ITER: Fusion Research at the Dawn of a New Era

R. Aymar 1), Y. Shimomura 2), M. Huguet 2), V. Chuyanov 1)
and the ITER International and Participant Teams

1) ITER JWS, Boltzmannstrasse 2, 85748 Garching, Germany
2) ITER JWS, 801-1 Mukouyama, Naka-machi, Naka-gun, Ibaraki-ken, 311-01, Japan

e-mail contact of main author: aymarr@itereu.de

Abstract: Given the expected success of on-going negotiations on the Joint Implementing Agreement for ITER construction and operation, a new era is opening for experimentation with reactor-relevant physics integrated with key reactor technologies in a licensed nuclear environment. The ITER design, cost estimate and safety analysis are supported by a large body of validating physics and technology R&D. The main features of the design, and analysis of its performance, give confidence that it will fulfil its technical objectives and demonstrate the environmental attractiveness of fusion. This paper gives illustrative confirmation of these expectations and an update on the technical preparations for construction, as well as the status of negotiations.

1. Introduction

The Engineering Design Activities (EDA) of ITER in July 2001 provided a mature design, cost estimate and safety analysis, that is supported by a body of validating physics and technology R&D [1]. With it, the ITER Parties and the world fusion development programme are technically ready to proceed to the construction of a next step tokamak device that bridges the strategic gap between the present generation of large tokamak experiments and a first demonstration fusion power reactor.

The success of the ITER EDA collaboration supported the Parties' declared policy interests to pursue the development of fusion through international collaboration. Therefore, following the EDA, negotiations between the Governments of the three current Parties (Euratom, Japan and the Russian Federation), and Canada who first proposed a site for ITER, were launched and are progressing towards the Joint Implementation Agreement for ITER construction and operation. This is expected to be ratified during 2003. The Agreement, inter-alia, should define the ITER Legal Entity (ILE) as an international body governed by international law, set up its organisation, identify the Director-General, select the ITER construction site, and determine the technical and financial contributions of each Party.

Confidently assuming their positive outcome, these negotiations show that fusion research no longer relies on its scientific interest alone, but now appeals, outside its own community, to those who consider it attractive as a possible large scale contributor to the energy mix needed for the second half of this century, even to the extent of putting emphasis [2] on a "fast track" to the first electrical fusion power source.

The main features of the ITER design, and its detailed performance analysis, provide the basis for being confident in the fulfillment of its technical objectives and in the demonstration of the environmental attractiveness of fusion. This paper gives illustrative confirmation of these expectations and an update on the progress of technical preparations for ITER construction, as well as the status of the above negotiations.

2. Characteristics of the New Era

With the impending construction and operation of ITER, fusion research is at the dawn of a new era.

- ITER will be the first machine to provide a plasma dominated by alpha particle heating, allowing a thorough and reactor-relevant investigation of the combination of external and self-heating processes and their effect on burning plasma physics over long timescales;
- ITER will be the first machine combining this realistic plasma physics with the key reactor technologies of high heat flux components and their remote maintenance, as well as the long-term reliable superconducting magnet technology necessary for power production.
- ITER will also be the first to experience a realistic fusion neutron radiation spectrum and to be acceptable, also to nuclear regulators, thereby permitting ITER to obtain an operating license, components will have to be hardened and materials qualified at a level which ensures adequate resistance to degradation over the plant life and their ability to be as far as possible recyclable later.

Thus ITER will focus development on the understanding of physics in combination with reactor-relevant technologies, even though the latter are to some extent preliminary.

The high costs of ITER construction and operation place a burden on all the participating fusion programmes. ITER will thus experience the greatest pressure to focus its objectives on the essentials and to operate as efficiently as possible. Such a burden also places responsibilities on the participants to make this international fusion facility a success. They will need to ensure their industry contributes high quality components during construction of the device. They also need to obtain as much participation as possible from their national fusion laboratories, exploiting the most modern communications technology to allow remote plasma and test blanket diagnosis, and participation in machine operation and its planning. Thus fusion laboratories will have to:

- build on their existing expertise to support the preparation of ITER operation by focusing on pertinent physics issues and on supporting necessary tools such as heating and diagnostics systems that will be deployed on ITER, continuing to examine alternative confinement schemes for their potentially superior promise, recognising that they will anyway benefit from the ITER achievements;
- adapt their programmes to embrace an ever-increasing emphasis on nuclear engineering and technology, i.e. low activation, more radiation-resistant, material development, and tritium breeding blanket development.

3. Design Overview

The major parameters of ITER are shown in Table I. This device satisfies the objectives for inductive operation, and is optimised for such performance, but is equipped for steady state operation.

A cutaway of the ITER tokamak is shown in Figure 1. The magnet system consists of 18 Nb_3Sn toroidal field (TF) coils, a central solenoid (CS) of six modules which can be powered separately, six NbTi poloidal field (PF) coils, and 18 NbTi saddle-shaped correction coils outside the TF coils to accommodate field errors due to manufacturing inaccuracies or to misalignments during assembly, as well as to control resistive wall mode instabilities. The TF coils form a wedged vault over their straight section, and a toroidal shell-like structure reacts overturning moments and circumferential torques.

The reaction chamber consists of a 9-sector vacuum vessel supporting exchangeable modular in-vessel components: 421 blanket modules with single-curvature faceted beryllium-coated separate first wall, remotely attachable to the vessel through 3 cm diameter access holes in the first wall, and a 54-cassette single-null divertor with carbon targets and tungsten high heat flux components. Six of the eighteen equatorial port plugs are used for heating antennae and neutral beam ducts, three are used for DEMO-relevant test blankets, two for plasma limiters, and the remainder for plasma diagnostics. The limiter and two diagnostic ports are also used for remote replacement of the blanket modules. Divertor ports accommodate 8 torus cryopumps, diagnostics, glow discharge cleaning system, pellet and gas injection, and an in-vessel viewing system. Three divertor ports are also used for the remote replacement of the divertor cassettes. The upper ports are used for diagnostics and electron cyclotron antennae to control plasma instabilities.

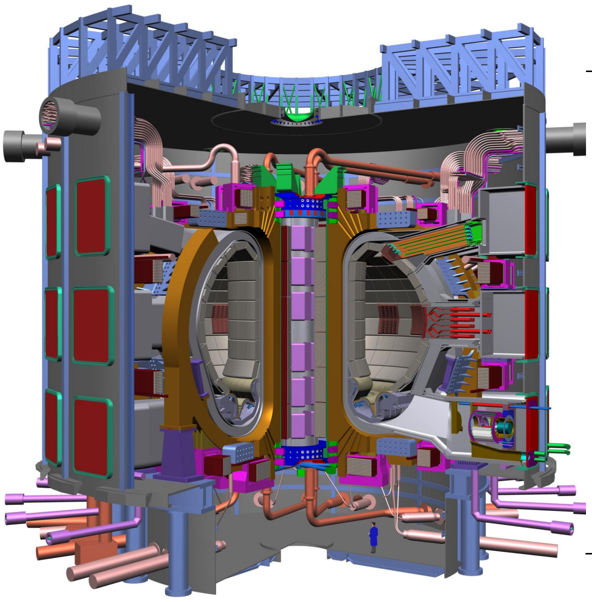

FIG. 1. ITER Tokamak Cutaway

TABLE 1

MAJOR PLASMA PARAMETERS AND DIMENSIONS

Total Fusion Power (MW)	500 *(700)*
Q — fusion power/additional heating power	≥ 10
Average 14MeV neutron wall loading (MW/m^2)	0.57 *(0.8)*
Plasma inductive burn time (s)	≥ 400
Plasma major radius (R, m)	6.2
Plasma minor radius (a, m)	2.0
Plasma current (I_p, MA)	15 *(17* [a]*)*
Vertical elongation @95% flux surface (κ_{95})	1.70
Triangularity @95% flux surface (δ_{95})	0.33
Safety factor @95% flux surface (q_{95})	3.0
Toroidal field @6.2 m radius (B_T, T)	5.3
Plasma volume (m^3)	837
Plasma surface (m^2)	678
Installed auxiliary heating/current drive power (MW)	73 [b]

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

(b) Up to 110MW may be installed in later phases.

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

ITER will operated in three phases:

- *H Phase* -to allow complete commissioning in a non-nuclear environment, where the peak heat flux onto the divertor target, electromagnetic loads, and heat loads due to runaway electrons, will be similar to those of the DT phase.
- *D Phase* - preliminary shots also with a small amount of tritium to allow integrated nuclear commissioning and a check of shielding performance.
- *DT Phase* - in which the full inductive operational conditions are reached, and non-inductive, steady state operation is developed, and in which DEMO-reactor-relevant blanket modules will also be tested. After about 10 years of operation, a second part of this phase will improve and optimise the overall performance and test components and materials with a higher neutron fluence and availability. This phase will also allow the introduction of new more reactor-relevant designs and materials if that is appropriate.

ITER is able to operate in a range of modes, which give flexibility to identify the optimum performance. In the initially planned operation, they include inductively driven operation modes with and without additional heating during current ramp up (500 and 400MW respectively) and at higher current (700 MW I=17 MA, Q = 20, burn time = 100s), non-inductive operation with weak or strong negative shear or with positive shear, and "hybrid" operation using current drive to stretch the burn (burn time ~ 1000s). Thus the basic hardware of ITER is very flexible and able to respond to new schemes which might result from a complementary but highly coupled experimental programme on the remaining supporting machines.

During the preparation of the detailed specifications for long lead-time items, now underway, the consequent detailed review of the ITER design, as well as the results of on-going manufacturing and design analysis, have highlighted detailed improvements which can be used to reduce costs or simplify assembly or manufacture. These include the reduction of the number of lower ports from 18 to 9, avoiding field welds on the port centreline, and an improved support system for the main vessel, easing assembly.

In addition to these design modifications, new systems and procedures are being prepared to reinforce project management, in particular quality management. The control of documentation (including models and drawings) is key for this. Given its highly distributed world-wide manufacture, documents must be viewable off-site from low power, platform-independent, interfaces at low or nil extra software cost. The system must support careful control of change. As part of this investigation, a pilot project is being carried out using the CERN EDMS system [3] which is being applied to the on-going manufacturing and assembly of LHC. This system is also being investigated for drawing management, along with alternative complementary systems.

4. Will ITER meet its objectives?

The ITER Final Design Report [1] of July 2001, is a detailed, complete, and fully integrated engineering design of ITER, elaborated to the extent necessary to allow a realistic assessment of its feasibility, performance and cost at a generic site. This section shows how the resulting design gives confidence that ITER will meet its objectives, using illustrations from the areas of physics performance, design feasibility, and safety assessment.

ITER's objective is to demonstrate the scientific and technical feasibility of fusion energy for peaceful purposes. This means that ITER would demonstrate moderate power amplification (Q = fusion power/input power) and extended burn of deuterium-tritium plasmas, with steady-state as an ultimate goal, demonstrate technologies essential to a reactor in a system integrating the appropriate physics and technology, and perform testing of high-heat-flux and nuclear components. It should achieve $Q \geq 10$ during an inductively driven burn of ≥ 300 s, and aim at demonstrating steady state operation with $Q \geq 5$, with an average 14 MeV neutron wall load ≥ 0.5 MW/m^2 and average lifetime fluence of ≥ 0.3 MWa/m^2. This satisfies the goal of a single device answering, in an integrated way, all feasibility issues needed to define a demonstration fusion power plant (DEMO), except for low-activation materials which will be essential for wide deployment of fusion power, and the demonstration of sufficient endurance of in-vessel materials exposed to 14 MeV neutrons.

4.1. Physics Performance

The physics basis for ELM-y H mode operation, the reference scenario for ITER, has been developed further in the last couple of years by experiments worldwide [4]. Both high triangularity and high-field side pellet injection have been successful means to help operation with high density ($n_e \sim n_{Greenwald}$) and good confinement quality. A significantly higher number of data points at high density have been so generated extrapolating to $Q > 10$ in ITER. In addition, ITER operation with this high density should lead to quite acceptable divertor target average heat loads.

While the projection of heat loads with type-I ELMs is still associated with large uncertainties, recent analyses, with realistic power flux time behaviour, have shown that the ITER design can accommodate a target heat load of about 1 MJ/m^2 [5] or about 6 MJ per ELM corresponding to about 6 % of the pedestal energy. While some extrapolations indicate this energy fraction to be somewhat larger [6] there have been signs that pellet injection could reduce it to more benign levels [7]. A supplementary increase in the divertor target plate inclination would also help in increasing the acceptable heat load, albeit with a reduction of the plasma shaping flexibility. Additionally, experiments are gradually expanding the operating range, particularly through higher shaping, with type-II ELMs which are very promising and would certainly help to extend the divertor lifetime even if a reduction of the plasma current would probably be necessary so as to operate at a larger edge safety factor [8].

Significant progress has also been achieved worldwide concerning advanced operation regimes [4], and ITER, with its flexible and diverse heating and current drive systems and correction coils to stabilise resistive wall modes, appears to be able to reap the benefit of this progress to meet its operational objectives. Plasmas relevant for steady state operation characterized by high plasma performance, high bootstrap current fraction and non-inductive current drive were demonstrated in JT–60U, JET, ASDEX, and DIII-D. Real-time control was tested on the electron temperature profile and neutron rate in JET [9]. Stationary high confinement regimes have been found in high β_N scenarios in ASDEX Upgrade (reaching $\beta_N = 3.5$ in steady state) [10], and JET at $\beta_N \sim 2.5$ in spite of Frequently Interrupted (FIR) (3,2) Neoclassical Tearing Modes (NTMs) and a relatively low Greenwald density.

Increases in triangularity have helped to improve the edge pressure limit as well as the stability of NTMs. However, generally peaked density profiles obtained with central fuelling (such as pellets) appear to reduce the β_N onset level of NTMs. Several experiments have now demonstrated NTM control with ECCD under active feedback; moreover sawtooth control in JET has been shown to significantly affect the onset β_N value.

The last couple of years have also seen important developments in predicting ITER performance with theory-based models [11]. While significant scatter in the required value of the pedestal temperature still exists between these models, $Q = 10$ should be achievable in ITER with a pedestal temperature in the range of 2-4 keV. The analysis of the pedestal database suggests that such temperatures should be achievable also at relatively high density.

4.2. Design Feasibility

Operational performance of the ITER machine has been subject to a large number of analyses to assess the combined effects of disruptions, vertical displacement events, and seismic events. Where necessary in particular the vacuum vessel has been strengthened to tolerate safely a large number of such events. The ITER machine is designed to be sufficiently versatile and robust to allow the wide range of scenarios mentioned previously to be thoroughly tested, and to overcome any shortcomings in basic hardware performance.

ITER's design feasibility has been demonstrated by relying wherever possible on conventional technology, and carrying out a comprehensive R&D programme underpinning any new concepts introduced [12]. Reactor-relevant technological solutions are adopted on ITER as far as possible, to provide the most realistic testing conditions. This is typified by the choice of superconducting magnets, the use of modular replaceable in-vessel components, and the development of components to handle high heat fluxes.

ITER will be the first device to use Nb_3Sn superconductors on a large scale. It adopts a novel design of superconducting cable in a conduit of steel. The performance of this superconductor has been thoroughly checked on the Central Solenoid [13] and Toroidal Field Model Coil [14] R&D projects, and those on the CS model coil inserts [15], which have set new performance records. Nominal operational performance has been demonstrated for ITER use, in particular low AC losses at ≥ 0.5 Hz and better than expected transient field stability. Nevertheless, detailed analysis of experimental test conditions show that, due to large mechanical loads, the conductor design margins in average current density are smaller than expected. Investigations are therefore underway to determine the best way in which design and/or strand quality can provide the extra margins originally envisaged. Improvements within the same conductor volume can be achieved by changing the design of the conductor e.g. by using more strand, higher current density strand, reducing the void fraction, or changing the conduit from steel to titanium. In conclusion, the detailed performance of cable-in-conduit conductor has been satisfactorily quantified for the new coil manufacturing technology (wind-react-transfer) using Nb_3Sn, thereby qualifying it for use in ITER and beyond.

Although the PF coils use the rather more conventional NbTi technology, the lower outer coils, which are essentially trapped, also have redundant pancakes which can be isolated, and the remaining turns have margin in current to compensate for their loss.

Plasma-facing components are necessarily experimental due to the need to explore a wide variety of physics regimes in ITER to find the optimum performance and to identify any promising development paths for future devices. It is therefore essential not only that they not be relied upon for machine operating safety, but that they be replaceable as simply as possible. To this end ITER uses a modular approach to in-vessel components, with the use of blanket modules and divertor cassettes that can be removed through the ports, and through the use of port plugs, all of which can be removed (within unshielded sealed casks) to the hot cell for repair and refurbishment in order to minimise waste, thereby minimising cost and environmental impact. This procedure has been demonstrated for both blanket and divertor at full scale [16 , 17].

The choice of plasma-facing materials for high heat flux component is a compromise between material durability, tolerance to peak loading, trapping of tritium, and plasma poisoning. Different materials have been chosen for different locations, depending on the relative importance of these effects: beryllium for the first wall, carbon for the divertor target plates, and tungsten for the divertor throat. Whether this workable solution is optimal will be determined from the operational results in ITER. A number of technologies for joining plasma-facing materials to cooling channels and to structural support have been developed, and prototypes of all components have been tested in high heat flux facilities to model the operational loads. The ability to withstand the required combination of high heat fluxes and the number of cycles has been demonstrated.

Plasma-facing materials will create tritiated, radioactive, chemically reactive or toxic dust which may have to be periodically removed to limit the potential for release or oxidation of this material in the event of a coolant or vessel leak.

Tritium co-deposition with eroded carbon is expected to be the dominant tritium retention mechanism in ITER even if the use of carbon is limited to the divertor strike plates. The quantification of the co-deposition rate in ITER [18] is still subject to large uncertainties. On-going experiments in specific facilities [19] have already shown that the low values of codeposition are obtained by maintaining the wall temperature above 100°C, and when the ratio of atomic hydrogen isotopes to hydrocarbons is large, a situation that will prevail in the ITER divertor pumping duct. Even if some uncertainties remain - such that ITER needs to be operated to quantify them - the initial phase of operation with H- and D- plasmas will allow problem areas to be explored. If necessary, some remedial action (e.g., tritium removal techniques, or replacing carbon with W) can be implemented to offset underestimated or unforeseen phenomena. Plasma edge and wall diagnostics in ITER, and adequate models sufficiently benchmarked against experiments, are an essential element to implement this strategy.

Regarding extrapolation of these technologies beyond ITER, the solutions adopted for ITER, for the divertor heat load handling, are at the present limits of proven technology with presently well-characterised materials. If the divertor heat loads cannot be kept in the same range as in ITER, a power reactor may require the development of more advanced technologies. Some degree of testing of these technologies may be possible later on ITER. In the case of the blanket, different structural materials will be necessary which exhibit the endurance and high temperature performance needed. The test blanket module programme on ITER in combination with materials characterisation on a dedicated fusion materials test facility should provide confidence in the necessary performance.

4.3. Environmental Impact

One of the main ITER goals is to demonstrate the safety and environmental potential of fusion power from its essential characteristics: low fuel inventory, ease of burn termination, self-limiting power level, low power and energy densities, low energy inventories, large heat transfer surfaces and heat sinks, and the fact that confinement barriers exist and must anyway be leak-tight for successful operation. Comprehensive and conservative design assessments [20] have thus been undertaken to confirm the safety and environmental acceptability of ITER.

These studies show that potential additional doses to members of the public (i.e. the most exposed individual) during normal operation, for a generic site, are less than 1% of the natural background level. For off-normal events, the most severe incidents from the range of those possible were analysed. In all of these, additional doses (to the most exposed individual) would be comparable to the average annual natural background exposure for a generic site (i.e. 2 mSv/a). Furthermore the ultimate safety margins of ITER have been examined by analysis of hypothetical events, which arbitrarily assume more and more technically unlinked failures. Even under the worst imaginable combination of events, the design and operation of the facility protects the public to such a degree so that there is no technical justification for dependence on public evacuation as a backup.

Two features of the ITER design illustrate the choices that have been made throughout the design to ensure it maintains the promise of the low environmental impact of fusion power. In the event of an in-vessel loss of coolant, which subsequently strikes hot surfaces, there is the potential for significant steam evolution inside the main vessel causing it to become pressurised. To avoid vessel overdesign, and to avoid having to prove the integrity of the vessel, and any port barriers such as diagnostic windows, at high pressure, a blowdown volume is provided. This is essentially a large tank part-filled with water in which, beyond a certain pressure limit, excess steam evolved in the vessel can be released to condense and lower the pressure. With such a system a maximum overpressure in the vessel at a reasonable level of 2 bar can be safely predicted, and diagnostic windows and seals designed accordingly. The probability of tritium reaching the environment is thus reduced to the problem of removing entrained tritium from the vessel in the blowdown tank.

Another feature is the introduction of a port cell surrounding each main vessel port. This relatively large volume is maintained by air conditioning at a pressure below that in the surrounding plant, ensuring that air leakage is always inwards. This feature offers additional protection in the plant against coolant leaks at the port connections during operation, and maintains a protective atmosphere in the region during the sealing and resealing of the port flange during maintenance cask docking. It also has allowed a simplification in the design of the cryostat at its connection to the vessel ports, extending the secondary confinement boundary function of the cryostat. The vessel and cryostat are inherent confinement barriers which prevent releases, but are anyway supplemented by air/water detritiation and filtration systems to treat any releases that do occur.

5. Participation Opportunities on ITER

The ITER Project Team, in constructing ITER, will provide the basic functionality required, but the experimental programme to be carried out will rely heavily on the tools developed by the scientists and engineers of the ITER Parties from their home laboratory base. This is particularly true with regard to the provision of heating and diagnostic equipment to drive and monitor the machine, and to contribute DEMO-relevant blanket modules for testing from the start of operation.

Most of the heating systems to be deployed on ITER have not yet completely demonstrated the full performance necessary on ITER, and this R&D should be completed at a high priority. The basic installation of heating systems on ITER involves the use of two neutral beam (NB) injectors (33 MW), one ion cyclotron (IC) antenna (20 MW), and one electron cyclotron (EC) launcher (20 MW) for a total of 73 MW of delivered power. EC power can also be delivered through the upper ports (7 MW/port) to help control neo-classical tearing modes. If lower hybrid (LH) launchers are added, up to 80 MW of RF heating may be deployed, and up to 50 MW of NB (3 injectors). The radio frequency systems are designed so that any frequency combination of 20 MW equatorial port units can be used.

With regard to diagnostics, the planned deployment of about forty different systems around the machine is envisaged [21]. These have been also prioritised into those necessary for machine protection or basic control, those needed for advanced control, and those desirable for physics studies. They have been further segregated into a startup set, and those that can be added later for DT operation. Most of these diagnostics presently exist at the conceptual level. However in many cases new concepts are needed to achieve the required relevance or accuracy (e.g. combined alpha particle density and energy spectrum, q profile) or to implement them in a nuclear environment. A significant effort is needed in participating laboratories to carry out the necessary R&D, to finalise their detailed design and monitor procurement and implementation, as well as to provide staff to operate the diagnostics.

The DEMO-relevant tritium-breeding blanket modules include small test articles and modules that can be inserted in three equatorial ports. ITER will not be able to provide data to qualify low-activation long endurance materials for DEMO. That data must be gleaned from the concurrent operation of a specific materials test facility. However, the ability to operate in a true fusion environment will allow the proof of principle of the modules' designs, enabling the benchmarking of fission reactor results on irradiation, the confirmation of neutronic and breeding calculations, tritium control and extraction experiments, and thermohydraulic analysis to confirm the ability to produce high grade coolant to drive electricity production. The concepts to be studied are water-cooled and helium-cooled solid breeder and lithium-lead, and self-cooled liquid lithium. Testing of breeding blanket modules must not interfere with ITER availability, decrease ITER reliability, or compromise safety. For this purpose the Test Blanket Working Group exists to oversee and coordinate the design efforts of the ITER Parties. Eventually such a group will represent the machine users.

In addition to the above, fusion researchers will be able to participate in remote operation, diagnosis, and data acquisition for ITER from many locations worldwide.

From the above it can be concluded that existing fusion laboratories will have many opportunities to contribute to the ITER construction and operation both in physics and technology and the success of ITER cannot be assured without their specialised expertise and continuous support for heating systems, diagnostics, and test blankets.

6. Evolution of Project Organisation and Status of Negotiations

Since negotiations began in mid-2001, the Negotiators have met 5 times to discuss progress in drafting the Joint Implementation Agreement (JIA) and related instruments, which also specify the outcome of their tasks to select the ITER construction site, agree who will provide and pay for the various ITER components/systems, and identify the Director-General for the ITER Legal Entity (ILE) and the organisation of its work. Pending a site decision, the various alternative arrangements have been considered, and the many common factors are being developed, documented, and agreed in detail.

The ITER Legal Entity (ILE) will be an international organisation under international law. It will be the owner of the license for construction and operation and will be responsible for enforcing its terms. The work of the ILE will be overseen by an ITER Council and subsidiary committees, and managed by its chief executive, the ITER Director General (DG). All staff of the ITER Project Team will report to the DG. The DG and his staff will be responsible for the fulfilment of ITER's objectives, the quality of the construction, and for the proper working of the complete ITER plant. Therefore they will have technical control of all procurements, provide their technical specifications, decide upon necessary design and schedule changes, and accept the final product.

The ITER Project Team will exist at several different locations. On the ITER construction site, a Central Team will take overall responsibility for project integration and project control, as well as design specification and configuration control. Field Teams, based on the territory of each Party, follow up the technical progress (including quality control) of the procurement contracts awarded in that Party's territory.

Each ITER Party will appoint a Domestic Agency (DA) for its contribution “in kind” to the project needs. The DA will primarily take care of the financial and legal arrangements and the actual placing of the relevant contracts to its industry. It may also take such care of procurements placed with that Party and paid from joint funds.

Before embarking on the new era in fusion research, a few remaining steps are needed. Four potential ITER sites are now being assessed against the requirements, but the decision between them will probably not be made on purely technical grounds and an agreement between governments may take time. The JIA is in the final stages of its evolution. The aim is now to conclude the negotiations in the first half of 2003, with the initialing of the Agreement by the Negotiators, prior to ratification by the Parties by around the end of 2003.

Concurrent with the above negotiations, and underpinning them technically, Coordinated Technical Activities (CTA) of the International and the Participant Teams have been preparing for an efficient start of ITER construction. They have included:

- study of design adaptations to potential sites and their regulatory environment;
- preparation of licensing applications by closer (formal in two cases) dialogue with potential regulators, which requires a formal review of the design ensuring its quality and its completeness;
- exploitation of physics R&D to take advantage of the latest experimental results, which continue to be analysed by world experts in the new framework of the International Tokamak Physics Activities (ITPA), replacing the EDA Physics Expert Groups;
- establishment of technical specifications for procurements which need to be launched early, i.e. mainly magnets, vacuum vessel, and buildings.

These technical activities, and new ones, will be continued in the frame of "Transitional Arrangements" (ITA), which will start in January 2003, leading up to ITER construction. The scope of the ITA includes organisational preparations that will enable the ITER Legal Entity to operate effectively without delay following the signature/ratification of the JIA, such as interim structures, procedures and staff assignments, and joint technical preparations directed at maintaining the coherence and integrity of the ITER design and at continuing to prepare for an efficient start of ITER construction. The structure for the ITA will be progressively in many respects prototypical of the joint implementation, including an ITER Preparatory Committee, forerunner of the ITER Council, and a Nominee Director General, responsible for design integrity and the work programme.

7. Conclusions

The construction of the ITER device heralds the dawn of a new era in the development of fusion as an energy source. It will be the first machine to work in plasmas dominated by self-heating by alpha particles, it will be the first to integrate both the physics and key technologies necessary for power generation, and it will be the first to bear the closest scrutiny by nuclear licensing authorities and have the potential and responsibility to demonstrate the environmental promise of fusion power. Such a fundamentally different machine to its predecessors places new strains on and introduces new challenges for the fusion research community.

The type of studies carried out on ITER will encompass all the plasma physics studies of today, and must permit the best physics ideas coming from within the programme to be tested out at ITER scale. It must also allow the optimisation of the nuclear technologies required for efficient and reliable power generation in the machines to follow ITER.

This paper has demonstrated the flexibility of the ITER design to react to lessons learned from its experimental nature, and has illustrated the features of the design that give confidence that it will be a technical success. There is an undisputed need for a sustained burning plasma experiment 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.

At this stage the remaining steps to its realisation are in the hands of the negotiators, who are close to agreement on the Joint Implementation Agreement and on the details of an efficient organisation to operate the ITER Legal Entity, and with their governments as they attempt to find a mutually beneficial compromise over siting and cost sharing. There are presently good signs that one can be confident on a positive outcome of these negotiations, leading to a report on ITER start of operations at the 25th IAEA Fusion Energy Conference!

8. References

[1] Technical Basis for the ITER Final Design, EDA Documentation Series No 22, IAEA, Vienna 2001

[2] “Fast Track” expert meeting, November 2001, EUR (02) CCE-FU 13.2.1.1

[3] CERN EDMS system: http://edmsoraweb.cern.ch:8001/cedar/cedarw.site_home

[4] SHIMADA, M., MUKHOVATOV, V., FEDERICI, G., GRIBOV, Y., KUKUSHKIN, A., MURAKAMI, Y., POLEVOI, A., PUSTOVITOV, V., SUGIHARA, M., Performance of ITER as a Burning Plasma Experiment, this conference

[5] FEDERICI, G., WUERZ, H., JANESCHITZ, G., TIVEY, R., Erosion of plasma-facing components on ITER in ITER, 6th International Symposium on Fusion Nuclear Technology, San Diego, 2002, to appeaer in Fus, Eng. Des

[6] LOARTE, A., SAIBENE, G., SARTORI, R., BECOULET, M., HORTON, L., EICH, T., HERMANN, A., MATTHEWS, G., JACHMICH, S., ASAKURA, N., CHANKIN, A., LEONARD, A., PORTER, G., FEDERICI, G., JANESCHITZ, G., SHIMADA, M., SUGIHARA, M., ELM Energy and Particle Losses in Present Tokamaks and their Extrapolation to Burning Plasma Experiments, EPS 2001

[7] POLEVOI, A.R., MEDVEDEV, S.Y., PUSTOVITOV, V.D., MUKHOVATOV, V.S., SHIMADA, M, IVANOV, A.A., POSHEKHONOV, Y.Y., CHU, M.S., Possibility of Q>5 stable steady-state operation in ITER with moderate βn and H, factor, this conference

[8] LOARTE, A, Effects of divertor geometry on tokamak plasmas, Plasma Physics and Controlled Fusion 43 (6), R183-R224, June 2001

[9] MAZON, D., LITAUDON, X., MOREAU, D., RIVA, M., TRESSET, G., BARANOV, Y., BECOULET, A., CHAREAU, J.M., CRISANTI, F., DUX, R.,

FELTON, R., JOFFRIN, E. Real-time control of internal transport barriers in JET, Plasma Physics and Controlled Fusion 44 (7), 1087-1104, July 2002

[10] [11] MUKHOVATOV, V., SHIMOMURA, Y., POLEVOI, A., SHIMADA, M., SUGIHARA, M,. BATEMAN, G., CORDEY, J. G. KARDAUN, O., PEREVERZEV, G., VOITSEKHOVICH, I., WEILAND,J., ZOLOTUKHIN, O., CHUDNOVSKIJ, A., KRITZ, A.H., KUKUSHKIN, A., ONJUN, T., PANKIN, A., PERKINS, F. W., Comparison of ITER Performance Predicted by Semi-Empirical and Theory-Based Transport Models, this conference

[12] SHIMOMURA, Y. (ed.), ITER Technology R&D, Fusion Engineering and Design 55 (2001) Issues 2-3

[13] TSUJI, H. et al., Progress of the ITER Central Solenoid Model Coil Programme, 18thIAEA Fusion Energy Conference, Sorrento, October 2000, paper ITER/4

[14] FILLUNGER, H., HURD, F., MAIX, R.K., SALPIETRO. E., et al. Assembly in the Test Facility, Acceptance and First Test Results of the ITER TF Model Coil, 17 International Conference on Magnet Technology, Geneva, September 2001

[15] HUGUET, M., AND THE ITER INTERNATIONAL TEAM AND PARTICIPANT TEAMS, The ITER Magnets: Preparation For Full Size Construction based on the Results of the Model Coil Programme, this conference

[16] KAKUDATE, S., OKA, K., YOSHIMI, T., HIYAMA, M., TAGUCHI, K., SHIBANUMA, K., KOIZUMI, K., MATSUMOTO, Y., HONDA, T., HAANGE, R., Status of the Extended Performance Tests for Blanket Remote Maintenance in ITER L6 Project, 18th IAEA Fusion Energy Conference, Sorrento, October 2000, paper ITER/P23

[17] MAISONNIER, D., CERDAN, G., CHIOCCHIO, S.,. DAMIANI, C., FRICONNEAU, J.-P., HAANGE, R., IRVING, M., MARTIN, E., PALMER, J., POGGIANTI, A., SIUKO, M., TADA, E., TAKEDA, N., TESINI, A., TIVEY, R., TURNER, A., ITER-FEAT Divertor Remote Maintenance, 18th IAEA Fusion Energy Conference, Sorrento, October 2000, paper ITER/P24

[18] FEDERICI, G., ANDREW, P., BARABASCHI, P., BROOKS, J.N., DOERNER, R., GEIER, A., HERRMANN, A., JANESCHITZ, G., KRIEGER, K., KUKUSHKIN, A., LOARTE, A., NEU, R., SAIBENE, G., SHIMADA, M., STROHMAYER, G., SUGIHARA, M., Key edge physics and plasma-material interaction issues of the ITER divertor ad PFCs, 15th International Conference on Plasma Surface Interactions in Controlled Fusion Devices, Gifu, Japan, 2002. to appear in J. Nucl. Mater.

[19] PHILIPPS V., COAD, J.P., MATTHEWS, G., RUBEL, M., WEINHOLD, P., KIRSCHNER, A., BROOKS, J., FEDERICI, G., contributors tothe EFDA-JET work-programme and the TEXTOR team, Recent results on long-term fuel retention in JET and TEXTOR and predictions for ITER, this conference.

[20] GORDON, C., BARTELS, H-W., HONDA, T., ISELI, M., MOSHONAS, K., RAEDER, J., TAYLOR, N., TOPILSKI, L., An Overview of Results in the ITER Generic Site Safety Report (GSSR), this conference

[21] COSTLEY, A., MALAQUIAS, A., SUGIE, T., VAYAKIS, G., WALKER, C., ITER Diagnostics: Design Choices and Solutions, this conference