstrate the scientific and technological feasibility of fusion energy, the ITER Project was

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launched in 1988. The Engineering Design Activities of ITER (ITER EDA) were concluded successfully in 2001. These provided a mature design, cost estimates and safety analysis supported by a body of validating physics and technology R&D [1]. Following the EDA, negotiations were launched between the Governments of the three then current Parties (Euratom, Japan and the Russian Federation), and Canada who first proposed a site for ITER. These negotiations, which have now been joined by the People's Republic of China, the United States of America, and the Republic of Korea, are currently progressing towards a Joint Implementation Agreement for ITER construction and operation, and are expected to be concluded at the beginning of 2004. This would mean that ITER operation could start as early as 2014.

This paper gives an overview of the status and prospects of the ITER Project, paying particular attention to developments in the materials area.

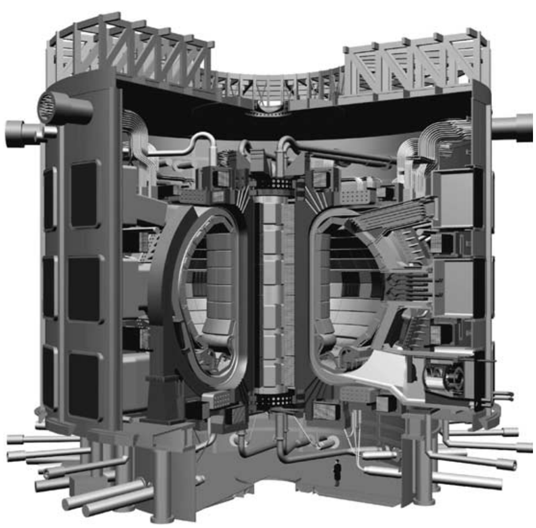

Fig. 1. ITER tokamak cutaway.

2. Design overview

The major parameters of ITER and a cutaway view of the tokamak are shown in Table 1 and Fig. 1, respectively. This device is designed to satisfy the following requirements

(1) to achieve extended burn in inductively driven plasma at $Q \geqslant 10$ ($Q =$ fusion power/auxiliary heating power), whilst not precluding the possibility of controlled plasma ignition ($Q = \infty$);
(2) to aim at demonstrating steady state operation using non-inductive current drive at $Q \geqslant 5$;
(3) to demonstrate availability and integration of essential fusion technologies;
(4) to test components for a future power reactor including high temperature tritium breeding blanket module concepts, with the 14 MeV neutron power load on the first wall $\geqslant 0.5$ MW m^{-2} and fluence $\geqslant 0.3$ MW a m^{-2} relying on tritium provided externally.

Table 1
Major plasma parameters and dimensions

Total fusion power (MW)	500 (700)
Q-fusion power/additional power	$\geqslant 10$
Average neutron wall load (MW/m^2)	0.57 (0.8)
Inductive burn time (s)	$\geqslant 400$
Major radius (m)	6.2
Minor radius (m)	2.0
Current (MA)	15 (17)[a]
Elongation (separatrix)	1.85
Triangularity (separatrix)	0.49
Safety factor (95% flux surface)	3.0
Toroidal field @6.2 m radius (T)	5.3
Plasma volume (m^3)	837
Plasma surface (m^2)	678
Heating/current drive power (MW)	73[b]

[a] Attainment of this current, with the other parameters shown in parentheses places some limitations over other parameters (e.g., pulse length).

[b] Total plasma heating power up to 110 MW may be installed in subsequent operation phases.

2.1. Tokamak housing

The tokamak is housed in a cylindrical pit whose concrete walls form the bioshield. The coil cryostat, a reinforced, stainless steel, single-shell cylinder 24 m high, 28 m diameter with flat ends, is located just inside the bioshield.

2.2. Magnet system

Eighteen Nb_3Sn toroidal field (TF) coils form a wedged vault over their straight section, and a toroidal shell-like structure reacts overturning moments and circumferential torques. The poloidal field coil system consists of six, vertically-stacked modules for the Nb_3Sn central solenoid and six NbTi poloidal coils and, with the thick metal wall of the vacuum vessel, provides the necessary control of the required plasma configurations. The NbTi correction coil system outside the TF coils – poloidally three sets and toroidally six sets of saddle-shaped coils – will correct error fields down to the order of $10^{-5}B_T$ ($B_T =$ on-axis toroidal field) and will stabilize resistive wall modes (RWMs) with feedback control, as necessary. RWMs will most likely appear at higher plasma pressure, during non-inductively driven pulses.

2.3. Vacuum vessel and ports

The vacuum vessel consists of stainless steel double walls. Large ports are attached to the vacuum vessel, nine divertor ports, 18 equatorial ports and 18 upper ports. In particular, 14 of the equatorial ports have large openings (1.8 m $\times$ 2.2 m) and access for the heating and current drive systems, diagnostics and test blanket modules, as well as for remote handling tools.

2.4. In-vessel components and plasma-facing components

All in-vessel components consist of modules which are attached to the vacuum vessel wall, but replaceable by remote handling techniques. The blanket module first wall is attached to an underlying shield blanket module body, and the divertor plasma-facing components to a divertor cassette body. Therefore, all first wall and plasma-facing components can be replaced in a hot cell, and the amount of radioactive wastes can be minimized during operation. The maximum allowable divertor target heat flux is 20 $\mathrm{MW\,m^{-2}}$. Shield blanket modules and divertor cassettes can also be modified, if necessary, to optimise the choice of plasma-facing materials, for instance.

2.5. Heating and current drive systems

Initially a combination of heating systems capable of coupling 73 MW to the plasma will be installed, consisting of 33 MW of neutral beam (NB) injection and 40 MW of radiofrequency (RF) wave power (ion cyclotron (IC) 20 MW and electron cyclotron (EC) 20 MW). Several upgrade scenarios have been investigated. The ultimate total installed power is limited to about 130 MW, either NB = 50 MW/RF = 80 MW or NB = 33 MW/RF = 100 MW. However, the total simultaneous heating and current drive power is limited in terms of initially installed power supplies to 110 MW.

The range of IC frequencies (40–56 MHz) covers various heating and current drive modes. The EC system (170 GHz) will be installed in an equatorial port with toroidally steerable mirrors, for core heating and current drive, and in upper ports with poloidally steerable mirrors, for localised current drive mainly for the stabilization of neo-classical tearing modes, a localised thickening of the nested toroidal magnetic surfaces which occurs at higher plasma pressure. The lower hybrid (LH) with 5 GHz system will provide an efficient current drive in a far off-axis area. The vertical injection position of NB with 1 MeV is adjustable in a range of +0.15 to −0.42 m relative to the plasma axis and this allows current drive near-axis or off-axis. This flexible arrangement of the heating and current drive systems enables a large variety of heating and current drive scenarios to be explored.

2.6. Fuelling and pumping

Gas puffing from the top, and high field side pellet injectors, will provide fuel. Fuel deposition for high field side injection occurs between 70% and 90% of the minor radius with a pellet size of ~5 mm and a pellet speed of ~300 m/s. Deeper penetration can be achieved for lower plasma current. Large divertor pumping capability is provided, e.g., 120 $\mathrm{Pa\,m^3/s}$ and 50 $\mathrm{m^3/s}$. This should be sufficient to control poisoning of the plasma core by impurities and helium ash buildup.

3. Plasma performance

In ITER, for the first time in the history of the development of fusion as an energy source, Q will be much greater than unity. The plasma behaviour will be dominated by self-heating from alpha particles generated in the reaction of deuterium and tritium fuel, rather than external auxiliary heating. In the reference inductively driven scenario with a low level of edge-localised plasma disturbances (ELMs) and in the high confinement mode in which the experimentally well-verified basic barrier to transport at the plasma edge (H-mode) is established, fuelling a high triangularity plasma from the inboard side would provide $Q > 10$ at densities approaching the lowest density limits observed experimentally. A typical range of performance is shown in Fig. 2.

The typical inductively driven burn time of ITER will be around 7 min but, for testing of steam-raising tritium breeding test blankets on ITER, longer burn time would be desirable as this would be more typical of power reactor conditions. Typical performance projections of ITER, as inductive current drive is increasingly replaced by non-inductive, so-called hybrid operation, stretching the burn time and ultimately leading to steady state

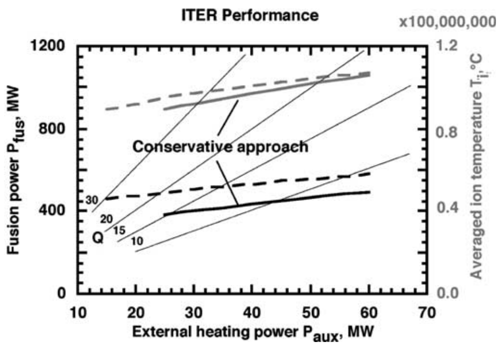

Fig. 2. Operating space for inductive operation. The solid and broken lines cover a range of assumptions about the relative confinement of fuel and helium ions within the plasma core.

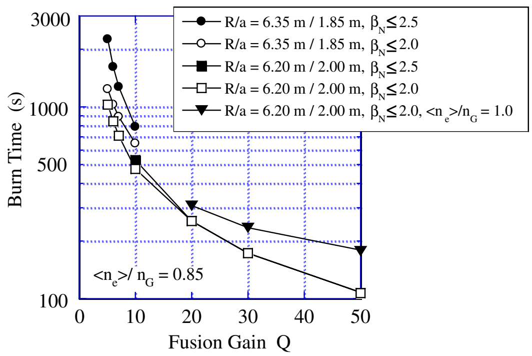

Fig. 3. Non-inductive performance predictions under nominal conditions.

operation, are shown in Fig. 3. The range of performance shown follows the predictions and limits (e.g. in plasma density) of the bulk of today's experiments. However there is experimental evidence that if the power entering the plasma core is sufficiently high, a second transport barrier occurs inside, limiting even more the heat conduction across the plasma. The control of such a barrier would enhance energy confinement, and would further reduce the need for external non-inductive drive systems, and enhance hybrid and steady state operation beyond that shown.

The ITER tokamak therefore gives confidence that it will achieve its basic level of performance, and it has considerable flexibility in plasma operations, which will allow it to provide a wide range of opportunities to develop and optimise inductive and non-inductive reactor core plasmas and to study burning plasma physics.

4. Technology R&D

Once deuterium operation has begun, it will be extremely difficult to repair the basic machine or to modify the main characteristics of ITER because of the activation. For this reason, maximum reliability has been incorporated into the design as well as flexibility. Therefore, the overall philosophy for the ITER design has been to use established technological approaches verified through detailed analysis and to validate their application to ITER through technology R&D, including fabrication of full scale or scalable models of all key components of the tokamak. In practice, ITER R&D focused on seven large R&D projects:

(1) Central Solenoid Model Coil (CSMC),
(2) Toroidal Field Model Coil (TFMC),
(3) Vacuum Vessel Sector,
(4) Blanket Module,
(5) Divertor Cassette,
(6) Blanket Remote Maintenance,
(7) Divertor Remote Maintenance

as well as a number of smaller but important projects in other areas, such as the development of radiofrequency power sources and the main torus cryopumps.

Major developments, fabrication and tests have been completed [2]. The technical output from the R&D has largely validated the technologies and confirmed the manufacturing techniques and quality assurance measures incorporated in the ITER design. Where there have been exceptions, the R&D has pointed to problem solutions. The R&D supports the manufacturing cost estimates for the key cost drivers. Valuable and relevant experiences have been gained in the management of industrial scale, international ventures. The success of these projects increases confidence in jointly constructing ITER in an international project framework.

5. Materials issues

ITER cannot provide a full test of the materials that will be needed for the subsequent power generating reactor, mainly because the fluence is limited (0.5 $\mathrm{MW\,a\,m^{-2}}$ at peak), and because the neutron power loadings are at least three times lower than needed (although the peak heat loadings are the same as would be expected) in a power reactor. However, ITER does provide the only opportunity to study materials behaviour under the relevant fusion spectrum, allowing benchmarking of higher dose fission spectrum irradiations. It also give the opportunity to study large volumes of materials under appropriate electromagnetic conditions, and identify any synergetic effects that might occur in reactor blanket components. Of particular interest is the effect of helium generation, particularly with regard to material embrittlement and components reweldability. It will be important also to coordinate the materials work to be done in ITER with that to be done in a complementary materials test facility with a high neutron fluence such as IFMIF, which is essential to establish endurance data for the coming fusion power experiments.

Most materials investigative work on ITER will be conducted as part of the integrated functional testing of test blanket module (TBM) components. TBMs with various combinations of coolants and breeder materials:

- helium-cooled ceramic breeder,
- water-cooled ceramic breeder,
- helium-cooled and water-cooled LiPb eutectic,
- self-cooled lithium,

etc., will be tested in the equatorial ports.

ITER will also be the most appropriate tool to build up operational experience on the appropriate plasma-facing materials for the subsequent devices, in particular to study erosion, deposition and tritium retention, and how to manage dust. Much of this will be carried out as an integral part of machine component operation. The modular approach allows different reactor-relevant materials to be tested, as necessary.

5.1. Vacuum vessel and in-vessel components

The main materials for these components (Table 2) have been selected during previous project phases, and for reasons discussed elsewhere [3]. The results of the ITER R&D program have indicated the feasibility of the selected materials and joining technologies to provide the required operational lifetime and structural integrity.

Currently, the materials activity is concentrated on the following main areas:

- preparation of the procurement of the materials for different components – the first priority is for materials for those items that need to start procuring earliest, such as the vacuum vessel;
- consolidation of the materials properties data (assessing the latest R&D results) and preparation of further recommendations for materials properties needed for the lifetime assessment of the in-vessel components;
- continuation of the R&D for important areas such as joining technologies (with the goal to reduce the manufacturing cost), study of those materials properties which depend on production technologies (e.g. CuCrZr alloys), and investigation of some additional materials properties needed for the design assessment.

Table 2
Vacuum vessel and in-vessel materials

Component	Material
Vacuum vessel, ports and blanket manifolds	Austenitic steel 316L(N)-IG Austenitic steel 304 Inconel 718 Precipitation hardened steel 660 Ferritic steel 430 Borated steel 304B7
First wall, blanket and limiter	Beryllium CuCrZr (or DS Cu) Austenitic steel 316L(N)-IG Inconel 718 Ti alloy T–6Al–4V Ceramic Al_2O_3 or $MgAl_2O_4$
Divertor	Carbon fibre composite Tungsten CuCrZr Austenitic steel 316L(N)-IG NiAl bronze

Procurement specifications for the main materials for the ITER vacuum vessel are being prepared in close contact with possible manufacturers. These specifications are based on the ASME/ASTM specifications and additionally include ITER-specific requirements such as chemical composition (reduction of the cobalt and niobium content, reduction of the boron for reweldable components) and some requirements for specific properties (saturation of magnetic flux for steel 430), etc. The specification for austenitic stainless 316L(N)-IG steel is being prepared using the extensive experience of the RCC-MR Code [4]. Due to an optimal combination and tight specification of the main alloying elements, such as carbon, nitrogen, nickel, chromium, manganese and molybdenum, this steel has high minimum tensile mechanical properties combined with good ductility and toughness and corrosion resistance.

The materials properties data needed for design assessments have been prepared in the following ITER documents:

ITER Materials Properties Handbook (MPH),
ITER Materials Assessment Report (MAR),
Summary of Vacuum Vessel Materials Properties,
App. A of the ITER Structural Design Criteria (SDC),
Safety Analysis Data List (SADL).

All these documents include recommendations for the materials properties that apply and assessments of the materials performance. The ITER R&D results and data from other sources (e.g. design Codes such as ASME, RCC-MR) and development programmes are the main basis for these assessments. To permit easy maintenance of the large amount of available data for the wide variety of materials shown in Table 2, the ITER Materials Properties Database (MPDB) has been reorganized. The database maintains the data in an easily accessible form and it is an important tool to maintain full traceability of the data, test details and data sources. This helps to make the recommendations highly reliable. This database includes detailed information about material physical and mechanical properties, chemical composition, production history, sample geometry, test conditions, irradiation parameters, etc. The format is simple and flexible and can be modified for specific needs. The raw data from the database can be presented on request to licensing authorities in support of approval. The MPDB includes the relevant data for relevant materials but, for design purposes, only qualified data are used. This database is the main supporting tool for recommending materials for use.

The current activity is focused on improvement of the ITER Materials Properties Handbook for vacuum vessel materials. The information is being arranged such that the recommended properties are fully traceable and prepared in accordance with internationally accepted procedures.

Meanwhile, R&D is ongoing mainly in the EU and RF. Currently, the main activities in the EU Party are focused on investigation of the effect of the manufacturing cycle on CuCrZr and CuCrZr/SS joints, CuCrZr creep-fatigue and in-pile tests. Different joining technologies are under evaluation such as high temperature HIP combined with fast cooling to provide solution annealing of CuCrZr followed by low temperature HIP for armour/Cu alloy joining, fast brazing, etc. The key issue is to avoid strength degradation after manufacture.

Data on low dose irradiation of Ti alloys for blanket support systems (the flexible attachments) are still being generated as well as data on stress relaxation in the Inconel 718 bolts that will attach the flexible attachments to the vessel. Further qualification of CFC manufactured in the EU at high temperatures (more than 2000 °C) is also being carried out. The RF is studying CuCrZr irradiation creep, the effect of manufacturing cycle on CuCrZr, DS Cu, and their joints, low dose irradiation of 'functional' materials (e.g. NiAl bronze, etc.) and in-pile bake-out of CuCrZr.

5.2. Plasma–surface interactions

Some of the most crucial plasma edge physics and plasma–material interaction (PMI) issues of the ITER tokamak, which are being addressed by R&D [5] are:

- carbon and tritium migration to remote areas;
- tritium removal;
- divertor and main chamber thermal loads during edge-localised modes (ELMs) and disruptions;
- material mixing;
- erosion and transport of W and compatibility with plasma scenarios;
- development and testing of PMI measurements/diagnostics.

One of the main challenges comes from energy deposition during type-I ELM thermal quench disruptions at the divertor and first wall [6]. In particular, there is concern that a Be first wall in ITER could be eroded due to melting if a significant fraction of the ELM and disruption energy is deposited at the wall. Clearly, further understanding of plasma physics is needed before drawing firm conclusions for ITER.

According to current ITER construction plans, a few years are still available for further R&D and physics input in the area of plasma wall interactions to validate the lifetime of armour materials and to finalise the design of the divertor/first-wall components, before procurement must start.

In addition to the issue of lifetime there are important operational issues that are being explored whatever materials combination is ultimately selected. For example, development of strategies to limit surface melting, tritium inventory and dust accumulation.

6. Negotiations and technical preparations

Since inter-governmental negotiations began in mid-2001, the Negotiators have met nine times to discuss the Joint Implementation Agreement (JIA) on ITER construction, operation and decommissioning, and related instruments. These include the requirements for the ITER construction site and host provisions, who will provide and pay for the various ITER components/systems, who will be the Director-General and senior staff for the ITER International Fusion Energy Organisation (the 'ITER Organisation') and what will be the organisation of its work. Pending a site decision, the various common factors are being developed, documented, and agreed in detail.

The ITER Organisation will be an international organisation under international law. It will be the license owner for construction and operation, and will be responsible for enforcing its terms. Therefore, it will have technical control of all procurements, provide their technical specifications, decide upon necessary design and schedule changes, and accept the final product.

The three potential ITER sites – Cadarache (France), Clarington (Canada), and Rokkasho (Japan) – have been assessed against the technical requirements, and each has been found to be acceptable, with only minor cost differences caused by siting the project in each location. The decision between them therefore cannot be made on purely technical grounds, and an agreement between governments usually takes time. To a large extent (∼80%) the cost sharing between the potential Parties to the JIA has been worked out, with the sharing of remaining items waiting for the choice of site. About 90% of the construction hardware will be provided 'in kind', by the Parties to the JIA. Currently it is planned to announce a decision on the site, senior management and cost sharing before the end of 2003. This will allow the remaining points in the JIA to be finalised, leading to signature/ratification of the JIA by mid-2004, allowing the ITER Organisation to be formally established by the end of 2004. However, with the appointment of a Director-General, recruitment of project staff can already begin, and decisions prefiguring the project organisation and operation can be implemented.

In view of the positive direction of Negotiations, at the end of 2002, ITER Transitional Arrangements (ITA) began in January 2003 and will continue until the ITER

Organisation is established. These involved and involve organisational preparations to enable the Organisation to operate effectively without delay following signature/ ratification of the JIA:

- establishing interim key elements of the organisation structure (e.g. Interim Project Team and Leader, Preparatory Committee of ITER Council, etc.);
- following site choice, establishing a provisionally staffed interim joint work site there;
- coordinating Parties' domestic preparations for contributing to ITER Joint Implementation;
- identifying and provisionally assigning potential senior staff;
- elaborating administrative procedures, documents and other tools foreseen for managing ITER Joint Implementation.
- continued joint technical preparations.

6.1. Current technical activities

These continue preparations begun during Coordinated Technical Activities (CTA) between July 2001 and the end of 2002, and are directed at maintaining the coherence and integrity of the ITER design and at preparing for an efficient start of ITER construction. This includes, but is not limited to:

- maintaining the documented design basis for ITER, in particular by reacting to the results of technology and manufacturing R&D and by exploiting physics R&D to take advantage of the latest experimental results;
- following studies of design adaptations to potential sites and their regulatory environment, site-specific design adaptation for the chosen site,
- preparing for the procurement process, by drawing up technical specifications for procurements which need to be launched early, i.e. for magnets, vacuum vessel, and buildings;
- developing, and introducing on a provisional basis, construction project management systems and other appropriate management tools;
- preparing licensing of ITER, by close (formal in two cases) dialogue with potential regulators, which requires a formal review of the design ensuring its quality and its completeness, and undertaking necessary safety analyses.

7. Conclusions

This paper has given an overview of the ITER design and has illustrated that the features of the design give confidence it will be a technical success. In particular it has focused on the developments in the materials area, emphasising the lessons learned from the R&D, and the incorporation of the results into the detailed design definition in preparation for procurement.

There is an undisputed need for a sustained burning plasma experiment and demonstration of fusion technologies at this stage of fusion development. The present ITER is the essential step needed for fusion research to advance towards the objective of becoming an energy source within a few decades.

Technical preparations are advancing to turn the design of ITER into technical reality. The inter-governmental negotiations aiming at an agreement on construction and operation of ITER are nearing completion and, if all goes well, can be expected to lead to the start of the establishment of the construction team in 2004. Assuming a 2 year period before the construction license is granted, a 7 year construction and assembly period, and 1 year of integrated commissioning of the plant, the first plasma can be expected in 2014.

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