ITER overview

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ITER Joint Central Team

ITER Joint Central Team and Home Teams

Abstract. The article summarizes six years of technical work carried out by the ITER Joint Central Team and Home Teams under the terms of the ITER Engineering Design Activities (EDA) Agreement. The major products of this effort are as follows: a comprehensive and detailed engineering design with supporting assessments, cost estimates and a schedule based on industrial studies; a comprehensive safety and environmental assessment; and technology R&D to validate and qualify the design, including proof of technologies and industrial manufacture and testing of full size or scalable models of key components. The ITER design is at an advanced stage of maturity and contains sufficient technical information to serve as a basis for a decision on construction. The operation of ITER will demonstrate the scientific and technological feasibility of fusion energy for peaceful purposes.

1. Introduction

The ITER project is being conducted under the auspices of the IAEA according to the terms of an agreement among the European Atomic Energy Community (EU), the Government of Japan (JA), the Government of the Russian Federation (RF) and the Government of the United States of America (US), referred to as the Parties.

> "The overall programmatic objective of ITER... is to demonstrate the scientific and technological feasibility of fusion energy for peaceful purposes. ITER would accomplish this by demonstrating controlled ignition and extended burn of deuterium–tritium plasmas, with steady-state as an ultimate goal, by demonstrating technologies essential to a reactor in an integrated system, and by performing integrated testing of the high-heat-flux and nuclear components required to utilize fusion energy for practical purposes." [1]

Fusion energy programmes throughout the world have benefited from a remarkable degree of openness and global co-operation which has brought with it dramatic progress in scientific understanding and performance achievement especially in the tokamak fusion area. The tokamak, a magnetic fusion concept in which a plasma carrying electric current is contained in a toroidal shaped reactor, was originally developed in the former USSR and has been intensively investigated in many countries. The leading tokamak fusion experiments, such as JET, JT-60U and TFTR, have realized their full performance potential, producing fusion power of 10–16 MW [2, 3], achieving an equivalent breakeven condition [4] and investigating operation modes which may lead to a steady state operation in ITER [5]. At the same time, supporting or specialized experiments in these and other devices, theory development and technology development are together broadening scientific understanding and establishing competence in fusion technologies. The logical next step for all the leading fusion programmes is now to progress to study the physics of burning plasmas and steady state operation, and to demonstrate and test the key fusion technologies and engineering to establish the feasibility of fusion as an energy source; ITER will fulfil this next step.

The ITER project arose from the recognition by the leading fusion programmes of the comparable positions reached in existing experiments and of the benefit to be derived from undertaking the next step jointly. Collaboration on ITER provides significant savings through sharing of costs, and more importantly the opportunity to pool experience and expertise gained over recent decades and to draw from the scientific and technological expertise of all the world's leading fusion experiments and programmes in an integrated and focused venture.

The original detailed technical objectives for achieving the overall programmatic objective of ITER were adopted by the Parties in 1992 [6]. ITER will have two phases of operation of roughly ten years each. The first phase (Basic Performance Phase) will address the issues of controlled ignition, extended burn, steady state operation and the testing of blanket modules. ITER's technical objectives require demonstration of controlled ignition and extended burn, in inductive pulses with a flat-top duration of approximately 1000 s and an average neutron wall loading of about 1 MW/m^2. ITER should also aim to demonstrate steady state operation using non-inductive current drive in reactor relevant conditions.

It is assumed that for the first phase there will be an adequate supply of tritium from external sources. The second phase (Enhanced Performance Phase) will emphasize improving overall performance and carrying out a component and materials testing programme with higher neutron fluence. Tritium breeding might be implemented for this phase. ITER must also be designed to demonstrate the safety and environmental acceptability of fusion as an energy source.

The original Engineering Design Activities (EDA) of ITER were completed by the Parties in July 1998 after six years of activities. Canada and Kazakhstan were also involved in the project through their association with the European Union and the Russian Federation, respectively. During this period, the Parties agreed to produce a detailed, complete and fully integrated engineering design of ITER and all the technical data necessary for a decision on the construction of ITER. The results of the EDA are available to the Parties to use either in international collaboration or within their domestic programmes. The results delivered at the end of the EDA met the original plan. The ITER design [7], supported by technology R&D, is at an advanced stage of maturity and contains sufficient technical information to serve as a basis for a decision on construction. Owing mainly to the delay of the construction decision, a three year extension of the EDA is foreseen. During this period, site specific design adaptations and safety analyses, preparation of licence applications, prototype testing and further physics studies, and preparation of documents for future procurement are planned. The Parties will also develop proposals and the necessary supporting information for the complete realization of ITER, including a draft agreement for construction and operation.

Owing to increasing financial constraints, it is difficult to secure a commitment to finance the construction effort at the originally agreed costs. Therefore the Parties are seeking cost reduction at the expense of assured performance. A Special Working Group of the Parties' representatives developed new technical requirements for possible changes to the original technical objectives with a view to establishing options of minimum cost that would still satisfy the overall programme objectives of the ITER EDA Agreement. The technical guidelines developed are as follows:

Plasma performance

- Achieve extended burn in inductively driven plasmas at energy gain $Q > 10$ for a range of scenarios, where Q = fusion power from plasma/auxiliary heating power to plasma.
- Aim at demonstrating steady state through current drive at $Q > 5$.
- Not preclude controlled ignition.

Engineering performance and testing

- Demonstrate availability and integration of essential fusion technologies.
- Test components for a future reactor.
- Test tritium breeding module concepts.

The new requirements are still consistent with the integrated 'one step' strategy for a demonstration power plant which produces significant electricity (DEMO). The Parties at the ITER Council in 1998 [8] approved the proposal to:

- Establish options of minimum cost aimed at a target of approximately half the direct capital cost of the present design, with reduced detailed technical objectives which would still satisfy the overall programme objectives of ITER.
- Use existing design solutions and associated R&D.

In order to select the major parameters and design features of a reduced cost ITER by the end of 1998, an intense effort of the Joint Central Team and Home Teams was carried out. The existing EDA technical output of design choices, generic technologies and major R&D results is generally directly applied to a reduced cost ITER. Therefore it will be possible to develop a reduced cost ITER in a relatively short period and the detailed design report will be available by July 2000, when the joint assessment of the ITER construction and operation by the Parties is planned.

Table 1. Nominal parameters and dimensions of ITER

Total fusion power P_f	1.5 GW
Neutron wall loading	1 MW/m^2
Plasma major radius R	8.1 m
Plasma minor radius a	2.8 m
Vertical elongation of plasma cross-section at 95% flux surface, κ_{95}	1.6
Triangularity of plasma cross-section at 95% flux surface, δ_{95}	0.24
Plasma safety factor at 95% flux surface, q_{95}	3
Plasma current I_p	21 MA
Toroidal field at 8.1 m radius/toroidal field coil	5.7 T/12.5 T
Divertor configuration	Single null
Auxiliary heating power P_{aux}	100 MW

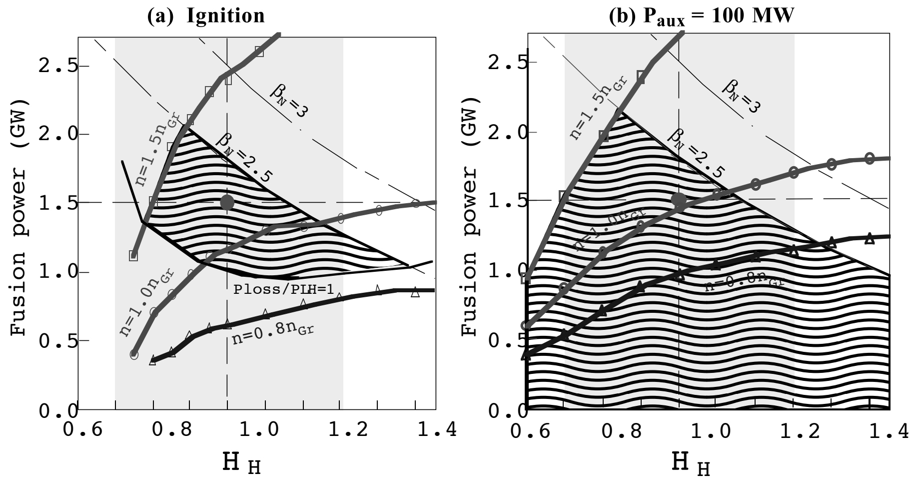

Figure 1. Operational domain at I_p = 21 MA ($P_{loss}/P_{LH} > 1$, $n/n_{Gr} < 1.5$, $\beta_N < 2.5$). Shaded regions represent typical ranges of uncertainties in confinement time.

In this article, the technical work carried out on the basis of the original technical objectives during the original EDA period (July 1992–July 1998) is summarized.

2. ITER design [9–14]

In accordance with the original plan, the main parameters summarized in Table 1 were defined after careful study of the balance between the physics requirements for plasma confinement, control and stability based on the ITER Physics Basis and Physics Rules [9, 10], and the engineering constraints, such as heat loads, electromagnetic and mechanical characteristics, neutron shielding and maintainability, necessary to ensure safe and reliable operation at a reasonable cost.

The plasma performance of ITER is assessed on the basis of the most recent experimental results and modelling. Physics issues have been identified and each of these issues has been studied thoroughly in a collaborative framework of voluntary ITER physics activities, co-ordinated through Expert Groups, which draws on the full range of physics expertise throughout the Parties' fusion programmes. The results are summarized in the ITER Physics Basis and Physics Rules [9, 10].

The three issues that most directly determine the plasma performance are summarized as follows:

(a) *Energy confinement, edge parameters, and capability to reach and sustain H mode.* A reliable database of global energy confinement times of ELMy H mode plasmas, in which ELMs produce an acceptable plasma edge condition with high confinement, is well developed. The H mode factor (H_H), which characterizes the energy confinement time in relation to its reference extrapolated value based on a reference scaling derived from the database, would be in the range 0.7–1.3, owing to uncertainties. In order to achieve and sustain this H mode, the power outflow (P_{loss}) across the magnetic separatrix surface must be higher than the threshold power (P_{LH}) for a transition from a low confinement mode (L mode) to a high confinement mode (H mode) plasma.

(b) *Plasma β (the ratio of plasma pressure to magnetic field pressure) and plasma particle density (n).* Theoretical modelling confirms that typical ITER plasma profiles are stable to ideal MHD modes for normalized β ($\beta_N = \beta$ (%) a (m) B (T)/I (MA)) as high as 3.5 (or β = 4.6%), but because of neoclassical island tearing modes, a reasonable range is up to 2.5. The plasma particle density limit in tokamaks fuelled with gas puffing is generally well represented by the Greenwald scaling, developed for the line averaged plasma density, i.e. n_{Gr} (10^{20} m^{-3}) I_p (MA)/(a^2 (m)). In present tokamaks, the limit for the line averaged density strongly depends on the density profile. The Greenwald scaling is more applicable for the edge density than for the central density. Higher volume averaged densities are possible if the central density profile can be peaked with deep fuelling of the plasma core. Therefore the possibility to achieve a density of up to $1.5n_{Gr}$ is included in the following analysis.

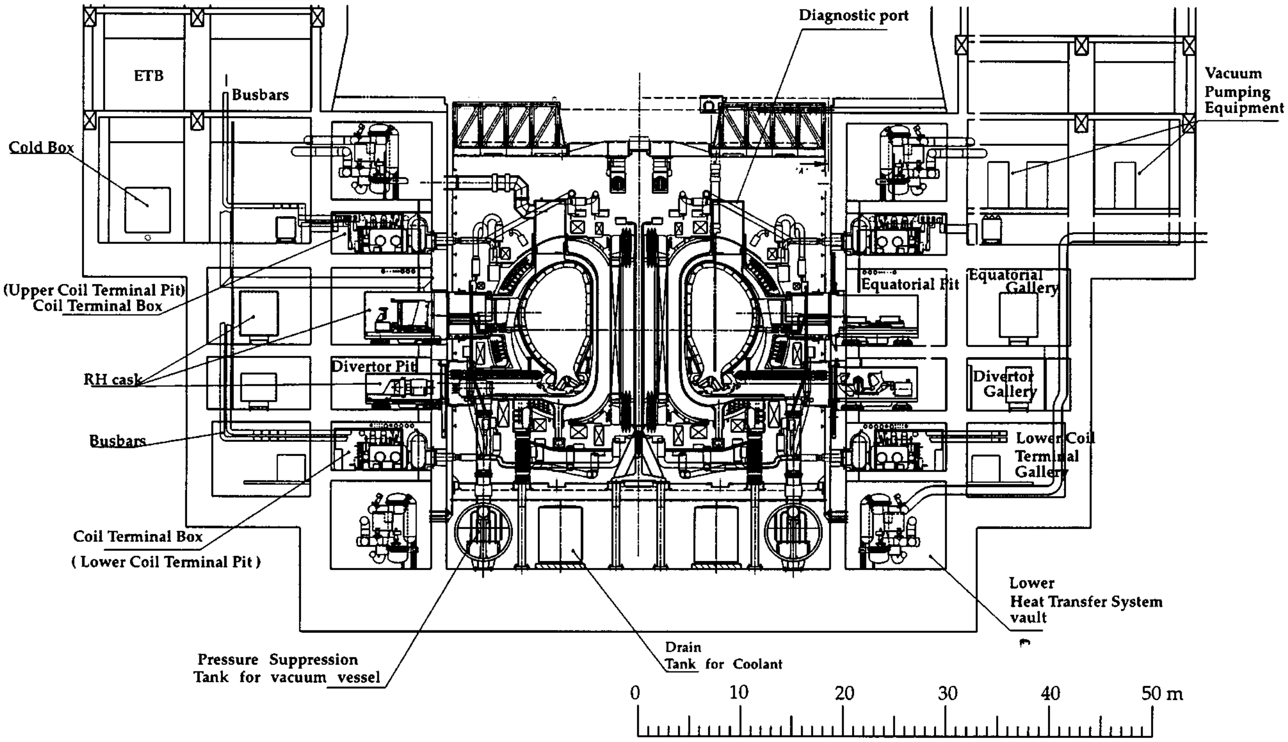

Figure 2. Elevation view of the equipment layout. RH: remote handling.

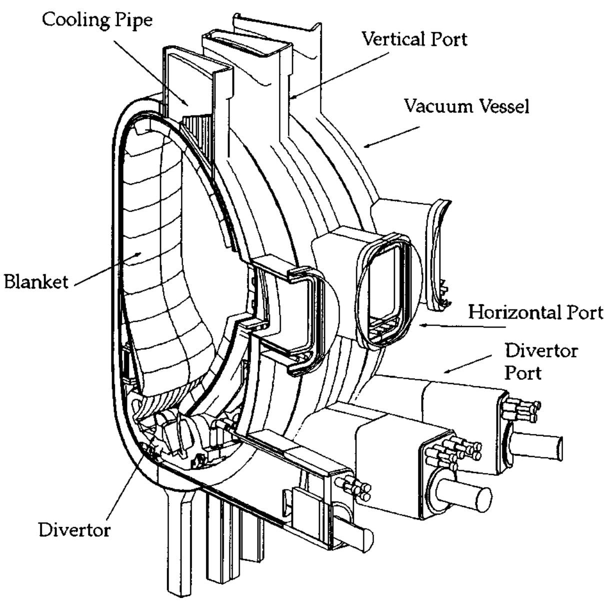

Figure 3. Isometric view of vacuum vessel, blanket and divertor.

(c) *Helium exhaust and divertor power handling.* In order to reduce heat load on the divertor target to an acceptable level, i.e. 10 MW/m^2, and to pump out helium with a reasonable pumping speed, i.e. 200 Pa·m^3/s, the diverted plasma must be cooled to a temperature of less than a few electronvolts by radiating major power in the main chamber and in the divertor channel by introducing impurities. This condition is consistently calculated in the following analysis with the required plasma performance in the main core plasma.

On the basis of these considerations, ITER performance and its nominal operational domain were studied and are summarized in Fig. 1, which plots fusion power for a 21 MA discharge as a function of H_H. The plots take into account the critical parameters, i.e. $P_{loss}/P_{LH} > 1$, $n/n_{Gr} < 1.5$ and $\beta_N < 2.5$, which are satisfied either in the ignited condition (a) or in driven mode with heating power $P_{aux} =$ 100 MW (b). In the case of ignition the available range of operational parameters around their normal values is commensurate with the possible uncertainties in extrapolation of the confinement time. In driven modes, the feasible region is extended to cover a larger range of uncertainties and $P_{loss}/P_{LH} > 1$ even at a low fusion power or a low plasma density.

The dynamic analysis and simulations indicate that time dependent requirements for plasma

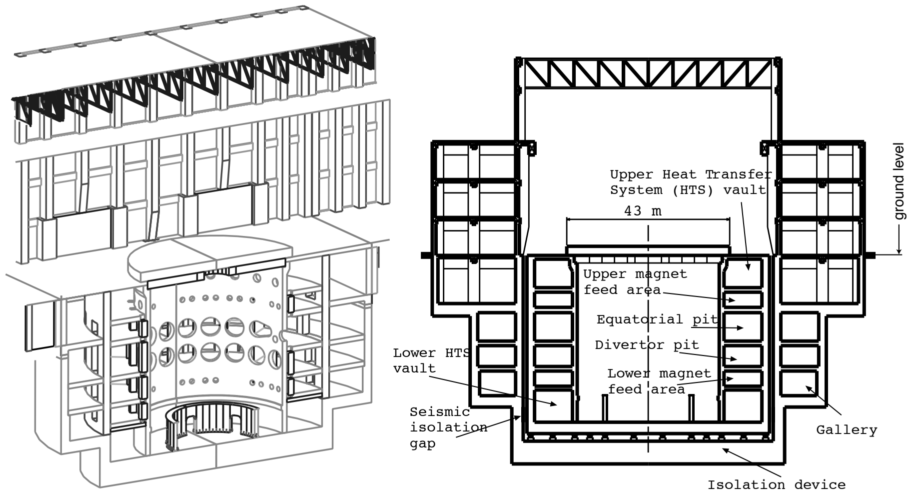

Figure 4. Tokamak building and pit. Left: cutaway view; right: cross-sectional view (in the case of seismic isolation).

operation — low divertor heat loads, helium pumping, H mode power thresholds, etc. — can be fulfilled and controlled simultaneously. The design incorporates all the requirements for the reliable operation and control of ignited or high Q driven burn DT plasmas with fusion powers in the 1–1.5 GW range and fusion burn durations of $\geq$1000 s. The nominal plasma parameters are chosen such that with 'reference' physics basis assumptions about attainable energy confinement, attainable plasma density, adequate divertor target heat load and projected plasma impurity content, sustained DT burn with power $\geq$ 1 GW is possible. Auxiliary heating and/or current drive powers of up to 100 MW are provided for the initiation of ignited burn and for the sustainment of high Q ($\geq$10) driven burn. The in-vessel plasma facing surfaces and nuclear shielding modules are designed for steady state power handling capabilities. The poloidal field coil system is sized and configured such that static and dynamic plasma equilibrium control at plasma currents of up to 24 MA is possible, and the system supplies sufficient inductive current drive to enable nominal 21 MA, 1600 s duration pulses (including a 1000 s fusion burn) to be produced. Somewhat shorter duration (500 s burn) inductively sustained pulses at 24 MA are possible. Extension of the controlled burn duration up to $\sim$6000 s in a reduced current driven burn mode is also feasible. A true steady state plasma operation with current driven by non-inductive methods in the 1 GW range of fusion power in a reverse shear configuration may also be accessible.

The ITER design is illustrated in Figs 2–4. Detailed designs for specific major components were developed, the compatibility of parts or subsystems with the whole was achieved and all outstanding design issues were resolved except site specific adaptations. On the basis of the design work and fabrication experience with the R&D components, detailed industrial-based cost estimates were performed and the costs were shown to be consistent with the originally agreed costs. The essential engineering features of the tokamak core include:

(a) An integrated structural arrangement in which the superconducting magnet coils (20 cased toroidal field coils, 9 poloidal field coils and a monolithic central solenoid) and the vacuum vessel are linked to provide an overall assembly which simplifies the equilibration of electromagnetic loads in all conditions, relying largely on the robustness of strong toroidal field coil cases (Fig. 2);
(b) Modular in-vessel components (blanket modules on backplate and divertor cassettes, shown in Fig. 3) designed to be readily and safely maintainable by a practical combination of remote handling and hands-on techniques.

The tokamak is contained in a cryostat vessel situated in an underground pit inside a building of about 50 m height (Fig. 2). Peripheral equipment, such as fuelling and pumping, heat transfer, auxiliary heating and remote handling equipment, is arranged in galleries around the main pit. The seismic condition depends on the site. If the seismic ground peak acceleration is larger than $0.2g$, isolation will be added (Fig. 4) by having a seismic gap at the pit wall, creating an isolated (64 m dia.) 'tokamak pit' supported by flexible bearings, which prohibits vertical movement but allows large horizontal movement ($\simeq$200 mm). This concept minimizes the design changes that would be required for different seismic conditions. The main services required for ITER, such as electrical power, cooling water, fuel treatment, information flow, assembly and maintenance facilities, and waste treatment, are distributed in ancillary buildings and other structures over a site of about 60 ha (6×10^5 m^2) area.

3. ITER safety [6]

The safety objectives of ITER are as follows:

- ITER shall be designed for possible construction on the territories of any of the four Parties.
- ITER shall be designed, constructed, operated and decommissioned in a manner that will ensure the protection of the public, site personnel and the environment.
- ITER should demonstrate the safety and environmental potential of fusion.

In order to ensure that ITER could be constructed by any of the Parties, it was recognized that a design was needed that would be robust to variations in safety approach and criteria. In other words, only a limited number of design changes would be needed to accommodate a Party's regulatory requirements. For this purpose, the ITER safety design guideline was developed with the participation of all Home Teams and has been implemented in the ITER design. This includes radiation dose and radioactivity release design guidelines established in accordance with internationally accepted conservative criteria and the ALARA principle (As Low As Reasonably Achievable), and the well established nuclear design concepts of defence in depth and multiple lines of defence.

A comprehensive safety assessment of the ITER design has been completed and the results include the following.

(a) A high level of safety is integrated into the ITER design. General safety design requirements were established with the participation of the Home Teams that include conservative radioactivity release limits. The design incorporates the well established concepts of defence in depth and multiple lines of defence to attain high confidence in the reliability of critical safety features of the facility and to ensure protection against postulated accidents.

(b) Radioactive effluents and emissions during normal operation are low. A comprehensive analysis of effluents and emissions shows that the total releases are well within ITER design release limits established in accordance with internationally accepted criteria and the ALARA principle. Although there are uncertainties in the release estimates, there is adequate flexibility in the design to modify atmospheric and effluent release systems for improved performance.

(c) The ITER design ensures protection of the public. A comprehensive analysis of reference event sequences has been performed using the best safety analysis computer programs available worldwide with conservative assumptions. Radioactivity releases are well within the conservatively established ITER design release limits. Two fundamentally different approaches, i.e. the bottom-up and the top-down approaches, have been applied to the identification of all potential accident sequences. It has been confirmed that the consequences of the identified sequences are included within the envelope of the assessed consequences of the reference events.

(d) A possible scenario of waste management and decommissioning has been developed. Waste streams have been studied in detail and a phased ITER decommissioning scenario has been developed that includes maximizing the use of existing facilities and equipment and taking advantage of cooldown effects. All in-vessel components can be dismantled using the remote maintenance equipment employed during ITER operation, and all ex-vessel components can be dismantled using conventional tools. Only the vacuum vessel requires remote operation for dismantling or mothballing for later dismantling with human access. The in-vessel components will form the major part of the final radioactive waste.

(e) A basic strategy of occupational safety has been developed. Radiation protection and ALARA analyses have been started. Throughout the life cycle of ITER, the radiation protection programme will continue to be updated and occupational safety considerations will be incorporated in the future design work.

(f) Ultimate safety margins are high. Ultimate safety margins have been studied in order to demonstrate the intrinsic positive safety characteristics of magnetic fusion.

- The fusion reaction is self-limiting, bounded by the β limit of the plasma. Under any failure conditions of the vacuum vessel or the in-vessel components, fusion reactions are physically impossible.
- The radioactive inventory is moderate and the ultimate performance of confinement barriers that needs to be ensured in the case of accidents will be to achieve a reduction of about one order of magnitude for tritium and dispersible metallic dust for ITER, whereas a reduction of six to seven orders of magnitude is required for iodine and rare gases in fission power reactors.
- Radioactive decay heat densities are moderate. Therefore structural melting of the plasma vessel is physically impossible and fast acting emergency cooling systems are not required.

Hypothetical accident sequences are investigated that would challenge the line of defence associated with the failure of safety functions and potential energy sources, i.e. decay heat removal or confinement barriers, coolant energy and related overpressure, plasma energy due to failure of power shutdown, steam–first wall material reaction due to overheating, hydrogen explosions and magnetic energy. It is shown that there is no technical justification for evacuation of the public even under the worst case conditions of these hypothetical accident sequences. This result derives mainly from the intrinsically positive safety and environmental characteristics of fusion, such as passive fusion power shutdown, fusion power limitation, modest radioactivity inventories, modest radioactive decay heat, and conservative structural design of the tokamak system, including the multiple boundaries of the magnetic fusion reactor, i.e. the vacuum vessel connected with the pressure suppression tank and the cryostat. These features will be common in magnetic fusion reactors and ITER has the same size, power and radioactivity inventory level as a fusion power plant. Therefore these favourable safety characteristics of ITER show fusion's safety and environmental potential.

The safety assessment concludes that ITER could be constructed and operated without serious risk to health and safety, and without significant environmental impacts. A study of the ultimate safety margins shows the favourable safety characteristics of magnetic fusion. Home Team Expert reviews indicate that a technical basis on which to start a discussion with regulatory authorities has been well developed.

4. ITER technology R&D

The overall philosophy for ITER design has been to use established approaches and to validate their application to ITER through detailed analysis and by making and testing large/full scale models and prototypes of the critical systems. The major technical challenges in ITER are:

- The unprecedented size of the superconducting magnets and structures;
- High neutron flux and high heat flux at the first wall/shield blanket;
- Extremely high heat flux in the divertor;
- Remote handling for maintenance and intervention procedures for an activated tokamak structure;
- Large radioactivity inventory;
- Unique equipment for fusion reactors, such as the fuelling and pumping equipment, heating/current drive system and diagnostics.

ITER is being supported by extensive technology R&D to validate key aspects of the design, including development and qualification of the applicable technologies and development and verification of industrial level manufacturing techniques with related quality assurance. Technology R&D for ITER is now focused on the Seven Large Projects, each devoted to one of the key aspects of the design.

Two of the projects are working towards developing superconducting magnet technology to a level that will allow the various ITER magnets to be built with confidence. The Central Solenoid (CS) Model Coil Project and the Toroidal Field (TF) Model Coil Project are intended to drive the development of the ITER full scale conductor, including the manufacturing of strand, cable, conduit and termination, and the conductor R&D in relation to AC losses, stability and joint performance. These model coil projects also integrate the supporting R&D programmes on coil manufacturing technologies, including electrical insulation, winding processes (wind, react and transfer) and quality assurance. In each case the Home Teams concerned are collaborating to produce relevant scale model coils and associated mechanical structures. The total planned production of 29 t of Nb_3Sn strand, from seven different suppliers throughout the four Parties, has been produced and qualified. For the CS model coil, the cabling and jacketing technology and winding techniques have been established and these activities have been completed. The next critical step, the heat treatment to react the superconducting alloy without degrading the mechanical properties of the Incoloy jacket, has been successfully achieved. All of the layers of the coil fabricated in JA and the US are under assembly and heat treatment.

For the TF model coil, forging and machining of the radial plates are complete. Cabling and jacketing work is also complete. Winding, reaction treatment and transfer of the reacted conductor on the radial plates have also been successfully demonstrated. All this work was performed in the EU. Dedicated coil test facilities, for the CS model coil in JA and for the TF model coil in the EU, have been completed and stand ready for the installation of the model coils for test programmes aimed at gaining broad experience in their operational flexibility and understanding of their performance margins. A 1 km jacketing has been completed in the RF which confirmed the fabrication feasibility of the full size in both length and cross-section.

Three projects focus on key in-vessel components, including development and demonstration of the necessary fabrication technologies and initial testing for performance and assembly/integration into the tokamak system.

In the Vacuum Vessel Sector Project, the main objective is to produce a full scale sector of the ITER vacuum vessel, to establish the tolerances, and to undertake initial testing of mechanical and hydraulic performance. The key technologies have been established and, in relation to manufacturing techniques, two full scale vacuum vessel segments (half-sectors) have been completed in industry, using a range of welding techniques, within the required tolerances. They were welded to each other at the Japan Atomic Energy Research Institute to simulate the field joint at the ITER site.

The Blanket Module Project is aimed at producing and testing full scale modules of primary wall elements and full scale, partial prototypes of coolant manifolds and backplate, as well as at demonstrating prototype integration in a model sector. The key technology has successfully developed, tested and qualified a range of crucial material interfaces such as Be–Cu and Cu–stainless steel, bonded using advanced techniques in the four Parties. A full scale model, without the attached components, has been completed in JA. The shield modules are attached to the backplate by mechanical means based on flexible connections to the backplate and interlocking, insulated keys between adjacent modules. These components have also been developed. Full scale modules with attached components are under fabrication in the EU and will be tested to confirm that they meet the requirements for anticipated loads, electrical insulation and remote handling together with the necessary accuracy of positioning.

The Divertor Cassette Project aims to demonstrate that a divertor can be built within tolerances and to withstand the high thermal and mechanical loads imposed on it during normal operation and during transients. To this end, a full scale prototype of a half-cassette is being built by the four Parties and subjected to high heat flux and mechanical tests in the US. The key technologies of the high heat flux components of the divertor have been demonstrated in the four Parties, using W alloy and CFC as plasma facing materials bonded to Cu cooled by high velocity water using both hypervapotron and swirl tube technologies.

The last two of the Large Projects focus on ensuring the availability of appropriate remote handling technologies which allow intervention in contaminated and activated conditions on reasonable timescales. These technologies should provide the flexibility needed for ITER to pursue its scientific and technical goals whilst satisfying stringent safety and environmental requirements. In this area, full scale tools and facilities should be developed and their testing extended over a long period of time in order not only to develop the right procedures but also to optimize their use in detail and minimize the intervention time, and to develop rescue procedures and equipment to recover equipment and components when necessary. This goal will require training of operators.

The Blanket Module Remote Handling Project is aimed at demonstrating that the ITER blanket modules can be replaced remotely. This involves proof of principle and related tests of remote handling transport scenarios, including opening and closing of the vacuum vessel, and of the use of a transport vehicle on a monorail inside the vacuum vessel for the installation and removal of blanket modules. The procedures have already been successfully demonstrated at about one fourth scale so as to reduce the risk and the cost compared with the development of full scale equipment. Work is now in progress on a full

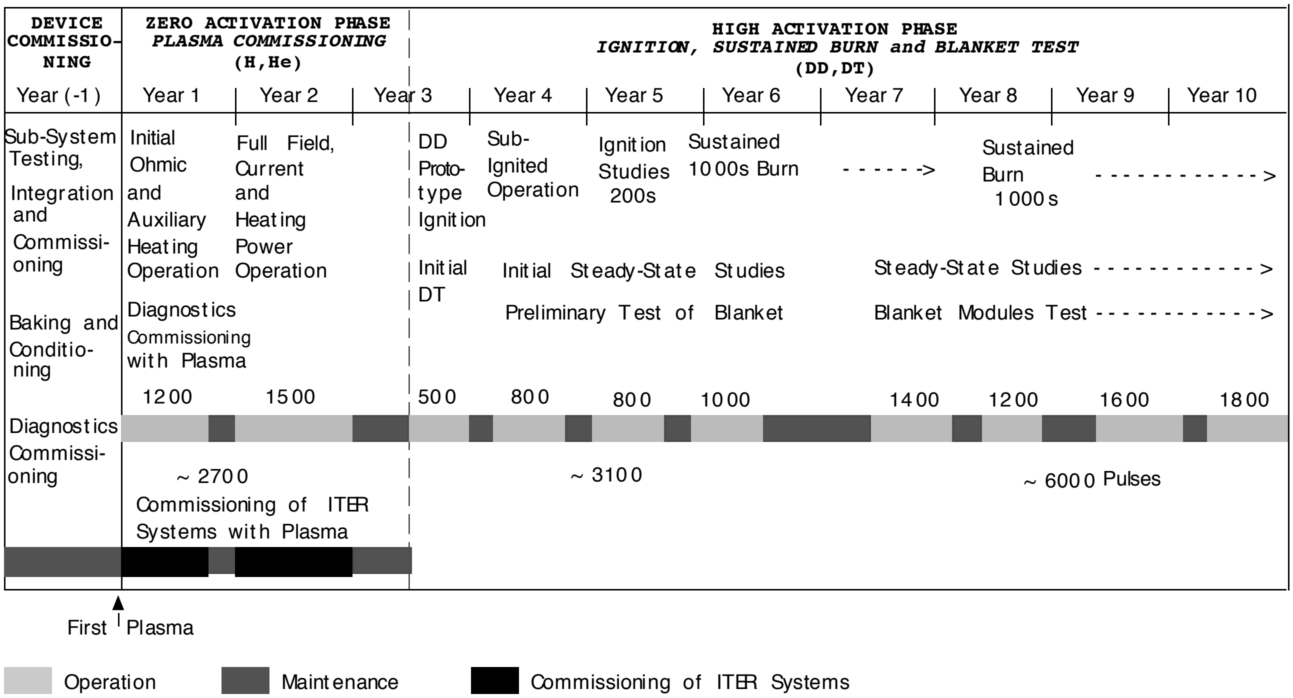

Figure 5. ITER plasma operation plan for the Basic Performance Phase.

scale demonstration. The fabrication of the full scale equipment and tools, such as a rail mounted vehicle type manipulator system, and welding, cutting and inspection tools for cooling pipes, was completed in JA. Integrated tests in a blanket test platform which simulates the full scale structure of a 180° ITER in-vessel region are providing comprehensive validation of the remote handling system so as to allow completion of the detailed design of the components and the remote handling equipment.

In the Divertor Remote Handling Project, the main objective is to demonstrate that the ITER divertor cassettes can be removed remotely from the vacuum vessel and remotely refurbished in a hot cell. This involves the design and manufacture of full scale prototype remote handling equipment and tools, and their testing in a divertor test platform (to simulate a portion of the divertor area of the tokamak) and a divertor refurbishment platform (to simulate the refurbishment facility). Construction of the necessary equipment and facilities has been completed mainly in the EU and integrated tests have been started.

In addition to these Large Projects, development of key components for fuelling, heating/current drive, tritium processing and diagnostic systems, as well as irradiation tests and safety related R&D, are in progress.

The technical output from the R&D validates the technologies and confirms the manufacturing techniques and quality assurance incorporated in the ITER design, and supports the manufacturing cost estimates for important key cost drivers. The activities are continuing to further the testing of prototype components and/or to optimize their operational use. Their performance offers insights useful for a possible future collaborative construction activity. Already much valuable and relevant experience has been gained in the management of industrial scale, cross-party ventures. The successful progress of these projects increases confidence in the possibility of jointly constructing ITER in an international project framework.

5. ITER operation

5.1. Operation plan

The construction schedule that leads up to the first hydrogen plasma operation was developed on the basis of an analysis of procurement, fabrication, installation and commissioning of all the ITER systems and foresees nine years from the first purchase order connected with the tokamak building and the superconducting cables. This period includes about one year of integrated commissioning — including vacuum pumping of a few months, discharge cleaning of a few weeks and coil excitation tests, which will ensure that all of the ITER plant is ready to operate, except for some subsystems such as the tritium plant, hot cells and radioactive material storage, which are unneeded in the first operation period with hydrogen plasma. ITER will have two phases of operation: the Basic Performance Phase and the Enhanced Performance Phase. Major operation features are summarized below.

5.1.1. First phase: Basic Performance Phase (ten years)

The operation of ITER will progress step by step from hydrogen plasma operation with low plasma current, low magnetic field, short pulse and low duty factor without fusion power to deuterium–tritium plasma operation with full plasma current, full magnetic field, long pulse and high duty factor with full fusion power. In each step, greater understanding of plasma characteristics will be obtained, which will significantly reduce uncertainties in the next step.

During the first 2.5 years, hydrogen plasma experiments will be carried out, no fusion reaction will occur and ITER in-vessel components will not be activated or contaminated by tritium. Under this non-activated condition, ITER will be commissioned with tokamak discharges at the maximum plasma current and the maximum magnetic field. A reliable plasma operation scenario to achieve the maximum plasma current will be developed. In this sense, this phase can be defined as the pre-nuclear commissioning phase. Then deuterium plasma experiments will start with a limited amount of tritium and the final ITER commissioning will be done, with particular attention to shielding performance. The fusion power and pulse length will be gradually increased. This approach ensures the safe and reliable operation of ITER. In the fifth year, the reference operation with 1.5 GW and 1000 s burn pulse is planned to be achieved. In parallel with the development of the reference operation, various operation modes, including steady state operation, will be studied. The plan of this phase is summarized in Fig. 5.

In the Basic Performance Phase, tests of the ITER tritium breeding blanket for the next phase, i.e. the Extended Performance Phase, and blankets for

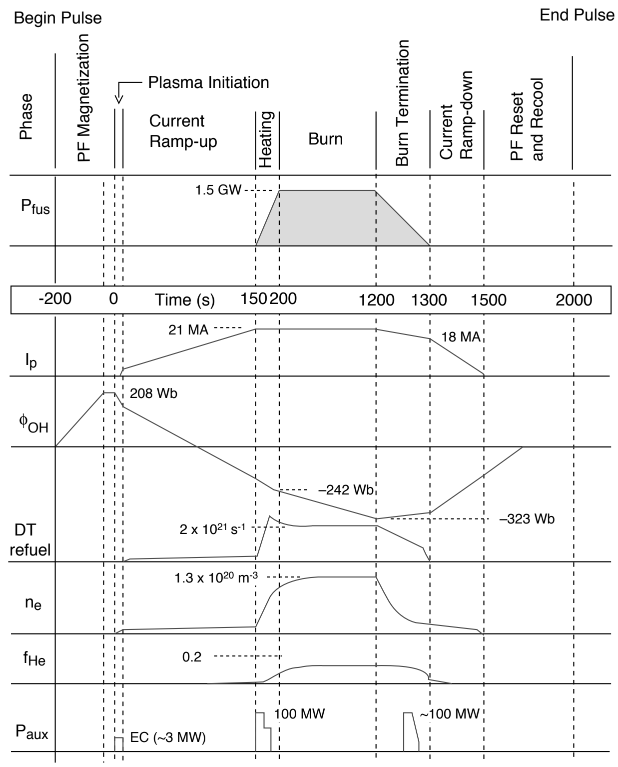

Figure 6. Poloidal field and plasma parameter waveforms for the nominal 21 MA plasma operation scenario.

DEMO will be started. At present, it is planned in the Parties' programme to test four DEMO relevant tritium breeding blanket concepts, in addition to a concept for the breeding blanket for the second phase of ITER to produce a large fraction of the tritium fuel. ITER has assigned four equatorial ports for testing tritium breeding blankets. Accumulation of average neutron fluence on the first wall is planned to be up to 0.3 MW·a/m^2. A possible external tritium supply is sufficient for this phase. The amount of net consumption of tritium increases from 0.6 kg/a to 6.5 kg/a during the 7.5 years of this DT phase.

5.1.2. Transition phase from the first phase to the second phase (two years)

The shielding blankets will be replaced by the breeding blankets because external tritium resources are not sufficient for a fluence significantly higher than that of the first phase. This process requires about two years. A tritium breeding ratio of about 0.8 would be sufficient to provide about 1 MW·a/m^2 during ten years of operation, assuming an external supply of 1.7 kg tritium per year.

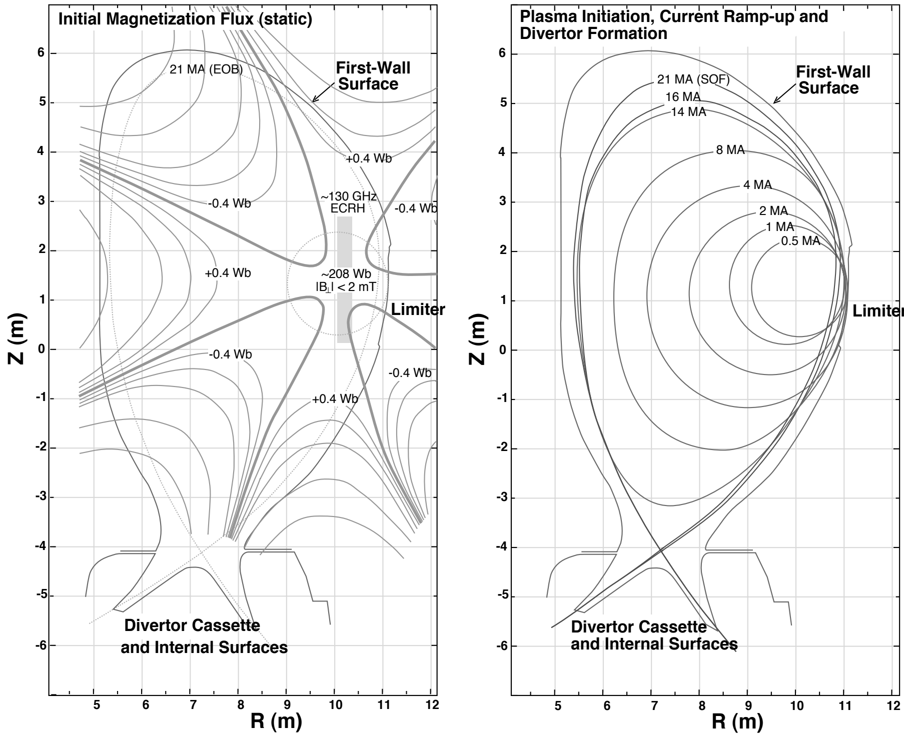

Figure 7. Plasma configuration evolution and features for the 21 MA plasma scenario: steady state and enhanced performance operation.

5.1.3. Second phase: Enhanced Performance Phase (about ten years)

A detailed operation plan for the second phase has not been developed because it will depend on the plasma performance and operating experience obtained during the first phase. However, it is foreseen that there will be less emphasis on physics studies and more emphasis on optimization of performance and reliable operation to produce high neutron fluxes and fluences, using the most promising operational modes developed during the first phase.

5.1.4. Remote experiment concept

In order to use ITER efficiently and to involve the fusion communities within the Parties, remote experimental capabilities are foreseen. In order to realize this mode of operation, the machine operation with plasma will have to be developed to a certain level of expertise and many excellent plasma and machine operation groups will have to exist to ensure that operational know-how is disseminated. The initial operation, i.e. the hydrogen phase, especially the first year, is the real ITER commissioning phase with plasma and the initial learning phase to develop machine and plasma operation and train future operational groups. Therefore, in this early phase, working at one site is fundamental and moderate operational shifts, i.e. two experimental shifts plus one night shift with limited activities such as discharge cleaning, similar to those of the present large tokamaks, may be appropriate. Remote experimental sites could be introduced after this period. Remote experiments, with the assistance of a limited number of staff at the machine site, will be performed within a range of parameters and conditions agreed to in advance or given by the on-site control room.

5.2. Plasma operation [15]

5.2.1. Inductive plasma operation

The reference plasma operation scenario for ITER is based on sawtoothing ELMy H mode operation wherein the 21 MA current flat-top required during the 1000 s burn is sustained by inductive current drive. The scenario concept is identical to that employed in the present generation of shaped cross-section divertor tokamaks. Figures 6 and 7 illustrate the scenario concept and show the plasma current–

shape–configuration evolution that plays a key role in the scenario concept.

Key features of the nominal plasma operation scenario include: (1) a 530 Wb poloidal field system flux swing; (2) inductive plasma initiation (Townsend avalanche breakdown), with EC assist in a high order multipole field null positioned near an outboard port-mounted startup/shutdown limiter; (3) minor radius and elongation expansion of the startup plasma on the limiter prior to divertor formation at $I_p \approx 15$ MA; and (4) maintenance of a precisely controlled single null divertor plasma configuration during the heating–burn–burn termination phase of the scenario. Termination of the plasma current is effected following burn termination with a controlled minor radius and elongation contraction on the limiter.

Simulations of the plasma startup and shutdown dynamics show that the required MHD stability (trajectory in the q–l_i domain) and the edge plasma power balance required to avoid a density limit disruption are satisfied with acceptable margins. These simulations also show that the plasma resistive flux (V·s) consumption during the startup and current ramp-up phase falls within the design basis guideline of $0.45\mu_0 R_0 I_p$ ($\sim$100 Wb) and that 80 Wb of poloidal field system flux swing will be available for sustaining the 21 MA plasma current during fusion burn. For the nominal estimated burn phase plasma resistive voltage, this flux swing will provide a 1300 s duration burn.

The nominal plasma operation scenario design basis is predicated upon a 'reference case' burn phase plasma with $I_p = 21$ MA, poloidal beta $\beta_P = 0.9$ and dimensionless internal inductance $l_i = 0.9$. The sizing of the poloidal field coils and their power supplies is such that plasma equilibrium control and in most cases 1000 s inductively sustained burn can be obtained for 21 MA plasmas with $0.7 < \beta_P < 1.2$ and $0.7 < l_i < 1.1$. Plasma operation with a higher plasma current, e.g. $I_p = 24$ MA, and the corresponding β_P and l_i for 1.5 GW fusion power is also feasible. The inductively sustained burn duration at 24 MA is about 500 s.

The scenario concept illustrated in Figs 6 and 7 will also support ITER operation with ohmic and auxiliary heated hydrogen and deuterium plasmas during initial plasma commissioning, and extended pulse inductively sustained driven burn operation with reduced plasma current (e.g. $\sim$6000 s burn at $\sim$1 GW with $I_p = 17$ MA and 100 MW auxiliary heating power).

The ITER design also incorporates the hardware provisions, including sufficient poloidal field system flexibility and plasma magnetic control capability, and also various options for radially localized heating and current drives, that are now anticipated to be necessary to support steady state plasma operation with plasma current sustained entirely by non-inductive current drive and bootstrap current. Assessments of the feasibility of achieving such steady state operation in ITER confirm that the major capabilities of the present design are consistent with known requirements for the plasma operation modes with weak or negative magnetic shear ($s = r/q\ \mathrm{d}q/\mathrm{d}r$) that are now obtained (mostly on a transient basis) in present tokamaks [16]. However, since the physics basis understanding of these modes and the plasma operation features required to sustain and control them on a steady state basis are still subjects of physics R&D, at the present time the degree to which steady state operation can be achieved in ITER and the details of how such operation will be controlled remain subjects of research to be undertaken in the future.

A number of considerations related to ITER plasma operation and control in a reverse shear mode (a mode with negative magnetic shear) have already emerged. First, this configuration can readily be produced in ITER by the same current and/or shape ramping combined with early auxiliary heating methods that have been successfully used to obtain enhanced performance reverse shear modes in present tokamaks. Second, the ITER poloidal field system and divertor system are compatible with the production and stabilization of a high q, low l_i, high elongation, high triangularity plasma that can be obtained by shifting the plasma outward with decreased minor radius (e.g. outward shift ≈ 0.5 m, $R_0 \approx 8.6$ m, $a \approx 2.35$ m, $I_p \approx 12$ MA, $q_{95} \approx 5$, $l_i \approx 0.4$, $\kappa_{95} \approx 2.0$ and $\delta_{95} \approx 0.45$). Third, non-inductive maintenance of a suitable reverse shear current profile and a non-inductively sustained 12 MA current with about 9.5 MA of bootstrap current is consistent with 100 MW of current drive power of a 1 MeV neutral beam system and a 170 GHz radiofrequency system suitably apportioned between on-axis and off-axis deposition. These features all support the premise that non-inductively sustained steady state plasma operation in the 1 GW power regime is possible in ITER if an appropriate energy confinement enhancement (e.g. $\sim$1.3 times ELMy H mode confinement) and a high normalized β (e.g. $\sim$3.5) are achieved.

6. Conclusions

(1) The ITER design, given in the final design report [7], is at an advanced stage of maturity and contains the necessary technical information to satisfy the purpose of the EDA Agreement and to serve as a basis for the site specific design adaptations.

(2) The ITER design incorporates the necessary provisions for the reliable operation and control of ignited and/or high Q driven burn DT plasmas with fusion powers in the 1–1.5 GW range and fusion burn durations of 1000 s. Extension of the controlled burn duration up to ∼6000 s in a reduced current driven burn operation mode appears feasible. A true steady state operation plasma with 1 GW fusion power in an operation mode with negative magnetic shear could be studied, but obtaining detailed understanding of the physics basis for the attainment and control of such plasmas remains an ongoing task for the world magnetic fusion programme.

(3) The safety assessment shows that ITER could be constructed and operated without undue risk to health and safety and without significant environmental impacts, showing the favourable safety characteristics of magnetic fusion energy production. A technical basis to satisfy the regulatory authorities of any potential host country has been developed.

(4) The programme of technology R&D, embodied in the Seven Large Projects and other supporting tasks, validates the key aspects of the ITER design, including development and qualification of the applicable technologies and development and verification of industrial techniques in manufacturing component prototypes, with related quality assurance. It also provides a substantial industrial database for cost estimates. At the same time, the projects have successfully pioneered efficient modes of international collaboration which could be possible precursors for a collaborative construction of ITER and which are a valuable asset to any possible future collaboration in fusion development.

(5) Operation of ITER is planned to progress step by step from hydrogen plasma operation with low plasma current, low magnetic field, short pulse and low duty factor without fusion power, to the full deuterium–tritium burn operation in the fifth year. In each step, the characteristics of plasma will be better understood and uncertainties in the next step will be significantly reduced. This approach enhances the safety and reliability of the ITER operation. Remote experimental capabilities are foreseen in order to use ITER efficiently and to involve the fusion communities within the Parties.

(6) The ITER project is an unprecedented model of international co-operation in science and technology. All participants benefit not only from the technical results but also from the experience of different approaches to project organization and management. The ITER project has proven to be an effective and efficient vehicle for the fusion engineering needed to realize any concept of commercial magnetic fusion reactor. Bringing ITER to full realization through joint construction and operation will continue this process.

(7) A three year extension of the EDA is now foreseen. During this period, the Parties need to resolve the key issues of siting and regulatory clearance, cost sharing and procurement arrangements, and establishing the legal framework and organization appropriate to a global venture of ITER's size and technical demands. The following technical work was originally planned to help reinforce the technical basis for a positive construction decision:

- Adapting the design to the specific characteristics of possible construction sites;
- Supporting preparations for formal applications for licences to build and operate ITER;
- Extending prototype testing to provide data on operational margins;
- Finalizing the design and procurement specifications and related documentation for the ITER systems, taking into account industrial capabilities;

  Consolidating the scientific basis of ITER operations.

However, owing to financial constraints, this extension will now be used to develop reduced cost options of ITER, aiming at a target of approximately half the direct capital cost of the present design, with reduced detailed technical objectives which would still satisfy the overall programme objectives of ITER. The existing EDA technical output of design choices, generic technologies and major R&D results is generally directly applicable to a reduced cost ITER. Therefore, it will be possible to develop a reduced cost ITER in a relatively short period and the detailed design report will be available by July 2000, when the joint assessment of ITER construction and operation by the Parties is planned.

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