STATUS OF THE ITER PROJECT

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ABSTRACT

The status of the International Thermonuclear Experimental Reactor (ITER) Project is presented in terms of its scientific, engineering, safety/environmental characteristics, and operational plan.

1. INTRODUCTION

The ITER Project is a multiphase project presently proceeding under the auspices of the IAEA according to the terms of a four-party 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 (US), referred to herein as "the Parties." [1] 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.

Fusion energy programs throughout the world have benefited from a remarkable degree of openness and global cooperation which has brought with it dramatic progress in scientific understanding and performance achievement. The leading 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 equivalent break even condition[4] and showing a possibility of steady state operation[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 programs is now to progress to study the physics of burning plasmas and to demonstrate and test the key fusion technologies and engineering to establish the feasibility of fusion as an energy source; ITER will fulfill this next step.

The ITER project arose from the recognition by the leading programs 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 programs in an integrated and focused venture.

The current phase of ITER, the Engineering Design Activities (EDA), was established by the Parties and was defined initially for the July 1992 - July 1998 period. Canada and Kazakhstan are also involved in the Project by associations with Europe and Russia. During this period, the Parties agreed to produce a detailed, complete and fully integrated engineering design of ITER and all technical data necessary for future decisions on the construction of ITER. The results of the EDA will then be available to the Parties to use either through international collaboration or within their domestic programs. Significant progress in design and research and development (R&D) has been achieved. The deliverables expected at the end of the EDA will meet the original plan. Mainly due to delay of the construction decision, a three year extension of EDA is foreseen. During this period, site(s) specific design adaptations and safety analysis, preparation of license applications, prototype testing design and R&D including physics studies, and preparation of documents for future procurement are planned. The Parties will also develop proposals and necessary supporting information for complete realization of ITER including a draft agreement for construction and operation.

Detailed technical objectives along with the technical approaches to achieve the overall programmatic objective of ITER were adopted by the Parties[6]. ITER will have two roughly ten-year phases of operation, the Basic Performance Phase and an Enhanced Performance Phase. The first 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 1000s 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 would emphasize improving overall performance and carrying out a higher fluence component and materials testing program. Tritium breeding might be implemented for this phase, depending on the availability of tritium from external sources, the results of breeder blanket testing, and experience with plasma and machine performance.

ITER must also be designed to demonstrate the safety and environmental acceptability of fusion as an energy source.

2. ITER DESIGN

The main parameters summarized in Table 1 were defined after careful study of the balance between physics requirements for plasma confinement, control and stability, and engineering constraints such as heat loads, electromagnetic and mechanical characteristics, neutron shielding and maintainability to ensure safe and reliable operation within reasonable cost.

Table 1 Main Parameters and Dimensions of ITER

Total Fusion Power	1.5 GW
Neutron Wall Loading	1 MW/m^2
Plasma Major Radius	8.1 m
Plasma Minor Radius	2.8 m
Plasma Height	9.9 m
Plasma Current	21 MA
Toroidal Field @ 8.1 m Radius	5.7 T
Maximum Field @ Toroidal Field Coil	12.5 T
Divertor Configuration	Single Null
Auxiliary Heating Power	100 MW

The essential engineering features include:

- an integrated structural arrangement in which superconducting magnet coils (20 cased toroidal field coils, 9 poloidal field coils and a monolithic central solenoid) and vacuum vessel are linked to provide an overall assembly which simplifies the equilibrium of electromagnetic loads in all conditions, relying largely on the robustness of strong TF coil cases; and
- modular in-vessel components (blanket modules on back-plate and divertor cassettes) 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 50m height.[7)] Peripheral equipment such as fueling and pumping, heat transfer, auxiliary heating and remote handling are arranged in galleries around the main pit. If the seismic ground peak acceleration is larger than 0.2 g, isolation will be added, placing a seismic gap at the pit wall, creating an isolated (64 m diameter) "tokamak pit" supported by flexible bearings, still vertically but allowing large horizontal movement ($\gtrsim$ 200 mm). This concept minimizes the design changes which could be necessary if the site selected for construction has a peak ground acceleration larger (0.4 g) than the reference (0.2 g).

The main services required for ITER such as the electricity supply, cooling water, fuel treatment, information flow, assembly and maintenance facilities, waste treatment, etc. are distributed in ancillary buildings and other structures throughout a site about 60 hectares overall.

Table 2 summarizes key engineering features of the design. Figure 1 shows a sector of the in-vessel components and the vacuum vessel with port extensions. Figure 2 shows an elevation view of the equipment layout in the tokamak pit and its galleries. Figure 3 shows the tokamak building and the pit in case of seismic isolation.

Table 2 Summary of Key Engineering Features of the Design

Superconducting toroidal field coils (20 coils)	
Superconductor	Nb_3Sn in circular Incoloy jacket in grooved radial plates
Structure	Pancake wound, in welded steel case
Maximum Field	12.5 T
Superconducting Central Solenoid (CS)	
Superconductor	Nb_3Sn in square Incoloy jacket
Structure	Layer wound
Maximum Field	13 T
Superconducting poloidal field coils (PF 1-9)	
Superconductor	NbTi in square Stainless Steel conduit
Structure	Double pancakes
Maximum Field	5 T (PF 1~8), 6.7 T (PF 9)
Vacuum Vessel	
Structure	Double-wall welded ribbed shell, with internal shield plates and ferro-magnetic inserts
Material	Stainless Steel 316 LN structure, SS 304 with 2% boron shield, SS 430 inserts

1st Wall/Blanket (Basic Performance Phase)	
Structure	Armor-faced modules mechanically-attached to toroidal backplate
Materials	Be armor
	Copper alloy heat sink
	Stainless Steel 316 LN structure
Divertor	
Configuration	Single null
	60 solid replaceable cassettes
Materials	W alloy and C plasma facing components
	Copper alloy heat sink
	Stainless Steel 316 LN structure
Cryostat	
Structure	Ribbed cylinder with flat ends
Maximum inner dimensions	36 m diameter, 30 m height
Material	Stainless Steel 304L
Heat Transfer Systems (water-cooled)	
Heat released in the Tokamak during nominal pulsed operation	2200 MW at ~4 MPa water pressure, 150°C
Cryoplant	
Nominal average He refrigeration /	
liquefaction rate for magnets and Divertor cryopumps (4.5K)	120 kW / 0.25kg/s
Nominal cooling capacity at 80 K	510 kW
Additional Heating and Current Drive	
Total injected power	100 MW
Candidate Additional Heating and Current Drive (H&CD) systems	Electron Cyclotron, Ion Cyclotron,
	Lower Hybrid , Neutral Beam from 1 MeV negative ions
Electrical Power Supply	
Pulsed Power supply from grid	
Total active/reactive power demand	650 MW / 500 Mvar
Steady-State Power Supply from grid	
Total active/reactive power demand	230 MW/160 Mvar

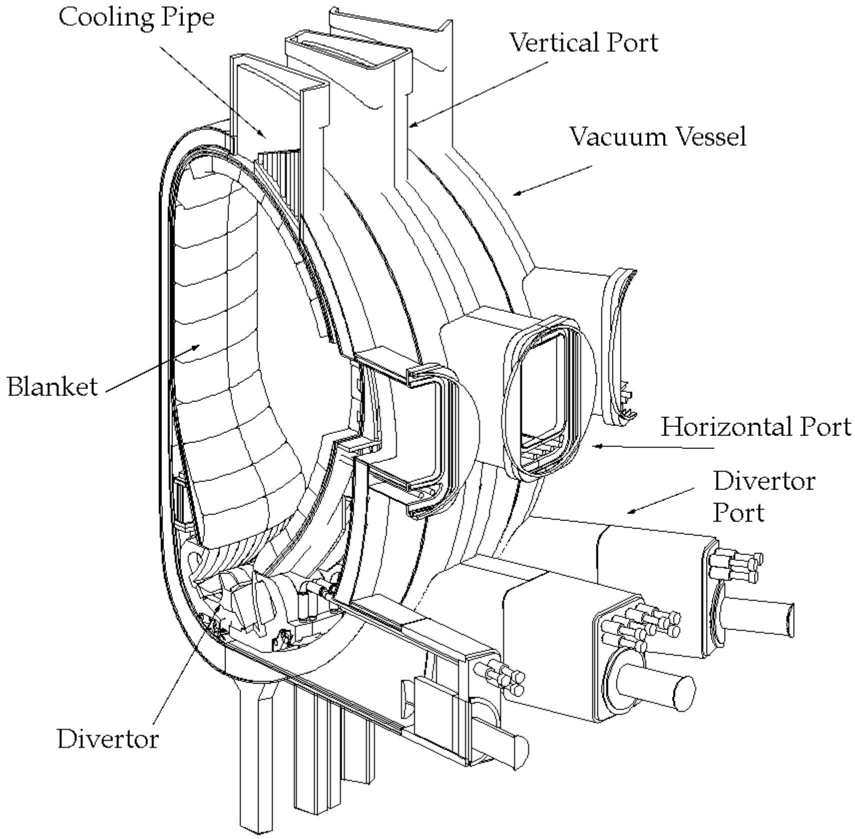


Figure 1 Isometric View of Vacuum Vessel, Blanket and Divertor

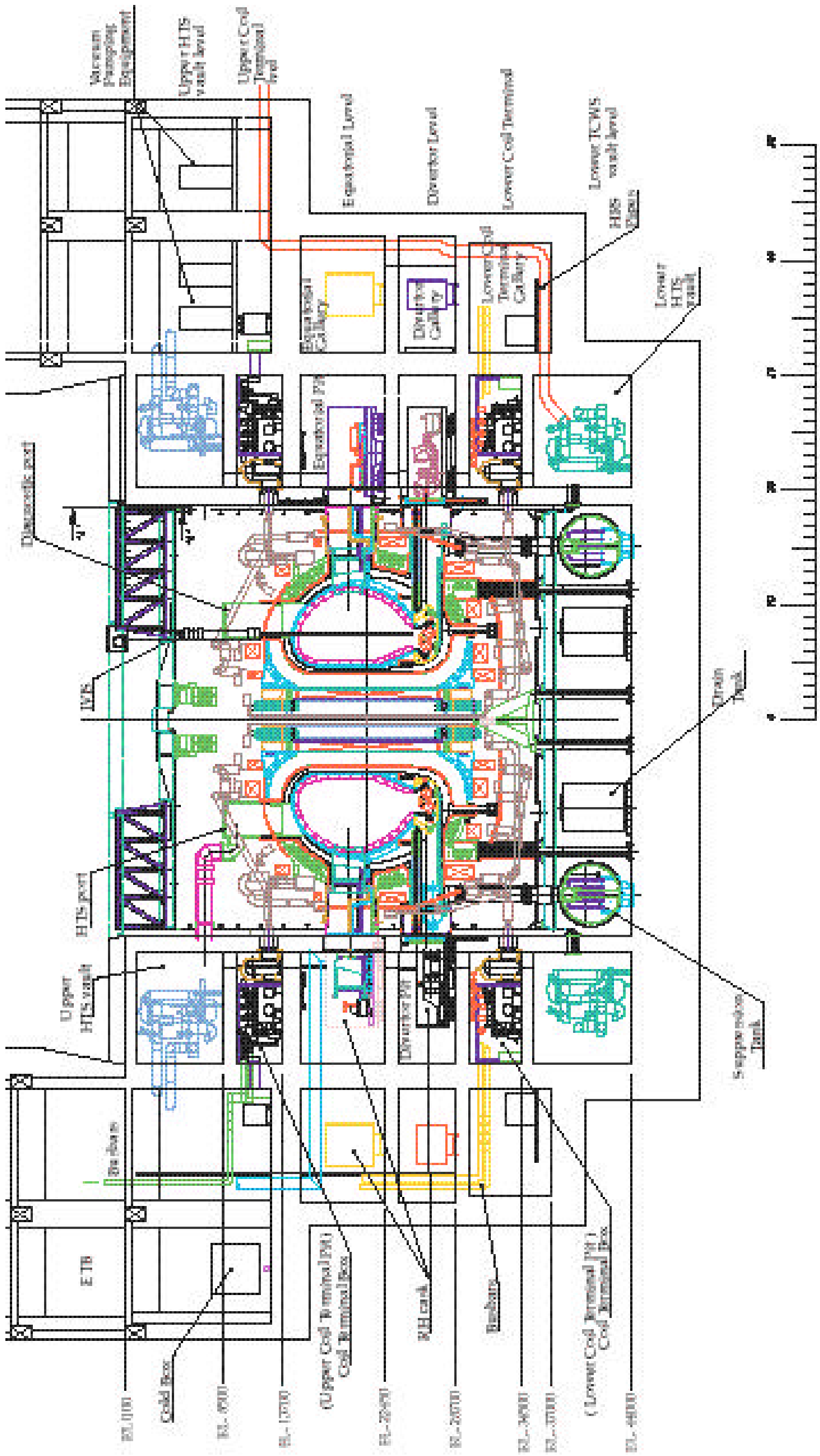


Figure 2 Elevation View of the Equipment Layout

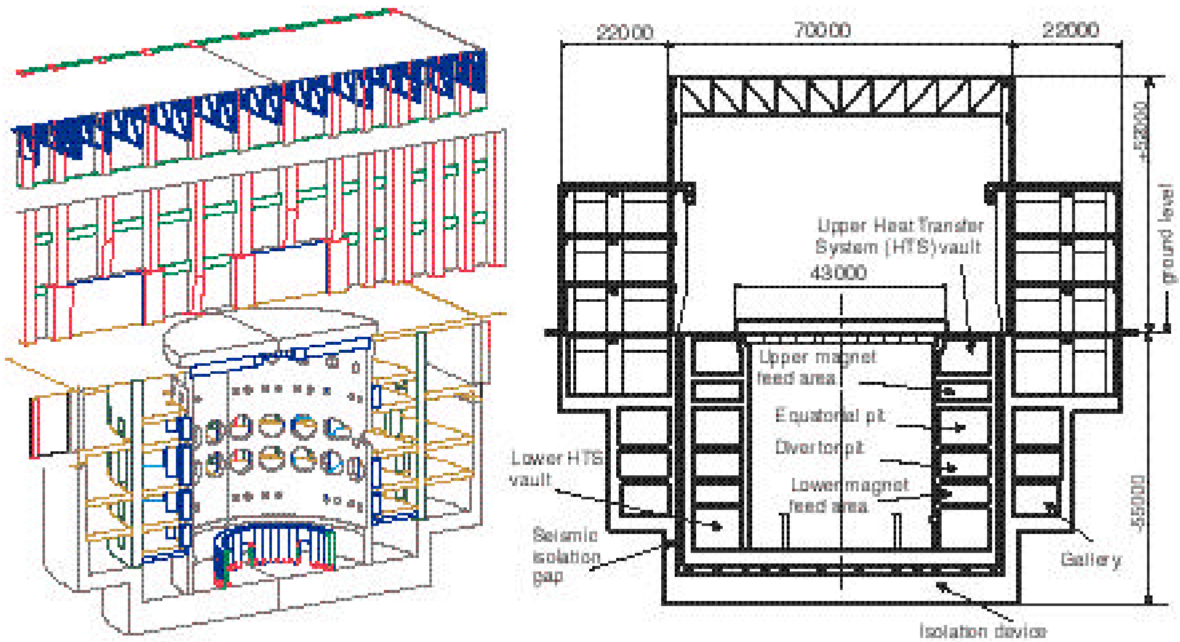


Figure 3 Tokamak Building and Pit. Left: north-south cutaway view. Right: east-west cross-section view. (in case of seismic isolation)

3. ITER SAFETY[8)]

Safety objectives of ITER are as follows:

- ITER shall be designed to be site-able in any of the Four Parties;
- ITER shall be designed, constructed, operated and decommissioned to ensure the protection of the public, site personnel and the environment; and
- ITER should demonstrate the safety and environmental potential of fusion.

In order to ensure ITER would be site-able 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 all Home Teams and has been implemented in the ITER design. This includes radioactive dose and release design guidelines established in accordance with internationally accepted conservative criteria and the principle of As Low As Reasonably Achievable (ALARA), and the well-established nuclear design concepts of Defense in Depth and Multiple Lines of Defense.

A comprehensive safety assessment of the ITER design has been completed and results show:

(a) A high level of safety is integrated into the ITER design.

General safety design requirements were established with Home Teams including the conservative radioactive release limits. The design incorporates the well-established concepts of Defense in Depth and multiple lines of defense to attain high confidence in the reliability of critical safety features of the facility and 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 principles of ALARA. Although there are uncertainties in the release estimates, there is adequate flexibility in the design to modify atmospheric and liquid control systems for improved performance.

(c) The ITER design ensures protection of the public.

A comprehensive analysis of Reference sequences has been performed using the best safety analysis computer programs available worldwide with conservative assumptions. Radioactive releases are well within the ITER design release limits conservatively established. Two fundamentally different approaches, i.e., bottom-up and 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 enveloped by the assessed consequences of the reference events.

(d) Waste and Decommissioning

Wastes stream has been studied in detail and a phased ITER Decommissioning Scenario has been developed by maximizing the use of existing facilities and equipment and using the advantage of cool down effects. All in-vessel components can be dismantled by the existing remote maintenance equipment used during ITER operation and all ex-vessel components can be dismantled by conventional tools with human access after thirty years. Only the vacuum vessel requires assistance of remote operation or a further mothball for a later dismantle with human access. The major final radioactive waste is the only in-vessel component.

(e) Occupational Safety

Radiation protection and ALARA analysis have been started. Throughout the life cycle of ITER, the Radiation Protection Program will continue to be updated and occupational safety consideration incorporated in the future design work.

In addition to these studies, ultimate safety margins have been studied in order to demonstrate the intrinsic positive safety characteristics of magnetic fusion such as:

- The fusion reaction is self-limiting bounded by the ß-limit of the plasma. Under any failure conditions of the vacuum vessel or the in-vessel components, the fusion reactions are physically impossible.
- The radioactive inventory is moderate and the ultimate performance of confinement barriers that needs to be assured in accidents will be about one order of magnitude reduction for tritium and mobilizable metallic dust for ITER, whereas six to seven orders of magnitude reduction is required for iodine and rare gas 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 defense associated with failure of safety functions and potential energy sources, i.e. decay heat removal or confinement barriers, coolant energy and relating over pressure, plasma energy due to failure of power shutdown, steam-first wall material reaction due to over heat, 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 is mainly induced from the intrinsically positive safety and environmental characteristics of fusion such as passive fusion power shutdown, fusion power limitation, modest radioactive 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 attached with the pressure suppression tank and the cryostat. These features are common in a magnetic fusion reactor and ITER has the same level of size, power and radioactive inventories in a Fusion Power Plant. Therefore these favorable safety characteristics of ITER show fusion's safety and environmental potential.

Concluding the safety assessment, ITER could be constructed and operated without undue risk to health and safety, and without significant environmental impacts. A study on ultimate safety margins shows favorable safety characteristics of magnetic fusion. Home Team Expert reviews indicate that a technical basis 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. Major technical challenges in ITER are as follows:

- unprecedented size of the superconducting magnet 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/intervention of an activated Tokamak structure;
- the first fusion machine with large radioactive inventory; and
- unique equipment for fusion reactors such as fueling, pumping, heating/current drive system, diagnostics etc.

ITER is being supported by extensive technology R&D to validate key aspects of design, including development and qualification of the applicable technologies and development and verification of industrial level manufacturing techniques with related quality assurance (QA). Technology R&D for ITER is now focused on seven large projects each devoted to one of the key aspects of the design as follows.

Two of the Projects are directed 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 strand, cable, conduit and terminations, and to integrate the supporting R&D programs on insulators, joints, conductor ac losses and stability, Nb_3Sn conductor wind, react and transfer processes, 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 tons, 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 were fabricated in Japan and the US and will be assembled soon.

For the TF model coil, forging of the radial plates and cabling and jacketing work are complete and the molds are ready for the heat treatment in EU. Dedicated coil test facilities, for the CS Model Coil in Japan and for the TF Model Coil in the EU, have been completed and stand ready to install the model coils for test programs aimed at gaining broad experience in their operational flexibility and to understand their performance margins. A 1 km jacketing has been completed in RF which confirmed the fabrication feasibility of the full size both in the length and the cross section.

Three Projects focus on key in-vessel components, including development and demonstration of 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 are currently being welded to each other at the Japan Atomic Energy Research Institute (JAERI) 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, and 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 Japan. 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. This approach will be tested with prototype components to confirm that it meets the anticipated loads, the electrical insulating and the remote handling requirements together with the necessary accuracy of positioning in EU.

The Divertor Cassette Project aims to demonstrate that a divertor can be built with tolerances and to withstand the very 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 US. The key technologies of the high heat flux components of the divertor have been successfully demonstrated in the four Parties, using W-alloy and CFC as plasma facing materials bonded to copper cooled by high velocity water using both hypervaportron and swirl-tube technologies.

The last two of the Large Project focus on ensuring the availability of appropriate remote handling technologies which allow intervention in contaminated and activated conditions on reasonable time scales. 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 on a long period of time in order not only to check the right procedures, but also to optimize their use in detail and minimize the intervention time. 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 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/cost for the development of full-scale equipment. Work is now in progress on a full-scale demonstration. The fabrication of the full-scale equipment/tools, such as rail-mounted vehicle/manipulator system, and cooling pipe welding/cutting/inspection tools has been completed in Japan. Integrated tests in a Blanket Test Platform which simulates the full scale structure of a 180 degree ITER in-vessel region, are aimed at providing a comprehensive validation of the remote handling system so as to allow completion of the detailed design of components and the remote handling equipment.

In the Divertor Remote Handling Development, 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 EU.

In addition to these large projects, development of key components for fueling, heating/current drive, tritium process and diagnostic systems, irradiation tests and safety relating R&D are in progress.

The technical output from R&D has direct importance in validating the technologies and related manufacturing techniques and QA incorporated in the ITER Design and in supporting the manufacturing cost estimates for some key cost drivers. The activities are foreseen as continuing beyond July 1998 to further the prototype component testing and/or to optimize their operational use. Their performance also offers insights 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.[9) 10)] The successful progress of the projects increases confidence in the possibility of jointly constructing ITER in an international project framework.

5. ITER OPERATION

The construction schedule that leads up to the first hydrogen plasma operation was developed based on analysis of procurement, fabrication, installation and commissioning of all the ITER systems and gives 9 years from the start of the purchase order for the Tokamak building and the super conducting 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 ensures that all of the ITER plant is ready to operate, except some subsystems such as tritium plant, hot cells and radioactive material storage, unneeded in the first operation period with hydrogen plasma. ITER will have two phases of operation “Basic Performance Phase” (BPP) and “Enhanced Performance Phase” (EPP). Major operation features are summarized below.

5.1 Basic Performance Phase (10 years)

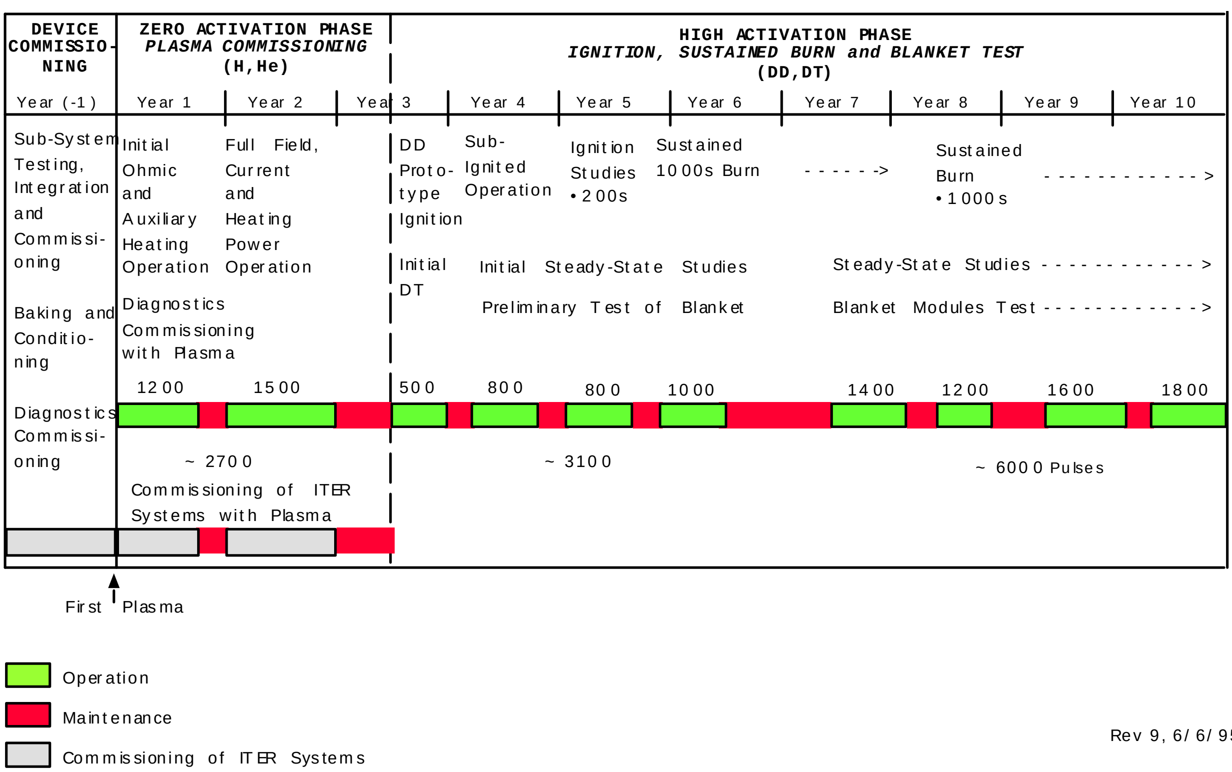
Operation of ITER will progress step by step from hydrogen plasma operation with low plasma current, low magnetic field, short pulse and low duty factor without fusion power to deuterium-tritium plasma operation with full plasma current, full magnetic field, long pulse and high duty factor with full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

During the first 2.5 years, hydrogen plasma experiments will be done, no fusion reaction occurs and ITER in-vessel components are not activated nor 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, especially with regard to shielding performance. The fusion power and pulse length will be gradually increased. This approach ensures safe and reliable operation of ITER. In the fifth year, the reference operation with 1.5 GW and 1000 sec burn pulse is planned to be achieved. In parallel with the development of the reference operation, various operation modes including the steady state operation will be studied. Nominal plasma parameters in the reference scenario and a possible range of burning plasmas are summarized in Table 3 and the plan of BPP is summarized in Table 4.

Table 3 ITER Plasma Parameters

Parameters	Nominal	Range
Fusion power (MW)	1500	500 - 1500
Burn duration(s)	1300	500 - 10000
Auxiliary power, P_{aux} (MW)	0	= 100
$ß_N$[={ß}(%)a(m)B(T)/I(MA)]	2.25	1-4
Safety factor, q	3	2.6 - 5.2
Electron density, {n} ($10^{20}m^{-3}$)	0.98	0.5 - 1.5
Average temperature, {T} (keV)	12.8	5 - 30
Energy confinement time, τ_E(s)	5.8	3.5 - 9
Impurity fractions (n_Z/n_e)(%): Be/He/Ar	2/9/0.17	-
Z_{eff}	1.8	-
Power to divertor targets (MW)	50	= 100

Table 4 Tentative ITER Plasma Operation Plan for Basic Performance Phase



In the Basic Performance Phase, tests of ITER tritium breeding blanket for the next phase, i.e. the Extended Performance Phase, and blankets for the Demonstration Fusion Reactor (DEMO) will be started.

At present, four tritium breeding DEMO relevant blanket concepts are planned in the Parties' program for testing in ITER, in addition to one for the breeding blanket for the EPP of ITER to produce a large fraction of the tritium fuel. ITER has assigned four equatorial ports during the BPP for testing tritium breeding blankets.

The ITER blanket test program during the BPP is intended to achieve the following main objectives:

1) Evaluate the performance of the ITER breeding blanket for the EPP and the DEMO relevant blanket designs;
2) Demonstrate tritium breeding performance and verify the on-line tritium recovery and control systems;
3) Demonstrate the high-grade heat extraction and the electricity generation available from one of the DEMO relevant blankets;
4) Validate and calibrate the design tools and the data base used in the blanket design process including neutronics, electromagnetic, heat transfer, hydraulics;
5) Demonstrate the integral performance of the blanket systems under different loading conditions; and
6) Observe possible irradiation effects on the performance of the blanket modules.

Accumulation of average neutron fluence on the first wall is planned to be up to 0.3 MWa/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.

5.2 Transient Phase from BPP to EPP (2 years)

For the EPP Phase, the shielding blankets will be replaced by the breeding blankets because external tritium resources are not sufficient for a significantly higher fluence than that of the BPP. This process requires about 2 years. A tritium breeding ratio of about 0.8 would be sufficient to provide about 1 MWa/m^2 during 10 years operation assuming an external supply of 1.7 kg tritium per year.

5.3 Enhanced Performance Phase (about 10 years)

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

5.4 Remote Experiment Concept

In order to use ITER efficiently and to involve large fusion communities within the Parties, remote experimental capabilities are foreseen. One possible mode of operation involving remote experimental site(s) is as follows:

3 shifts/day on site

Most people will work during the day, i.e, 1 or 2 shifts for experimentation. A limited number of people will work during the night shift for minimum machine monitoring and to support experiment from remote sites.

1 shift (or 2 shifts)/day on remote experimental site(s).

Remote experimentation will be done within the envelope of parameters and conditions agreed to in advance or given by the on-site control room.

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 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 of limited activities like discharge cleaning, similar to those of the present large Tokamaks, may fit in this phase. Remote experimental sites could be introduced after this period.

6. CONCLUSIONS

1. The ITER design, given in the final design report[11)], is at an advanced stage of maturity which contains the necessary technical information to satisfy the purpose of the EDA Agreement and to start the site(s) specific design adaptations.

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

3. The program 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 components prototypes, with related QA. 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 is a valuable asset to any possible future collaboration in fusion development.

4. 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, characteristics of plasma will be understood which will significantly reduce uncertainties in the next step. This approach enhances safety and reliability of the ITER operation.

5. The ITER project has so far proved to be an unprecedented and successful model of international cooperation in science and technology in which all participants benefit not only from the technical results but also from the experience of different approaches to project organization and management. It has proved to be an effective and efficient vehicle for the fusion engineering needed to realize any concepts of commercial magnetic fusion reactors. Bringing ITER to full realization through joint construction and operation will continue this process.

6. A three-year extension of the EDA is 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. It is also planned to continue the technical work for the project in tasks that will help reinforce the technical basis for a positive construction decision such as:

- adapting the design to the specific characteristics of possible construction sites;
- supporting preparations for formal applications for licenses to build and operate ITER;
- extending prototype testing to provide data on operational margin;
- finalizing the design and procurement specifications and related documentation for the ITER systems taking into account industrial capabilities; and
- consolidating the scientific basis of ITER operations.

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8. KEY WORDS

Fusion, Tokamak, ITER, Experimental Reactor, International.