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Present status and future prospect of the ITER project

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Abstract

The present status of the ITER project is summarised in terms of progress made on its scientific, engineering and safety/environmental characteristics and of its position in world-wide fusion development. The selection of materials and joining technologies for ITER is discussed, with emphasis on materials choices and test results for in-vessel components. Progress of major R&D projects to validate the technologies is presented. The future prospects for ITER are considered. © 1998 Elsevier Science B.V. All rights reserved.

1. Introduction

ITER – the International Thermonuclear Experimental Reactor – is a collaborative project being undertaken jointly by the world's leading fusion energy programmes with the objective of demonstrating the scientific and technological feasibility of fusion energy for peaceful purposes. It represents the next step for the magnetic fusion development programmes of the four main participants, Euratom, Japan, the Russian Federation and USA.

The current, Engineering Design Activities (EDA) phase of ITER was defined initially for the six years from July 1992 during which the Parties agreed jointly and on a basis of equality 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 will be available to the Parties to use either through international collaboration or within their domestic programmes.

The results of the technical work to late 1996 were embodied in a Detailed Design Report, Cost Review and Safety Analysis [1] which was presented to the ITER Council in December 1996 and approved, following domestic reviews in the Parties, in July 1997. The detailed design work and associated research and development programmes for ITER now continue towards the next major milestone – the Final Design Report – due to be presented in February 1998.

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2. Scientific and technological challenges

The main characteristics and parameters of the ITER Design follow from the agreed programmatic and detailed technical objectives for ITER. These provide the main design drivers for the EDA. The Physics requirements include:

- achieving controlled ignition (~1.5 GW for 1000 s) with finite but limited margins on most likely projections from current fusion experiments;
- ensuring 1 MW/m^2 of 14 Mev neutrons in driven burn;
- providing flexibility to explore a range of possible operating scenarios including recently established advanced Tokamak discharges.

To achieve and sustain such plasma performance over its planned twenty year operational programme, the ITER design has to address the engineering challenges foreseen for future fusion power stations, notably:

- reliable containment and control of burning plasma;
- very large superconducting magnet and structures;
- in-vessel structures (blanket and divertor) able to withstand high heat and neutron fluxes and electromechanical forces;
- remote handling systems for maintenance/intervention of an activated tokamak structure;
- D/T fuelling and fuel processing systems;
- tritium breeding capability (for ITER's second, Enhanced Performance Phase).

The main parameters and overall dimensions of ITER are as summarised in Table 1. Overall the parameters reflect a careful balance of physics requirements for

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Table 1
ITER main parameters and dimensions

Total fusion power	1.5 GW
Neutron wall loading	1 MW/m^2
Plasma inductive burn time	1000 s
Plasma major radius	8.1 m
Plasma minor radius	2.8 m
I_p	21 MA
k_{95} (ellipticity @ 95% flux surface)	1.6
q_{95} (safety factor @ 95% flux surface)	3
b_0 @ 8.1 m radius	5.7 T
b_{max} @ TF coil	12.5 T
TF ripple at separatrix	<1%
Auxiliary heating power	100 MW

confinement, plasma control and stability, and engineering/materials constraints (e.g., heat loads and electromagnetic characteristics, need for access) to ensure safe and reliable operation within reasonable cost. Materials issues are at the heart of design choices that must be made with regard to components to be used in an unprecedented complex of operating conditions. They are reviewed in the following section.

The essential engineering features of the Tokamak Design are illustrated in cross-section in Fig. 1. They 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, relying largely on the robustness of strong TF coils cases;
- 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, in an underground pit, inside a building about 50 m high. Peripheral equipment such as fueling and pumping, heat transfer, auxiliary heating and remote handling and management are arranged in galleries around the main pit. The main services required for ITER such as electricity supply, cooling water, fuel treatment, information flow, assembly and maintenance facilities, waste treatment etc are distributed in ancillary buildings and other structures on an overall 60 hectare site.

The Detailed Design Report incorporated the results of a first comprehensive non-site-specific safety and environmental assessment of ITER design and operation, which have shown that the ITER design provides a robust system which meets the safety-related requirements by exploiting the inherent safety characteristics of fusion as compared with other power sources and by following a design policy of defence in depth and providing

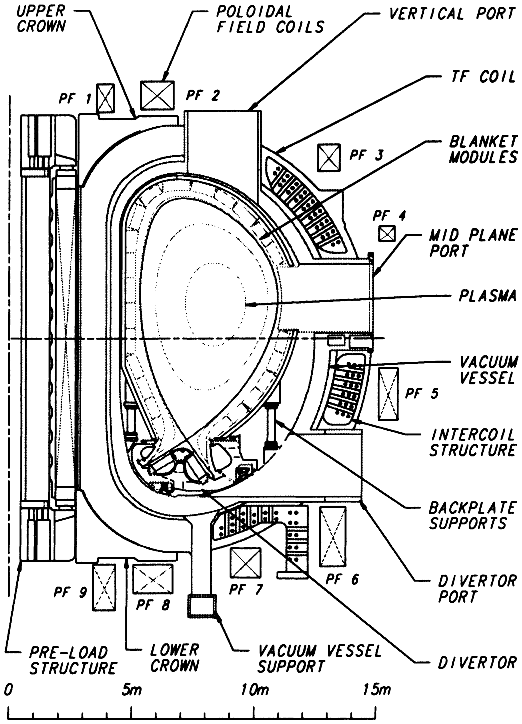

Fig. 1. Cross-section of the ITER Tokamak.

structural margins in key areas. Table 2 summarises key features and materials choices in the design.

3. Operating scenarios and physics performance

The plasma discharge scenario for ITER will follow the familiar sequence used in present Tokamaks. Operation with a sawtoothing, Edge-Localized Mode (ELMy) H-mode is used as the basis for projections of ITER performance in the ignited or high-Q driven-burn regime. Long pulse ELMy discharges with ITER-like plasma geometries and parameters (in relevant dimensionless values) have been demonstrated in all major divertor Tokamaks, H-mode being achieved when sufficient heating power is applied to the plasma.

In driven burn operation, auxiliary heating power of 100 MW allows extension of pulse lengths and maintenance of high neutron wall load, providing high fluence for testing. From the reference pulse at 21 MA, 1.5 GW and 1000 s, the pulse length can be extended through a range of values of I, GW, and t, up to 8000 s at about 1 GW with a plasma current of about 16 MA. True steady-state operation can be obtained at 12–15 MA with the available auxiliary heating/current drive capability to supplement a large bootstrap current fraction (>70%). In this scenario, the radial distribution of current density provides a configuration with "reversed

Table 2
Main engineering features of the ITER systems

Superconducting toroidal field coils (20 coils)	
Superconductor	Nb_3Sn in circular Incoloy 908 jacket in grooved radial plates
Structure	Pancake wound, in welded steel case wind, react and transfer technology
Superconducting Central Solenoid (CS)	
Superconductor Structure	Nb_3Sn in square Incoloy 908 jacket layer wound, 14 layers, four conductors in-hand wind react and transfer technology
Superconducting polodial filed coils (PF 1-9)	
Superconductor	NbTi in square stainless steel conduit
Structure	Double pancakes, typically two conductors in-hand
Vacuum Vessel	
Structure	Double-wall welded ribbed shell, with internal shield plates and ferromagnetic inserts
Material	Stainless Steel 316 LN structure, SS 304 with 2% boron shields, SS 430 inserts
1st Wall/blanket (Basic performance phase)	
Structure	Armour-faced modules mechanically-attached to toroidal backplate
Materials	Be armour 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 (4.5 K)	120 kW/0.25 kg/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

shear'', which has been demonstrated successfully in recent Tokamak experiments with very good performance in confinement, but for limited duration.

The ITER divertor plasma will be operated in the ''detached'' or ''partially detached'' regime achieved on modern divertor Tokamakas. This regime has low heat loads on the divertor targets and provides adequate helium exhaust.

Recent experimental results generally confirm and strengthen the physics basis for the ITER design. According to the present most validated way of projecting ITER plasma performance, ignition in ITER is probable with the reference plasma parameter and confinement values. With up to 100 MW of auxiliary heating, a sustained burn at 1–1.5 GW of fusion power will be achieved, while simultaneously satisfying divertor heat load and H-mode power threshold, within a wide range of plausible conditions and reduction in confinement by ~30% from the reference value.

Whilst uncertainities in extrapolation from present plasmas to the ITER size and parameters cannot be completely avoided, the wide-ranging reviews and analyses of the Expert Groups, have not indicated limitations in principle which could preclude ITER from achieving the required performance and objectives.

4. ITER engineering and materials

The overall philosophy for ITER engineering has been to use established approaches and well characterised materials 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. This approach carries through into cost estimating whereby established industrial firms with relevant experience throughout the Parties are undertaking costing studies on about 80 "procurement packages" each based around typical prospective procurement contracts for the main components of ITER.

The selection of materials and joining technologies to be used in ITER is a trade-off with many, often conflicting requirements. Materials selection must encompass a total engineering approach, considering not only physical and mechanical properties and processing, but also the maintainability, reliability, replaceability, and recyclability of each material. The chemical composition must also be optimised to reduce waste disposal problems to a minimum.

In line with the general design philosophy, the material choice has been oriented, as far as technically feasible, toward industrially available materials and well established manufacturing techniques. This is very much the case for the structural materials of the basic machine (cryostat, magnet case, vacuum vessel), for which a critical factor is the availability of industrial suppliers with experience in forming and joining technology. The structural integrity of these components throughout the entire design lifetime is important for the machine availability and safety. Austenitic stainless steels are the most suitable, as these materials are qualified in many national design codes, have good properties and a large experience base for cryogenic applications. Moreover, they have good weldability, forging, and casting potential.

The materials for the in-vessel components will operate under the simultaneous influence of different life-limiting factors, such as neutron irradiation, hydrogen atmosphere, dynamic stresses, thermal loads, cyclic mode of operation, and water cooling environment. Even though no safety functions are attributed to the invessel components, to achieve good performance and adequate availability of the whole machine they have to remain highly reliable throughout the design lifetime. Ease of fabrication, good weldability, resistance to corrosion, good strength and fatigue resistance, adequate ductility and fracture toughness after neutron irradiation are essential requirements.

The following paragraphs summarise briefly the choice of materials for the in-vessel components. A more complete treatment can be found in [2]. The related data base on physical and mechanical properties, including the effect of neutron irradiation, is reported in [3].

The chosen structural material for the in-vessel components is 316L(N)-IG (ITER Grade), based on the AISI 316L specification with a narrower composition range of the main alloying elements and a controlled addition of nitrogen. It has satisfactory resistance to stress corrosion cracking, and high levels of strength and fracture toughness. There is an extensive database in the unirradiated and irradiated condition, and a large industrial experience in nuclear applications [4]. The ongoing characterisation of solid and power HIPed steel confirm that HIPing can be successfully applied to the manufacture of the shielding blanket module, including the copper to steel joint. For the fluence expected in the first, Basic Performance, phase of operations, the behaviour under neutron irradiation, makes it possible to design according to the properties of unirradiated material.

The present development of low activation structural materials does not allow their massive use in ITER. In the future, these materials, after irradiation to a much higher neutron fluence than expected in ITER, should show a much lower amount of long lived radioactive components and retain their satisfactory mechanical properties.

Two copper alloys have been selected for the heat sink of the plasma facing components (PFCs), an age-hardenable CuCrZr alloy and a dispersion strengthened (DS) alloy. The composition, manufacturing process and heat treatment have been optimised. Mechanical properties of both alloys are sufficient for the components to sustain thermal and mechanical loads and achieve the design lifetime, even considering the damaging effects of neutron irradiation. In the relevant temperature range, CuCrZr-IG, if correctly heat treated, has better tensile properties and a higher fracture toughness than CuAI-25-IG. As a consequence of component manufacturing (high temperature HIPing, armour brazing), however, CuCrZr-IG may loose, by over-ageing, strength and thermal conductivity. In that case, CuAl25-IG is recommended as heat sink material [5]. With the specified water chemistry, uniform corrosion rates are relatively low in copper alloys and there is no evidence of erosion-corrosion effects.

Beryllium has been chosen as the armour material for $\sim$80% of the total surface exposed to the plasma (primary wall, upper baffle and port limiters), on the basis of low plasma contamination, absence of chemical sputtering, oxygen gettering capability, low retention of tritium and possibility of armour repair. The selected reference grade is S-65C beryllium (with DShG-220 as back up), because of its resistance to thermal fatigue, availability, and previous experience in JET. Beryllium plasma spray (PS) is also considered for both the initial fabrication of the primary wall and for repairing the damage created by interactions with the plasma. PS has yielded thick (10 mm) coating with 98% of theoretical thermal conductivity at 98% density. The heat flux limit is $\sim$2 MW/m^2, making PS Be coatings suitable only for the primary wall. Its high sputtering rate makes beryllium less suitable in areas where charge exchange sputtering is the dominant erosion mechanism (divertor lower baffle, upper vertical target); here, tungsten provides the best erosion lifetime.

No matter what grade is chosen, beryllium and tungsten in the colder region near the heat sink will be brittle after neutron irradiation. The solution to armour embrittlement is found by designing the tile to reduce the thermal stress. In the brush-like or lamellar design (Fig. 2), the armour is subdivided in small rectangular lamellae, individually attached to the heat sink via a pure copper compliant layer. The single elements are free to expand under the heat flux, reducing the thermal stress in the tile. This design prevents the propagation of fabrication flaws at the interface.

Remarkable results have been obtained in the development of new methods to join Be, W and CFC to the heat sink. The characterisation under thermal fatigue conditions of these joints has demonstrated that for each armour material more than one joining method is available which fulfils the thermal fatigue requirement of the PFCs. Table 3 summarises the best results of the thermal fatigue tests on small scale mock-ups of the PCFs.

In areas hit by large thermal fluxes during normal operation and large energy dumps plasma instability (divertor lower vertical target, dump plate), Carbon Fibre reinforced Carbon (CFC) is selected [6], because it can resist very high heat fluxes and does not melt. However, its use has to be restricted to these regions, because of the problems of chemical erosion and tritium retention, especially in the codeposited layers. The development of Si-doped CFC could mitigate the problem of chemical erosion. Again, the solution to the high heat fluxes is found in the tile geometry; the monoblock or the saddle-block both have very high thermal fatigue resistance, also after neutron irradiation.

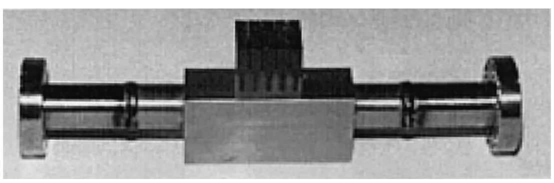

Fig. 2. Lamellar W/Cu divertor element mock-up.

In summary, industrially available materials have been optimised for the ITER conditions and the database on their design relevant properties significantly improved during the EDA. New technologies have been developed for the armour/heat sink joints and repair methods devised to reduce the amount of radioactive waste. Plans are to complete the qualification of materials and manufacturing methods to confirm the positive results obtained. In particular, the thermal fatigue performances have to be confirmed on larger mock-ups and on full-scale prototypes and the effect of neutron irradiation on the joints has to be ascertained.

5. Validating Technology R&D

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 QA. Technology R&D for ITER is now focussed on seven large projects each devoted to one of the key aspects of the design. All the projects are expected to yield important results by July 1998. But, in general, the test programmes can be seen as extending beyond that date to yield further data on operating margins and on optimising flexibility and to provide possible test-beds and training facilities for future operators.

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 Model Coil Project and the Toroidal Field Model Coil Project drive the development of the ITER full-scale conductor including strand, cable, jacket/conduit and terminations, and integrate the supporting R&D programmes 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 and the required testing facilities. The total planned production of 29 tonnes of Nb_3Sn strands, from seven different suppliers throughout the four Parties, has been produced and qualified. The cabled strands have been drawn in a jacket/conduit made of Incoloy (a high Nickel alloy), the thermal expansion of which matches that of Nb_3Sn.

Since the reaction treatment to produce Nb_3Sn requires a temperature of about 700°C, the Project has had to address the major material issue arising from the susceptibility of high nickel alloys such as Incoloy to cracking under certain conditions of temperature, stress and oxygen concentration. Intensive study of the so-called SAGBO (stress accelerated grain boundary oxidation) phenomenon showed that the problem can be

Table 3
Results of the thermal fatigue test on small scale mock-ups of PFCs

Armour/geometry (mm)	Technology/parameters	Cu alloy	Coolant temp.(°C)	Heat flux (MW/m^2)	Cycles	Observ.	Ref.
S-65C, 16 × 16 × (5–10)	HIP (625°C, 1 h, 155 MPa) [(Be+PVD Al&Cu) + AIBeMet + (Expl. BondedTi onto CuNiBe)]	CuNiBe	20	5 10	1000 1000	No damage No damage	US
TGP 5 × 5 × 5 × 32 tiles	Brazing CuInSnNi, (∼820°C, few min)	CuCrZr	20	12	2000	No damage	RF
W lamella type ∼20 × 18 × 5	Casting of oxygen free high conductivity copper + CuInSnNi brazing to Cu heat sink (800°C, few min)	CuCrZr	20	15	2150	No damage in joint, Some creep deformation	RF
W–La_2O_3 macrobrush 4.5 × 4.5 × 10	Active metal casting of oxygen Free high conductivity copper (2 mm)+ e-beam welding	DS Cu	40	9 16	1000 1000	No damage 1 tooth fell off after 927 cycles, some creep deformation	EU
Dunlop C I, Mono-block, 40 × 22 mm thickness 6/12 mm	Active metal casting + Ti brazing (880°C, 10 min); (6 mm when flux ⩾ 15 MW/m^2)	CuCrZr	20	15 24	1000 1000	No damage No damage	EU
3D-CFC Saddle	Cu–Mn braze on oxygen free copper clad DS–Cu	DS–Cu	20	20	1000	3 tiles no damage. 1 tile eroded	JA

avoided by careful control of tension stress and oxygen concentration throughout the process. This has allowed the next critical step for the CS model coil – the heat treatment to react the superconducting alloy without degrading the mechanical properties of the jacket – to be successfully achieved by the Home Teams in Japan and the US. Fig. 3 shows coil winding for the model CS in progress.

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. Each 18° sector weighs about 220 t, including as much as 20 t of weld material. The development of fabrication procedures and control of welding distortions within required tolerances to achieve dimensional accuracy is therefore critical. The key technologies have been established and, in relation to manufacturing techniques, two full-scale vacuum vessel segments (half sectors) have been fabricated in Japanese industry to tolerances of 2.5 mm using a range of welding techniques including Tungsten Inert Gas (TIG), Metal Active Gas (MAG) and Electron Beam (EB). The two half sectors will now be welded together. Fig. 4 shows a completed half sector.

The in-vessel components that face the plasma must operate reliably and predictably in the demanding conditions of ITER operation, including extreme heat fluxes and neutron fluence, electro-magnetic forces from VDE's and disruptions and internal temperature gradients. The systems must also be amenable to remote maintenance and replacement; for this reason a modular approach is adopted. In general the components must

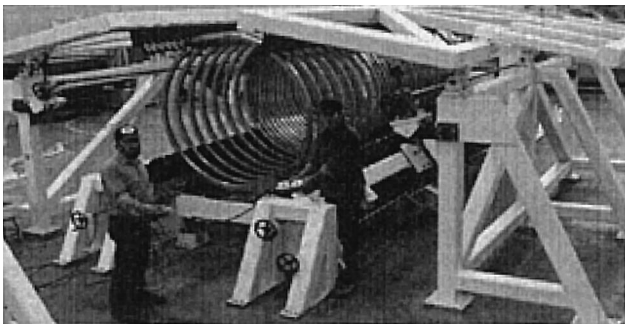

Fig. 3. ITER central solenoid model coil winding.

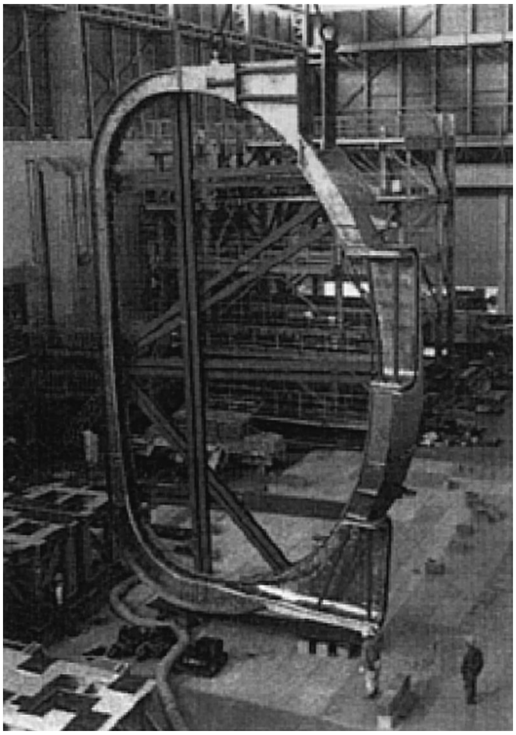

Fig. 4. Half-sector prototype of the vacuum vessel.

incorporate appropriate combinations of armour, heat-sink and structural materials; and the development and validation of practical material bonding and fabrication techniques are therefore essential.

The Blanket Module Project is aimed at producing and testing full scale modules of primary wall elements and at demonstrating prototype integration in a model sector. The technology development has successfully developed, tested and qualified a range of critical material interfaces such as Be/Cu and Cu/Stainless Steel, bonded using advanced techniques such as single step "solid" HIP and powder HIP. Fig. 5 shows a shield blanket tube gallery prepared for incorporation into a prototype module manufactured with powder HIP technology.

The Divertor Cassette Project aims to demonstrate that a divertor can be built within tolerances and to withstand the thermal and mechanical loads imposed on it during normal operation and during transients such as ELM's and disruptions. To this end, a full-scale prototype of a half-cassette is being built and subjected to high heat flux and mechanical tests. The critical materials requirement is to sustain heat loads of about 5 MW/m^2 during a required lifetime of 3000 full power discharges. As noted above, significant progress has been made in the development and testing of silver-free armour/heat sink bonds. The project also includes tasks to understand erosion mechanisms, to develop methods

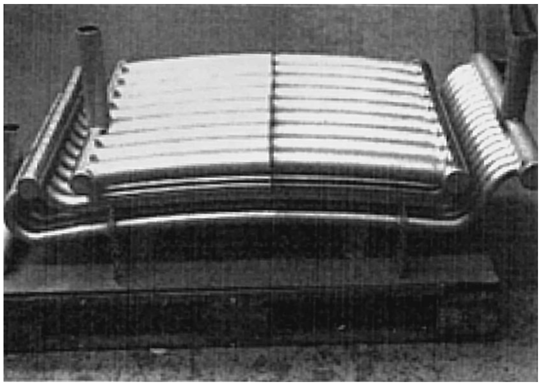

Fig. 5. Preparation of tube gallery for powder HIP fabrication of a blanket/shield module.

of dust removal and of outgasing tritium codeposited with Be or C.

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 on a long period of time in order not only to check the right procedures, but also to optimise their use in detail and minimise 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. 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 full scale demonstration. 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 is in progress.

The technical output from the Seven Large R&D Projects has direct importance in validating the technologies and related manufacturing techniques and QA for ITER and in supporting the manufacturing cost estimates for key cost drivers. The activities are foreseen as continuing beyond July 1998 to further the prototype component testing and/or to optimise 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. The successful progress of the projects increases confidence in the possibility of jointly constructing ITER in an international project framework.

6. Conclusions and outlook

1. The ITER design is now at an advanced stage of maturity. There is a stable design which provides the basis for continuing engineering activities, aimed toward providing the necessary technical information to satisfy the purpose of the EDA Agreement. Materials choices have been made based on industrially available materials to meet the challenges of ITER construction and operating conditions.
2. The Parties' domestic reviews of the ITER DDR support the views that:
   - the physics basis of the design is sound;
   - the engineering is coherent, feasible and consistent with satisfying the ITER programmatic and detailed technical objectives;
   - the safety and environmental characteristics of the design are soundly based.
3. ITER stimulates impressive world-wide progress in fusion engineering, from concept definition to assembly planning. Focussing effort in the different areas of fusion technology to meet the demands of a single device provides concrete challenges to engineers and leads to practicable solutions for the different components/systems now being validated as models or prototypes. Whilst the engineering follows a conservative philosophy, the unprecedented requirements of ITER have extended the capabilities of the various technologies.
4. The programme of technology R&D – the seven large projects and other supporting tasks – is expected to validate the key aspects of the ITER design, including development and qualification of the applicable technologies/materials for ITER conditions, and development and verification of industrial techniques in manufacturing component prototypes, with related QA. It will provide a substantial industrial database for cost estimates. The projects have also pioneered efficient modes of international collaboration in fusion technologies which could be possible precursors for a joint construction of ITER and which provide a valuable asset to any possible future collaborations in fusion development.
5. The next milestone report – the Final Design Report – to be presented in February 1998 will be based on detailed sets of design description documents covering each ITER system with the necessary data on design, specifications, interfaces, supporting R&D documents and safety/environmental data. The industrial costing studies of procurement packages are expected to provide a firm basis for the cost estimates.
6. The motives for undertaking ITER in a framework of international collaboration remain valid. The experience with ITER has shown such a framework to be robust, efficient and beneficial to all participants. It is therefore reasonable to conclude that ITER should proceed in an international collaborative framework towards realising the overall aims of the Project through its construction and operation.
7. To this end, the Parties need to resolve the key issues of siting and regulatory clearance, cost sharing and procurement arrangements, and establishing the legal framework and organisation appropriate to a global venture of ITER's size and technical demands. While such strategic considerations are in progress in the Parties' domestic systems and in top level joint interactions, it is also important 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 applications for licences to build and operate ITER;
   - extending the ongoing supporting R&D;
   - finalising the procurement specifications and related documentation for major ITER systems;
   - consolidating the scientific basis of ITER operations.
8. 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 broadening of capability that comes from exposure to different approaches to project organisation and management. It has proved to be an efficient vehicle for the kind of fusion engineering and materials developments needed to realise any concepts for practical magnetic fusion reactors. Bringing ITER to full realisation through joint construction and operations will continue this process. The time has come to prepare for taking this next step together.

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