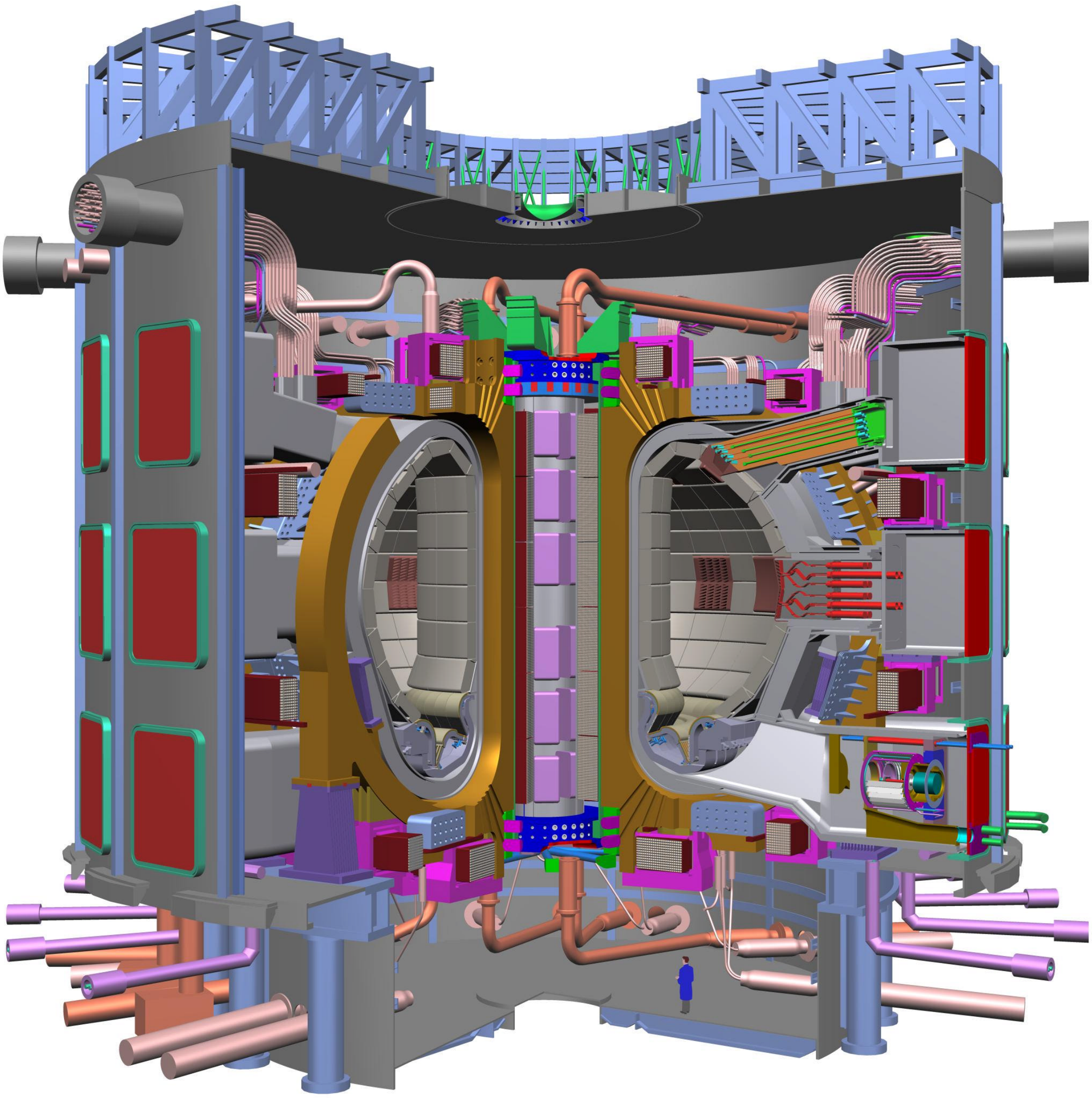


ITER
The Design Phase

Historical Background

ITER (meaning "the way" in Latin) is an international collaborative project 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.

Since 1958 a remarkable degree of openness and global co-operation in the development of nuclear fusion has brought with it dramatic progress in scientific understanding and performance achievement. The leading fusion experiments today are exploring the fusion domain around the threshold of break-even conditions, when thermal power from fusion reactions begins to exceed the thermal power supplied externally to keep the plasma hot. Meanwhile, smaller supporting experiments and theoretical developments are together broadening scientific understanding and establishing competence in fusion technologies.

The logical next step is now to study, in an integrated way, the physics of plasmas fusing deuterium and tritium to produce helium and neutrons at power-reactor scale, and to establish the key technologies that will make fusion a practical energy source. ITER not only is designed to fulfil this role, but it is the key element in the strategy to build the first electricity-generating fusion power plant as the subsequent experimental step.

The ITER project arose from the recognition, by the world's leading fusion programmes, of the benefit of undertaking the next step jointly. Collaboration on ITER offered the opportunity to pool the scientific experience and technological expertise of all the world's leading fusion experiments and programmes.

At the Geneva Superpower Summit Meeting in November 1985, a proposal was made by the then Soviet Union to build a next generation tokamak experiment on a collaborative basis involving the world's four major fusion programmes, in Europe, Japan, the Soviet Union, and the United States of America (the "Parties"). This led to the establishment of a collaboration under the auspices of the International Atomic Energy Agency (IAEA), and the start of Conceptual Design Activities (CDA) for ITER in April 1988 which were successfully completed in December 1990.

The CDA helped to bring about a convergence of the Parties' views on the overall programmatic and technical objectives for a next step machine, and gave them confidence that it could be achieved through international collaboration. Common understandings were reached on the choice of the tokamak confinement concept and on the technology R&D that needed to be carried out.

In July 1992, the four Parties (now with the Russian Federation replacing the Soviet Union) entered into an intergovernmental agreement to begin the Engineering Design Activities (EDA) of ITER. Canada and Kazakhstan became involved in the Project by association with Euratom and the Russian Federation respectively.

The EDA was defined initially for a six-year period during which the Parties agreed jointly (and on a basis of equality) to produce a detailed, complete and fully integrated engineering design and all technical data necessary for future decisions on the construction of ITER. Six years of international collaborative work culminated in the approval by the ITER Council in June 1998 of the ITER Final Design Report, Cost Review and Safety Analysis. This report provided the first comprehensive design of a fusion reactor based on well-established physics and technology.

At that time, due to financial constraints, the Parties decided to call for an investigation of whether, with a set of reduced technical objectives and margins, and at a cost of ~ 50% of the direct capital cost of the then current design, the same programmatic and strategic objectives could be reached. The EDA was extended by 3-years to allow the details of such a design to be worked out, and to undertake other work aimed at enabling a possible future construction decision. The US committed itself unilaterally to one year only, and withdrew in 1999. The revised design was finalised, documented in the ITER Final Design Report (FDR), and approved by the remaining Parties at the end of the EDA in July 2001. The technical basis was established on which to start the ITER construction phase.

Negotiations

Quadripartite meetings on Negotiations on the Joint Implementation of ITER began in mid-2001. The initial "Participants" were Euratom, Japan and the Russian Federation, plus Canada, which made a government-backed site offer. In February 2003 they were joined by the People's Republic of China and the United States of America, and in May 2003 by the Republic of Korea. The Negotiators should draft the ITER Joint Implementation Agreement on Construction, Operation and Decommissioning, examine proposals for the ITER construction site (there are four proposals - Cadarache, Clarington, Rokkasho-mura, and Vandellòs), agree on the procurement rules and management, who will provide the various ITER components/systems and how the costs will be shared, and identify the Director General for the ITER Legal Entity (ILE), and the organisation for its work.

The Negotiators were supported on technical aspects by Coordinated Technical Activities (CTA), between July 2001 and the end of 2002, and subsequently are supported by ITER Transitional Arrangements (ITA) in the run-up to construction. These maintain the integrity of the project so as to prepare for joint construction and operation. The work of the Participant and International Teams during the CTA/ITA involves preparation for an efficient start of construction, including design adaptations to potential sites and their regulatory environment, and formal review and modification to ensure design completeness, preparation of licensing applications by close dialogue with potential host regulators, exploitation of manufacturing R&D, and of physics R&D to take advantage of latest experimental results, and preparation of technical specifications for procurements which need to be launched early.

The timescale for the Negotiations foresees that the Joint Implementation Agreement should be initialled during 2003. Formal signature (and/or ratification) should take place in early 2004, leading to the establishment of the ITER International Fusion Energy Organisation (IIFEO), the organisation which will build ITER, shortly thereafter.

ITER People

Over the years of the EDA, a large number of scientists and engineers from Europe, Japan, Russia and the United States worked together on this unprecedented international collaboration (see attachments). The Director, reporting to the ITER Council (itself supported by Management and Technical Advisory Committees), led a Joint Central Team of approximately 150 professional staff formed by the Parties to develop and coordinate the design, define R&D tasks carried out by "Home Teams" of each Party (the Home Team is defined to include all those people and organisations working on ITER tasks) and credited to its contribution. The Joint Central Team was located at three (later two) Joint Work Sites at Garching, Naka, and formerly in San Diego. The Home Team tasks involved companies and organisations distributed throughout the territories of the participants. Over the EDA, the Joint Central Team and the Home Teams dedicated nearly 2000 professional person years of effort, and the Home Teams spent $660M (1989 values) on supporting R&D. The total cost of the design phase has therefore been in the region of $1B.

Performance

ITER aims to produce a 400 s inductively driven plasma burn at a fusion power level of 500 MW, and to extend the burn ultimately to steady state through the use of non-inductive current drive by neutral particle beams and microwaves. The ITER plasma will be a power amplifier - the ratio of thermal power released from the plasma to the thermal power input to the plasma (Q) will be at least 5-10. Heat loads on divertor plasma-facing components will be comparable to those in a reactor generating electrical power, but the neutron wall loadings will be about 30%, and the neutron dose to bulk materials about 3%, of those such a reactor would experience. Reactor nuclear component integration testing (in particular, "test blankets") will therefore be possible on ITER, but endurance testing and qualification of materials, to give confidence in operational reliability of the subsequent power reactor, will require a separate materials test facility (e.g.IFMIF[1]) to be operated in parallel.

To provide a sound basis for this performance, the ITER Physics Basis[2] was assembled, by experimenters in the fusion programme worldwide, to bring together in one place all the relevant data, and to build a consensus on its interpretation. The most promising inductively driven operation regime was concluded to be "ELMY H-mode", Here the plasma, constrained by a poloidal divertor "separatrix" (i.e. the last closed magnetic flux surface surrounding the plasma), forms a "transport barrier" just inside the separatrix, the H-mode, which slows down thermal conduction outwards. This barrier undergoes minor oscillations - edge-localised modes (ELMs) - but can be maintained indefinitely provided the power lost (P_{loss}) across the separatrix exceeds a certain value, scaling with plasma parameters such as field, density, and size. The amount of external heating needed to compensate this loss depends on the plasma energy confinement time, which also scales with such parameters, and with plasma current and geometry.

The ITER Physics Basis established the most probable form of these scalings, allowing them to be used to extrapolate to ITER dimensions. Further checks have been made by normalising experimental results according to dimensionless parameters - ρ^*, the ratio of the radius of gyration of plasma ions round field lines divided by the plasma minor radius, β, the ratio of plasma thermal pressure to magnetic pressure, and ν , the collisionality i.e. the ratio of field line connection length to trapped particle mean free path. Using today's experiments, conditions in the plasma core of ITER can be simulated precisely, with the exception of ρ^*, which will be five times smaller in ITER.

Based on the above scalings, and corroborated by the dimensionless analysis, the nominal design parameters for ITER were derived: plasma current 15 MA, major radius 6.2 m, minor radius 2.0 m, on-axis toroidal field 5.3T, plasma elongation at separatrix 1.85. Given the many dependent parameters, and the uncertainties in extrapolation still remaining, a wide range of plasma operation is foreseen, to be sure of meeting the objectives. For example, a typical range for inductively driven plasma operation is shown in Figure 1, which shows achievable power amplification, as plasma current and external heating power needed to establish the H-mode are varied within limits, for a range of assumptions about helium confinement.

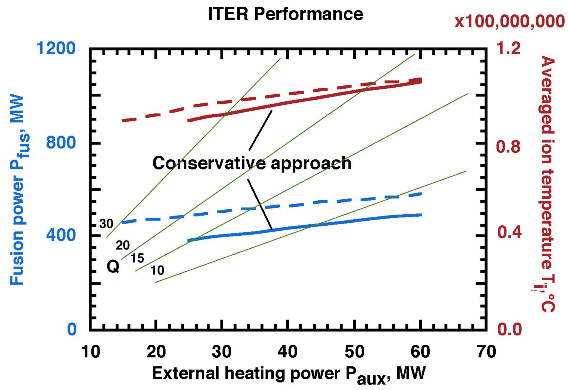


Figure 1 - Inductive performance for nominal parameters

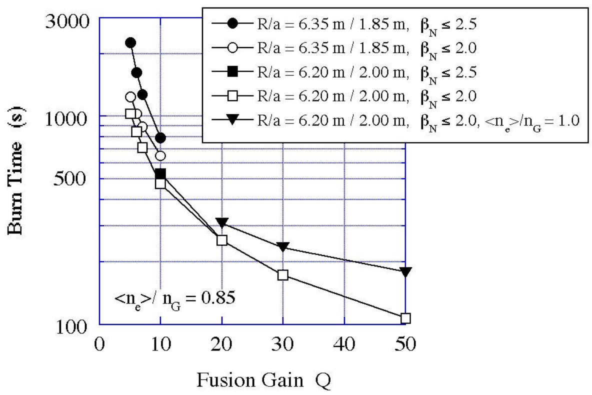


Figure 2 - Non-inductive performance (n_e/n_G=0.85)

1 see http://insdell.tokai.jaeri.go.jp/IFMIFHOME/ifmif_home_e.html

2 Nuclear Fusion 39, 12, 2137-2664, December 1999

Stretching the burn time should make the steam-raising tritium-breeding test blankets on ITER increasingly relevant, since a power reactor would ideally operate with very long pulses, at least of order hours, compared to the ~7 minutes of nominal ITER inductive operation. Typical performance projections of ITER, as inductive current drive is increasingly replaced by non-inductive, so-called hybrid operation, stretching the burn time and ultimately leading to steady state operation, is shown in Figure 2. The burn time can be stretched as current drive power is increased, with a corresponding decrease in "fusion gain" (Q). There is some advantage in reducing plasma minor radius and displacing the plasma major axis outwards. Typical fusion power output over the range shown varies from 400-700 MW.

The range of performance shown above follows the predictions and limits (e.g. in plasma density) of the bulk of today's experiments. The pressure increases sharply at the H-mode transport barrier, establishing "pedestal" values which in some experiments constrain the internal radial temperature and density profiles. The energy contained in the plasma due to the pedestal is typically 1/3 of the total plasma energy, and may scale differently with plasma parameters. ELMs appear as a pseudo-periodic relaxation of the pressure gradient at the plasma boundary, due to an instability dependent on the detailed shape of the magnetic surfaces near the separatrix. If the ELM frequency drops, the amplitude increases and the energy removed from the pedestal per ELM becomes larger, increasing divertor erosion. In some experiments, if the power entering the plasma core is sufficiently high, a second transport barrier occurs inside, limiting even more the heat conduction across the plasma. If such a barrier can be controlled, it would lead to improved confinement. Furthermore, the "bootstrap current", a part of the plasma current driven by the plasma itself, would be enhanced by such an internal barrier, reducing the need for external non-inductive drive systems, and enhancing operation beyond that shown in Figure 2.

Because fusion power scales with $\beta^2 B^4$, there is a strong incentive to operate at the highest β allowed by plasma stability. However, numerous experiments show that certain magnetic surfaces spread increasingly volumetrically and toroidally asymmetrically as β is increased. The resulting magnetic "islands" would cause plasma disruptions (rapid termination of the plasma burn) if not controlled. The onset of these "neoclassical tearing modes" can fortunately be controlled by locally driving additional plasma current using electron cyclotron waves, and this method will be employed on ITER using predominantly the upper ports. "Error" fields, caused by imperfections in the machine assembly or in the detailed manufacture of the magnets or of ferromagnetic components, can also cause magnetic islands, but it is planned to eliminate these using the correction coils, which produce a small amplitude controlled helical field.

Plasma disruptions result in large heat flows along field lines to the surrounding material surfaces, followed by rapid termination of plasma current, which can cause "runaway electrons" and further wall erosion. If repeated often enough at the same location, the damage caused can require a refit of components at that location. Although ITER in-vessel components are designed for a certain number of disruptions, they are clearly to be avoided whenever possible.

Magnetic surfaces outside the separatrix are "open", i.e. cross the surrounding material surfaces. Due to the largely toroidal orientation of these open surfaces and their field lines, particles scattered into this "scrape-off layer" travel a considerable distance before striking a material surface at a very oblique angle. Nevertheless, the power flow in the scrape-off layer is large, and it would be beyond the limit of heat removal, typically in the range of 10 MW/m^2, unless additional measures were taken. These are the enhancement of radiation loss by impurity injection, and from the ionisation of a large neutral particle density built in front of the divertor target plates. This allows "partially detached" plasma operation. Plasma pressure decreases but plasma density increases significantly towards the target, so plasma temperature at the target is very low and power can be brought to acceptable levels. Essentially the power is dissipated as radiation and by the impact of "charge exchange" neutral particles on the divertor side walls. Impurities entering the plasma in the divertor region are stopped from entering the plasma by fuel flow towards the target. Divertor operation also controls the removal of helium from the plasma, which would otherwise gradually poison the reaction volume.

To ensure the helium from fusion reactions is largely thermalised inside the plasma, ferromagnetic inserts are placed in the shadow of the toroidal field coils outboard of the plasma to reduce field ripple caused by the discreteness of the toroidal field coils, which would bring high energy particles into contact with the walls.

The divertor, along with fuelling by pellets which allow the establishment of a favourable centrally peaked density profile, is therefore used in ITER to control plasma fuel density. Nevertheless, it is generally observed that average plasma electron density is limited to the Greenwald density n_G (10^{20} m^{-3}) = I(MA)/πa^2, and hence the limit to operation shown in the figures.

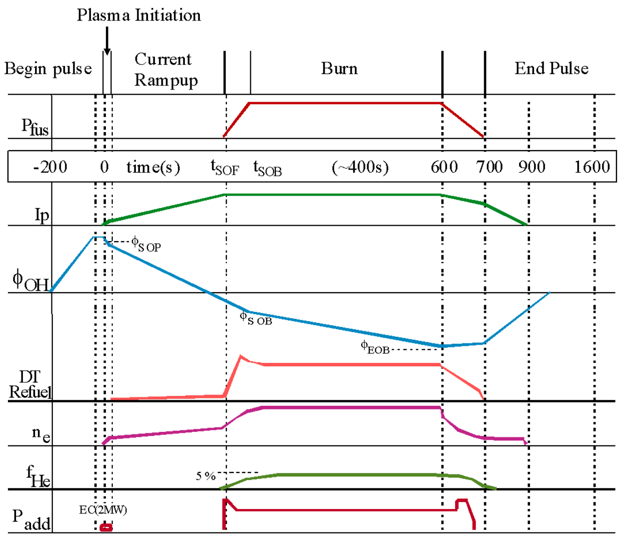


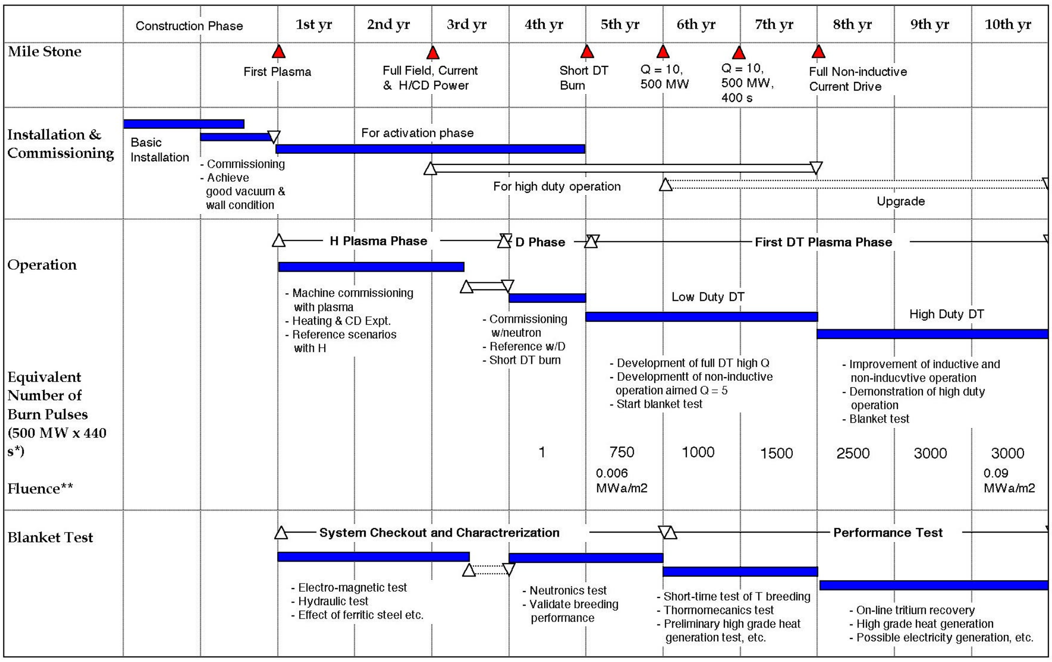
Figure 3 - Typical Inductive Plasma Pulse

A typical plasma pulse (Figure 3) is initiated by a sharp change in central solenoid and poloidal field coil currents.

This creates a toroidal loop voltage (~ 12 V) in the plasma chamber, initiating gas ionisation and discharge, which is facilitated by a short burst (2 s) of electron cyclotron heating. The plasma current is then ramped up and the divertor field configuration established, over the subsequent 100 s. External heating is applied and used with fuelling and pumping to bring the power level to 500 MW after a further 50 s. During the whole period, including the subsequent burn and ramp-down, the plasma shape, position, and configuration are controlled by the poloidal field coils. The pulse is terminated by reducing the fuelling and heating, and reducing the plasma current in a controlled way to avoid plasma disruptions.

The ITER Physics Basis resulted from the strong collaborative effort during the EDA organised through the ITER Physics Committee, which directed experimental efforts worldwide to solving issues raised by planned ITER performance. Plasma physics experiments and modelling continue worldwide and feed into the ITER design via the successor organisation, the ITER Tokamak Physics Activity (ITPA).

ITER will be the first controlled fusion experiment dominated by internal plasma heating through the fusion reactions themselves, rather than external heating, thereby breaking new ground. From what is thought today to be important for ITER, of particular interest will be the influence of ELMS on divertor life, the effect of further internal transport barriers, the influence of magnetohydrodynamic instabilities caused by the plasma configuration or by error fields introduced by machine imperfections, and the impact/control of plasma disruptions (rapid termination of the plasma burn). These interests shape the initial operational plans (Figure 4).



* The burn time of 440 s includes 400 s flat top and equivalent time which additional flux is counted during ramp-up and ramp-down.
** Average Fluence at First Wall (Neutron wall load is 0.56 MW/m2 in average and 0.77MW/m2 at outboard midplane.)

Figure 4 - Operation Schedule for the first 10 years

Design

The main design features of the ITER tokamak and site are shown in Figures 5 to 10. Plasma linear dimensions are twice those of the largest experiment today. ITER's superconducting TF magnet consists of 18 D-shaped coils containing circular cross-section conductor, composed of Nb_3Sn strands, embedded in grooved radial plates. The central solenoid (CS) uses square cross-section Nb_3Sn conductor and has six modules which can be powered separately. The six poloidal field (PF) coils are made using Nb-Ti conductor in double pancakes. The lower PF coils are designed with redundant turns and a margin in current to avoid the need to replace the coils in case of local damage in one of the coil pancakes. To accommodate field errors due to manufacturing inaccuracies or to misalignments during assembly of the magnet coils, as well as to control resistive wall mode plasma instabilities, superconducting saddle-shaped correction coils are placed around the machine outside the TF magnets.

When energised, the TF coils press together along their straight sections, forming a vault. The coils are encased to aid their support and to transfer loads across keys between the cases. The poloidal field crossing the TF coils creates overturning moments and circumferential torques on each TF coil. A shell-like structure between the coils, to the extent port penetrations allow, permits these forces to be reacted within the magnet structure, and provides a strong support for the poloidal field coils.

The reaction chamber consists of a vacuum vessel supporting remotely exchangeable modular in-vessel components. The vacuum vessel consists of 9 toroidal sectors, joined by field welds. The vessel is a double-walled stainless steel welded ribbed shell, with internal shield plates and ferromagnetic inserts to reduce toroidal field ripple. The 421 blanket modules have a single-curvature faceted separate first wall attached to a shielding block which is remotely attached to the vessel through 3 cm diameter access holes in the first wall. To accommodate differential thermal expansion and electromagnetic loads, these attachments are stiff radially, but flexible transversely. The plasma-facing components are beryllium armour attached to a copper substrate, mounted on a water-cooled stainless steel support. The outboard modules may later be replaced with tritium-breeding modules. The 54-cassette single null divertor has carbon targets and tungsten high heat flux components, again mounted on a copper substrate, and water-cooled stainless steel structure bolted to rails on the vessel floor. The targets can accommodate heat loads of more than 20 MW/m^2 for 20 s, but the more normal peak heat load will be 5 - 10 MW/m^2.

Six of the 17 accessible vessel equatorial port plugs are used for heating antennae and neutral beam ducts, three are used for power reactor test blankets, two for plasma limiters, and the remainder for plasma diagnostics. The limiter and two diagnostic ports are also used for remote blanket module replacement. 9 divertor ports accommodate eight 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, which are inserted radially and then slid toroidally and clamped to rails. The 18 upper ports are mainly used for diagnostics. Three contain electron cyclotron antennas to control plasma instabilities (neo-classical tearing modes).

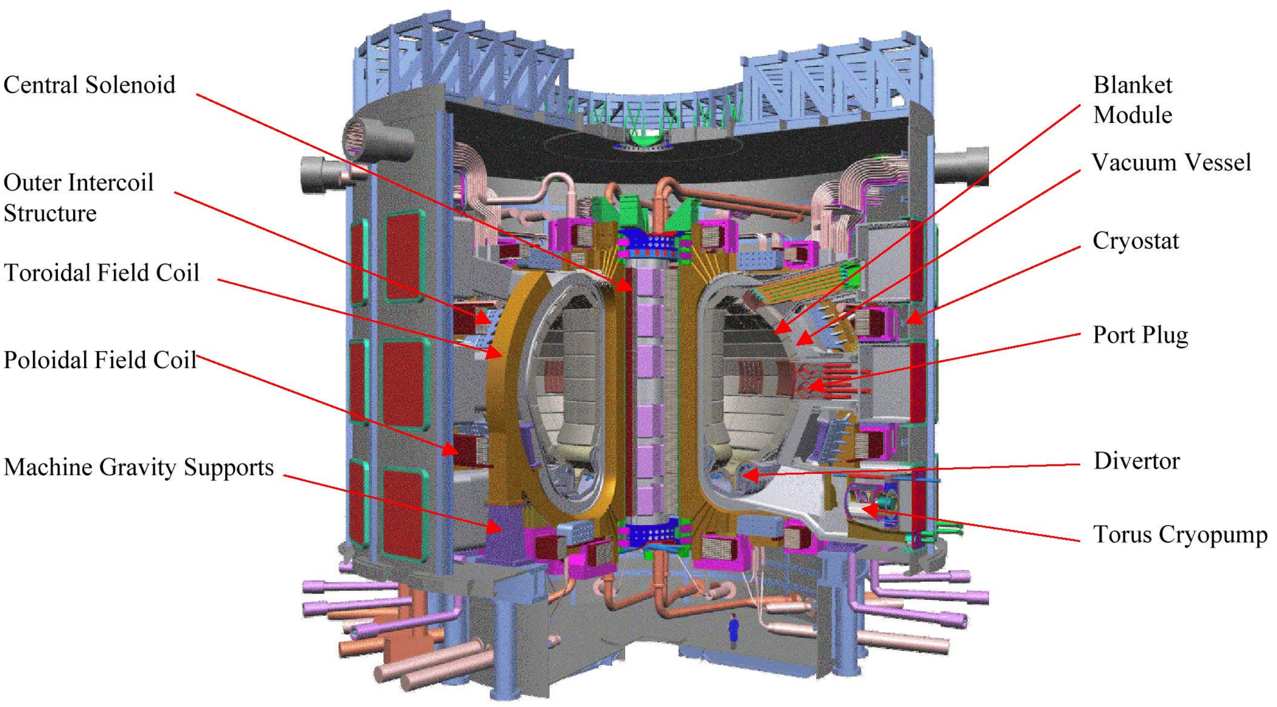


Figure 5 - Cutaway of ITER Tokamak inside the Cryostat/Biological Shield

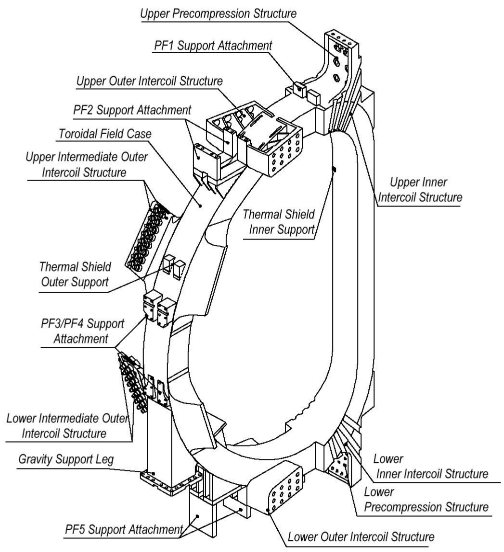


Figure 6 - TF Coil and Structure

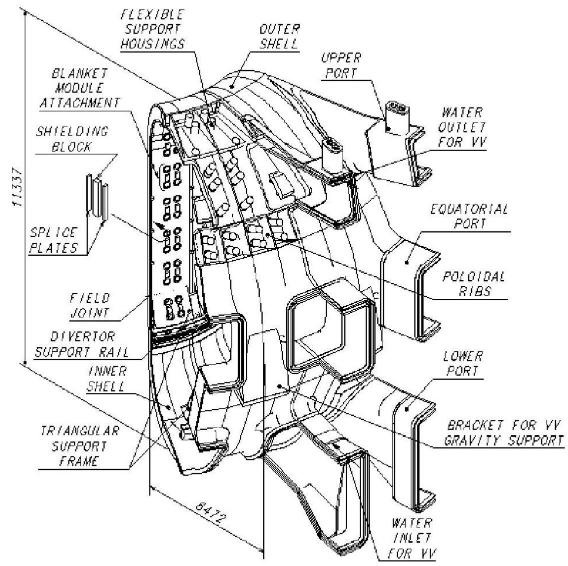


Figure 7 - Main Vessel

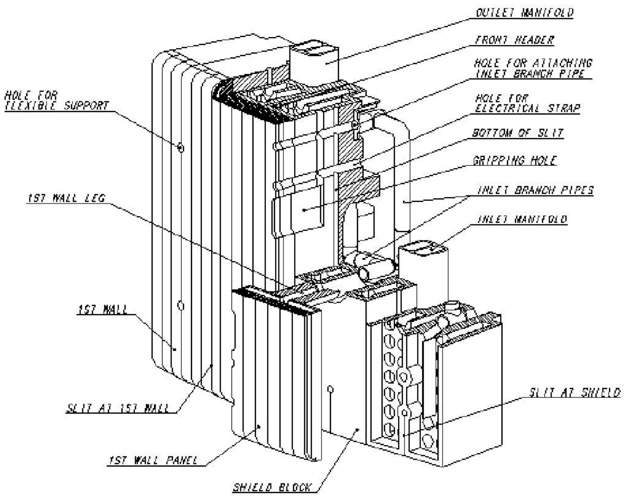


Figure 8 - Blanket Module

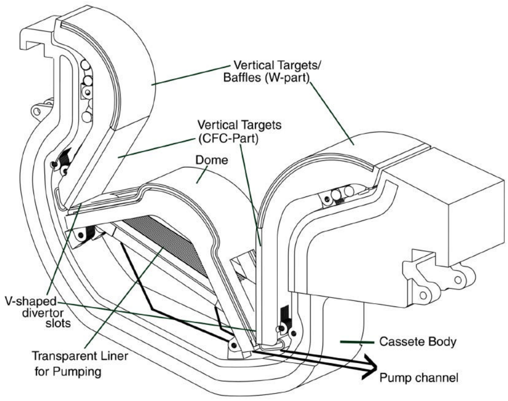


Figure 9 - Divertor Cassette

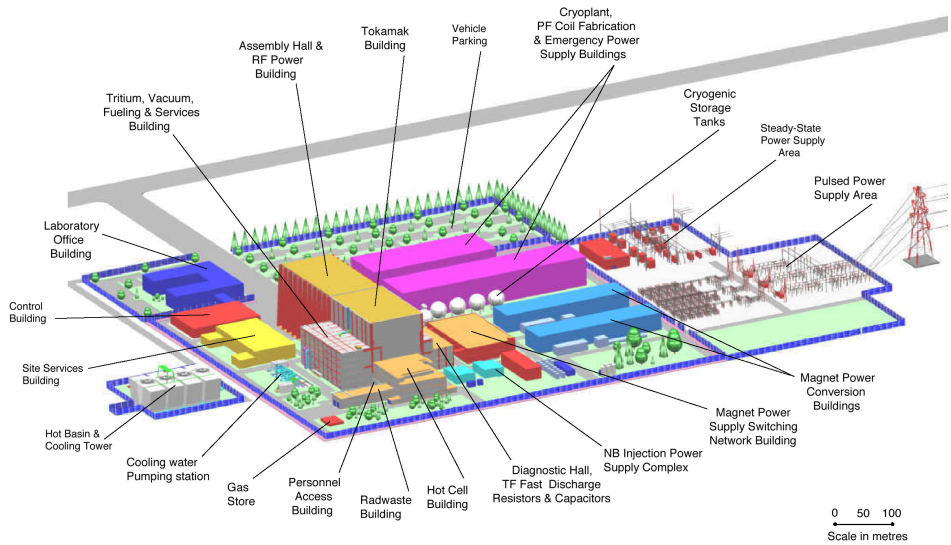


Figure 10 - ITER Generic Site Layout

A cryostat surrounds the coils. It is essentially a reinforced single-shell cylinder 24 m high and 28 m diameter. Shielding thicknesses are arranged to permit personnel access at the port terminations or, exceptionally, for repairs in the cryostat-coil interspace, after shutdown. To reduce heat inleak to the coils from radiation from surrounding warm surfaces, thermal shields are used between the vacuum vessel and the toroidal field coils.

The tokamak is water-cooled by separate circuits feeding the blanket (3 circuits in parallel), divertor and limiter (1 circuit), and vacuum vessel (2 circuits in parallel). The vessel cooling circuit alone can remove, by natural convection, all decay heat after shutdown in all vessel and in-vessel components. Typical water inlet temperature is 100°C, and pressures are in the range of 3-4.2 MPa. Baking of in-vessel components to remove adsorbed impurities is carried out at 240°C (200°C for the vessel).

The plasma is heated (and current may be driven) by a combination of electron cyclotron, ion cyclotron, lower hybrid and 1 MeV negative-ion-accelerated neutral beam systems. The initial setup will involve two neutral beams and electron and ion cyclotron systems, but the radio-frequency systems are designed in exchangeable modular units (20 MW/port) to allow various mixes to be tried, and three neutral beams can be accommodated on the machine. A heating power in excess of 110 MW is thus attainable.

ITER is assembled inside a cylindrical "pit" embedded up to the equatorial port level. After installation of the lower cryostat, PF coils and supports, 40° sectors of the vacuum vessel are combined with two TF coils and appropriate thermal shielding, and welded to adjacent sectors in the pit. The upper coils, ports and services are connected, and the cryostat is closed by a flat lid with heavy segmented shielding.

Cost and Schedule

The joint implementation of ITER foresees three main types of procurements for the project: those provided by the host (e.g. building and basic site infrastructure), high technology content items to be provided by each ITER Party "in kind", and remaining shared items to be procured centrally from a joint fund. To ensure each Party contributes a predefined share of these items, the Parties must agree beforehand which ones each will contribute. To do this, all Parties must agree on the value of each item to the project. For items provided "in kind", using the purchasing procedures and funding arrangements each Party prefers, the actual costs may not correspond to the project valuation - it may differ due to competitive tendering as well as different unit costs locally.

To provide such a valuation, ITER construction was broken down into about 85 "procurement packages" each defined at the level of detail appropriate for an actual procurement contract. Industrial companies or large laboratories with relevant experience were invited by the Parties to analyse the manufacturing processes and to generate estimates of the physical quantities to be supplied, including manpower, materials, and tooling, and other costs pertinent to their normal industrial practices, to fit a given delivery schedule. After consolidation across the Parties, the resulting physical quantities were multiplied by a single set of labour rates (depending on speciality and type of labour) and material unit costs across all packages, independent of Party.

The resulting valuation of ITER is shown in Figure 11. The total size of the “cake” is $3B in 1989 US $. With management staffing and R&D during construction, 20 years of operation, and decommissioning, total project costs are estimated to be $7.5B (1989 US $) spread over 35 years.

The construction schedule of ITER is shown in Figure 12. In a success-orientated schedule, construction takes 7 years from the granting of the construction license to the start of integrated machine commissioning, with the first tokamak plasma planned one year later. The schedule is constrained by the procurement rate for the magnets and vacuum vessel, as well as the buildings which will house the tokamak and those which will be used to wind large poloidal field coils on site. Some purchase orders can be released before the construction license is granted, where the procurements are not critical for licensing.

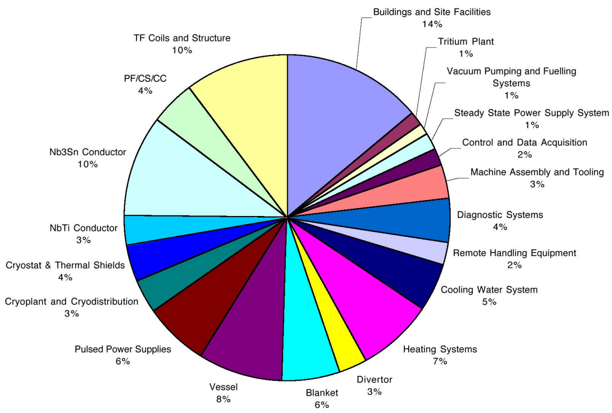


Figure 11 - ITER Hardware Cost Valuation

Preparations during the ITER Transitional Arrangements aim to have the procurement packages for long lead items ready to issue for bidding purposes as soon as the ITER Legal Entity (ILE = IIFEO) is established.

Safety

One of the main ITER goals is to demonstrate the safety and environmental advantages of fusion power, namely 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. Extensive design assessments have thus been carried out to confirm the safety and environmental acceptability of ITER and to ensure that ITER can be sited in the territory of any of the Parties with only minor changes to accommodate or take advantage of site-specific features.

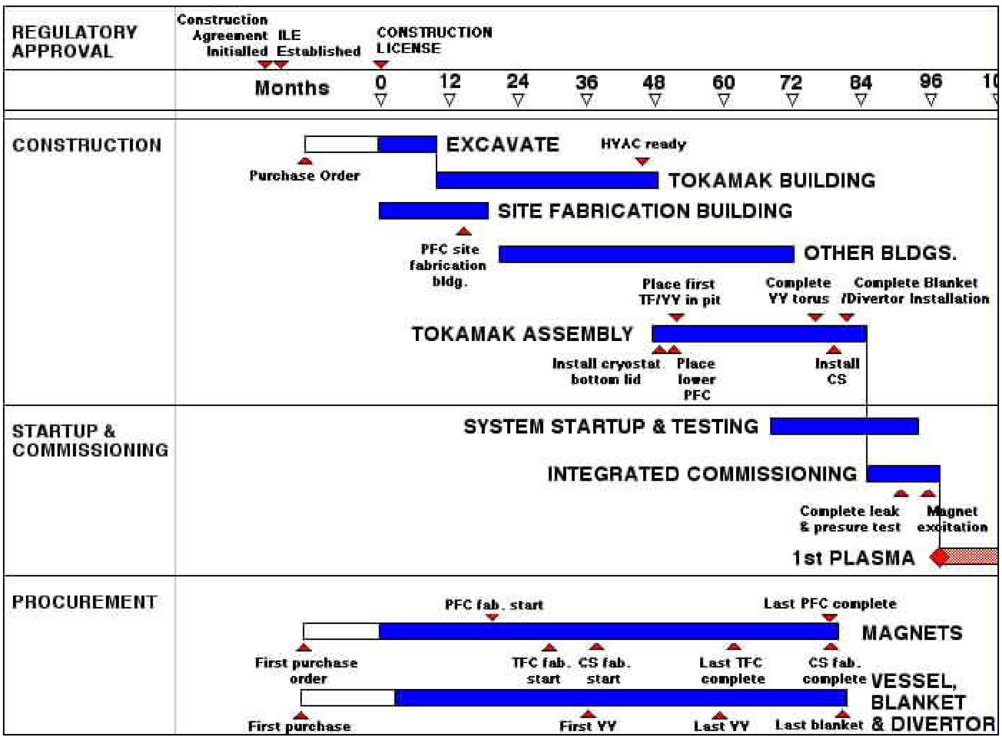


Figure 12 - ITER Construction Schedule

ITER incorporates many features that ensure that the environmental impact during normal operation will be insignificant, including inherent confinement barriers - the plasma vacuum vessel, cryostat and elements of the building design - which prevent releases, as well as air/water detritiation and filtration systems to treat any releases that may occur. In a comprehensive and conservative analysis, sources of potential releases were identified, release pathways determined, and design features and release control systems assessed. Potential 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.

A similarly comprehensive and conservative analysis of off-normal events has also been performed. This analysis thoroughly examined possible ways for tritium, activated corrosion products in coolants, and neutron-activated tokamak dust, to be released to the environment, and selected the most severe incidents from the range of those possible. Radioactive releases for all such events are well below the project release guidelines that would lead to additional doses (to the most exposed individual) comparable to the average annual natural background exposure for a generic site.

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 combination of events from internal origin, 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.

At the end of ITER's operating life, due to decay and decontamination, a significant fraction of activated material, increasing with time, can be cleared, according to IAEA regulations, for release from regulatory control, allowing unrestricted re-use. The present assumption is that radioactive material not below the clearance level after 100 years is "waste", requiring disposal in a repository. Estimates of material masses show that about 30,000 t of material will be radioactive at shutdown, and that 80% of it can be cleared within 100 years.

Assessments of system maintenance demonstrate that ITER will already maintain occupational exposures below the project guidelines, and further reductions continue to be made through design improvements, as part of the general project policy to maintain doses as low as reasonably achievable (ALARA).

Design Feasibility

Beyond the extrapolation still in plasma physics, the major technical challenges of building and operating ITER are the large size of the superconducting magnets, structures and reaction chamber, the high heat and neutron fluxes on plasma-facing in-vessel components, and the ability to rapidly and remotely repair and maintain in-vessel components. The overall philosophy of the ITER design has been to use advanced but proven technological approaches, verifying their application to ITER through detailed analysis, and validating them through technology R&D. The bulk of this R&D expenditure was therefore deployed on seven large R&D projects to develop and demonstrate the key technologies needed. These projects:

- were typically multi-stage activities involving multiple Party contributions and cross-dependencies, and high industrial content;
- each had a unified management structure and organization in which Project responsibility was shared between Central and Home Teams.

Such activities included development and qualification of the applicable technologies by testing at different scales, development and verification of industrial techniques to be used for component prototypes manufacturing, and definition and verification of the comprehensive quality control and quality assurance programmes. The hardware of these large projects was produced and assembled, and the testing programmes of these prototypes - to determine their operating margins in performance, to optimise their flexibility in operation, and sometimes to train their future operators - are now completed. In some cases the hardware is still in use for further developments.

The technical output from the Seven Large R&D Projects has directly supported the manufacturing cost estimates for key ITER cost drivers. The Projects also have been prototypes for cross-Party complex ventures, demonstrating on a smaller scale the management and procedures needed for joint construction of ITER.

Apart from the confidence given by the successful outcomes of these projects, some major and innumerable minor spin-offs have arisen from the work. Major spin-offs include the development of new high field conductors with application to enhanced magnetic-resonance imaging, the demonstrated ability to manufacture large complex vacuum-tight structures to high accuracy and to join them remotely (space-applications), as well as the development of special high heat flux composites (aerospace applications).

The main objectives and achievements of each large project are described below.

L1. Central Solenoid Model Coil

The technology required to build the ITER Central Solenoid (CS) required a significant advance in superconducting coil manufacturing technology. The main manufacturing issues were addressed in the CS model coil conductor, and were all successfully solved. These included the production of a substantial quantity of Nb_3Sn strand to a uniform quality, and the jacketing of a cable of this strand using a heavy square section of superalloy "Incoloy 908" to provide structural support against magnetic forces. The conductor was accurately bent to the winding shape, heat treated in a controlled atmosphere, and insulated in an "unspringing" process, before stacking to form the winding and impregnating with epoxy resin. Finally, the different layers were connected by joints on the top and bottom of the coil.

The CS Model Coil consists of two modules nested inside each other. The Inner Module (Figure 13, US) has an inner diameter of 1.8 m, outer diameter of 2.7 m and height of 2.8 m. The Outer Module (Figure 14, JA) has an inner diameter of 2.7 m, outer diameter of 3.6 m and height of 2.8 m. Additional coils may be inserted and tested within the bore of the main coil, with conductor length ~80 m and a bending radius of about 70 cm, and their own power supply and instrumentation. The model coil has been used to test the CS Nb_3Sn conductor, an alternative in NbAl, and the TF Nb_3Sn conductor. Strand and conductor for the modules and inserts were also provided by the EU and RF.

The installation of the CS Model Coil and CS Insert was completed in October, 1999. A maximum field of 13.5T, maximum current of 46kA and stored energy of 640MJ (the highest stored energy in any Nb_3Sn coil) was achieved in April 2000. Ramp-up rates of 1.2T/s and rampdown rates of -1.5T/s were achieved in insert coils, well above the respective goals of 0.4 and - 1.2T/s respectively, and a 10,000 cycle test was completed in August 2000 (ITER will typically experience 30,000 cycles). Tests on the NbAl insert coil were successfully completed in March 2002.

Figure 13 - Inner Coil Module

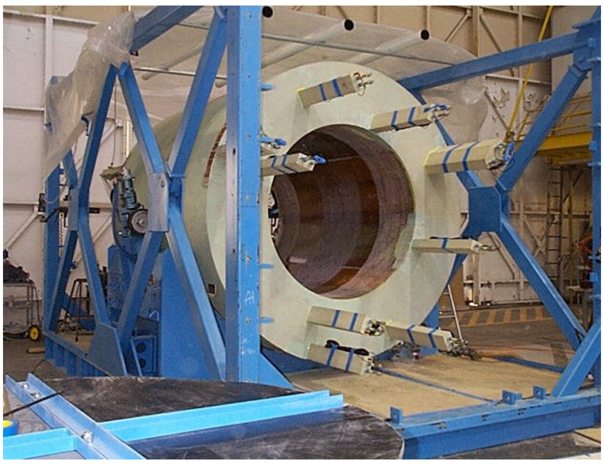


Figure 14 - Outer Coil Module

Apart from testing of the insert coils, operation of the main coil verified the insulation performance under load, and demonstrated the integrated performance of joints and conductor for several different conductor lengths and two different joint concepts, confirming that manufacturing variability can be controlled.

L2. Toroidal Field Model Coil

The objective of the Toroidal Field Model Coil Project (led by the EU) was to develop and demonstrate the superconducting magnet technology to a level that would allow the ITER TF coils to be built with confidence. It allowed design and analysis to be validated, industrial manufacturing methods to be demonstrated, confirmed the performance of each component integrated in the magnet, and tested and demonstrated reliable operation.

The project was aimed at defining the critical steps in this process by the manufacture of a subsize coil, about 4m high and 3m wide, and two full-size sections of the outer housing, and includes the key technical features and manufacturing approaches foreseen for the actual ITER TF coils. For example the insulation of the reacted conductor is shown in Figure 15. Because only a single coil was made, the conductor cannot be fully tested for superconducting properties (this was done instead by testing a TF insert coil in the CS model coil), and the manufacturing was done to define appropriate tolerance targets, procedures and quality control steps. The test of the subsize coil creates realistic magnetic loads to demonstrate the structural concept.

The assembly of the winding pack was completed in 1999. The impregnation of the individual radial plates with an outer insulation wrap was finished in July 2000 and the outer joints between pancakes were electron beam welded in August 2000.

When the winding pack was put into the case and closure welded, problems were found initially with the weld quality, but this was resolved by the supplier. Vacuum impregnation of the winding pack-case gap (pre-filled with coated sand granules) was completed in late 2000, and the finished coil delivered to the TOSKA facility at Karlsruhe, Germany, which had been adapted to accommodate the coil and its test programme.

Assembly and checking of the coil in TOSKA (Figure 16) was completed in June 2001. The coil went superconducting in July 2001. The coil was ramped to 80kA (above the value needed in ITER and the maximum possible in the facility) with a field of 8T. This is the highest current ever driven in a superconducting coil. All joints and superconductor behaved as expected, as did temperature increases during fast ramp-down and safety discharge. Temperature margins were found (also in the CS Model Coil) to be somewhat lower than expected due to strand strain sensitivity, and this has led to a tightening of the strand procurement specification (higher current density) and/or the use of less contractive jacket material.

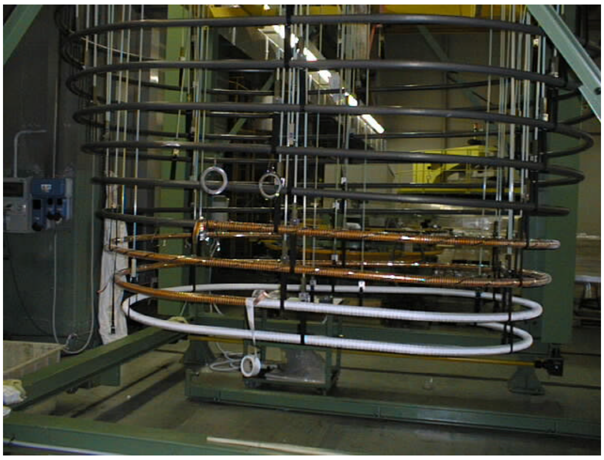


Figure 15 - Wrapping Insulation on the TF Model Coil Conductor

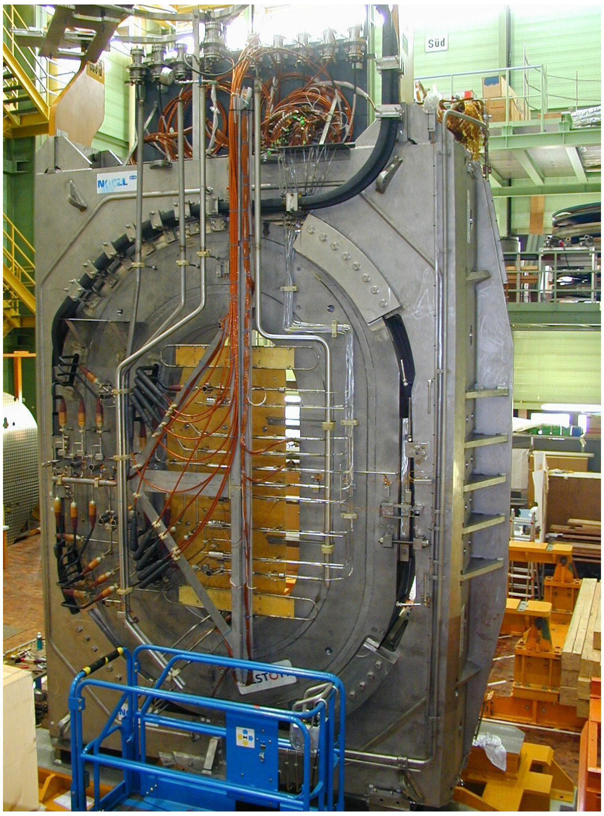


Figure 16 - Complete TF Model Coil prepared for Low-Temperature Tests

L3. Vacuum Vessel Sector

The main purpose of this project was to provide input required to complete the design, especially with regard to critical issues of fabrication technology - dimensional accuracy, welding distortions and achievable tolerances. The key issues can only be properly resolved by building a model at full scale (15 m high by 9 m wide). This was already completed for the earlier (1998) design of ITER, whose dimensions and mass were even larger than in the final design.

The main part of the project is a full scale sector model, manufactured by Japan. Hitachi and Toshiba have each built half sectors (Figure 17), giving the opportunity to test and compare different candidate weld schemes.

After the half sectors were fabricated, they were leak and pressure tested, and mechanical tests were performed to determine their structural characteristics. The welding together of the two half sectors demonstrated the welding techniques and verified the ability to undertake joint inspection by ultrasonic testing. The magnitude of sector fabrication tolerances and welding distortions due to the field joint welding were found to be lower than expected. Dimensional accuracy over this large scale was maintained at ±3 mm.

In parallel, the RF manufactured a full scale model equatorial port extension. This model was also used to develop and demonstrate fabrication technologies to the required specifications and tolerances, and related inspection techniques and procedures. The US developed a fully remotized welding/cutting system (Figure 18) to integrate the port extension in the sector model. This technology was subsequently transferred to Japan, and used to demonstrate integration of the RF port extension to the Japanese vessel sector, using also non-destructive testing techniques developed by the RF. Weld shrinkage in the port axis direction was about 5 mm.

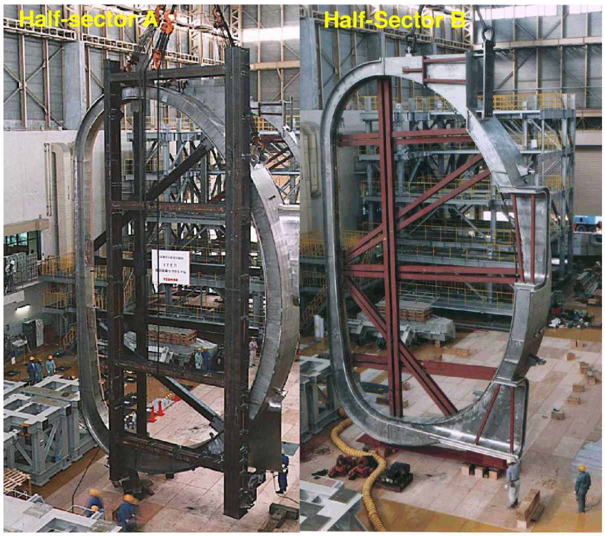


Figure 17 - Vacuum Vessel Sector Halves prior to Welding

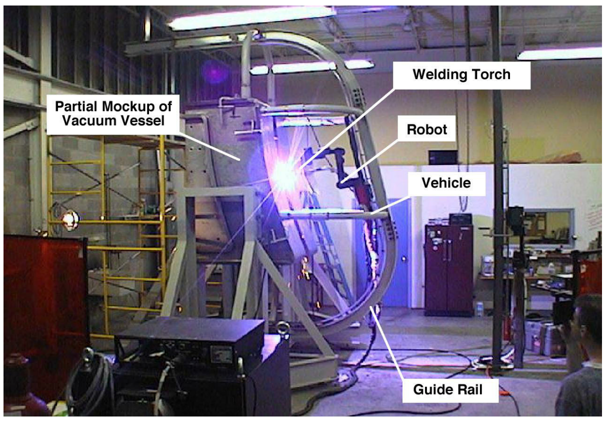


Figure 18 - Fully Remote Welding/Cutting Equipment for Vessel and Ports

L4. Blanket Module

The objectives of this project were to develop and fabricate prototype components for the shielding blanket, in order to assess their manufacturing feasibility, assemble them together and develop bolting, welding and cutting tools for the remote removal of the components, demonstrate the performance by testing representative parts of the components under relevant conditions, and to obtain confirmation of the design choices by results from accompanying R&D on materials, joining techniques and neutronics using a fast neutron source. The EU had lead responsibility for implementing the project, in collaboration with Japan and the Russian Federation.

Prototypes demonstrating and testing various candidate methods of joining the plasma-facing material beryllium to a copper substrate, and the copper to a stainless steel structure, were made and subsequently tested in high heat flux facilities. The most promising solutions were subsequently adopted in the design. The blanket modules will be attached to the vessel by slotted cylinders which are stiff axially but can flex perpendicularly, and these were shown to be stable to buckling at 3 times the load expected in ITER, giving adequate margin. In addition, large scale module manufacture was demonstrated, using various techniques of solid and powder hot isostatic pressing (Figure 19).

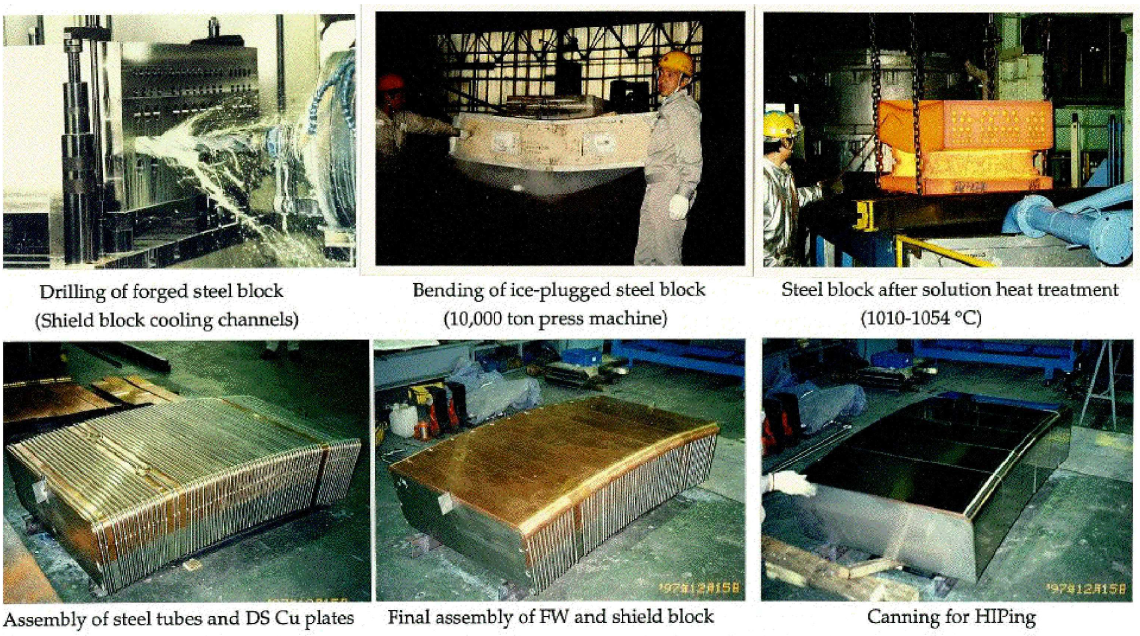


Figure 19 - Blanket Module Manufacture

L5. Divertor Cassette

The objectives of this project were to develop the technology needed to construct full-scale armoured components capable of meeting the inter-connected issues of providing adequate armour lifetime, armour-substrate joint lifetime (CfC-Cu & W-Cu) and substrate lifetime, sustaining thermo-hydraulic and electro-mechanical loads, whilst seeking the most cost-effective and reliable manufacturing options. All four Parties contributed to the development.

High heat flux performance tests were carried out on joints between Cu substrates and carbon-fibre composite and/or tungsten armour. Monoblock geometry proved to be the most reliable with no reported detachment of tiles. Tungsten brush armour was developed (Figure 20), and proved to be a solution to having a Cu-W joint capable of overcoming large difference in thermal expansion of the two materials under the high heat loads. One of the main aims of the project was also to integrate key plasma-facing components onto a realistic prototype of the cassette body. It is not essential to use all the real materials for this, and dummy components were made to save money (Figure 21). These were thermohydraulic equivalents of the real components and, in parallel, partial full size prototypes of the real components were tests at high heat fluxes. Major issues included the bonding of different plasma-facing materials on the same component, the selection of the substrate material (CuCrZr preferred), and the demonstration that it maintains its properties after manufacturing. Much was learnt during this manufacturing and testing programme, and the necessary features were incorporated into the design.

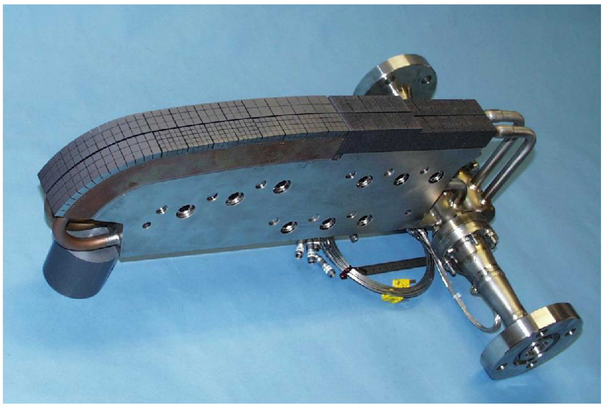


Figure 20 - Curved Vertical Target with Carbon and Tungsten Protection

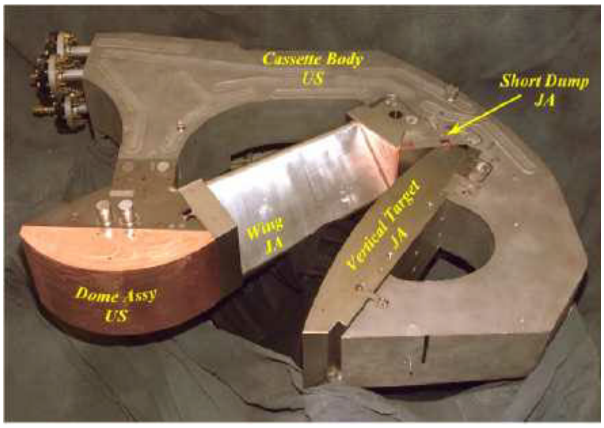


Figure 21 - Divertor Cassette Body Inboard Channel

L6. Blanket Module Remote Handling

The objective of this project was to develop and demonstrate the ability to remotely maintain blanket modules, including manipulating a 4 t module at a distance of 8 m with an accuracy of 1 mm, as well as welding/cutting of branch pipes, attachment/detachment from the vessel, and in-vessel viewing for inspection and monitoring.

A rail-mounted vehicle system was developed to handle the heavy blanket module within the limited space and with the required precision (Figure 22). After development and prototype demonstration of the main systems and techniques, full-scale testing and verification were carried out at JAERI's Naka laboratory (Figure 23). The test platform comprised module handling equipment, port handling equipment, auxiliary remote handling tools and a blanket mock-up structure to reproduce the physical environment of a 180° ITER in-vessel region. The payload capacity of the prototype was 1.2 t, to model the effects of the full system without the costs, and to bridge the technological divide between present and required capabilities.

Results showed that final positioning accuracy could be achieved within 0.5 mm and 0.1°, compensating for arm deflections. The rail could be deployed 90° around the torus in about 30 minutes. An advanced control system was developed using position/reaction force detection sensors and a graphical user interface to provide real-time observations of in-vessel transporter motion instead of a viewing camera.

A blanket module installation/removal test was successfully performed by using a teaching-play back procedure, keeping 0.25 mm clearance tolerances. This system was also tested with the advanced control system using information from sensors to detect positions/reaction forces. The principle has been proven of the use of auxiliary equipment and tools such as a transfer cask, double door, and pipe welding and cutting tools, including the development of radiation-hardened components.

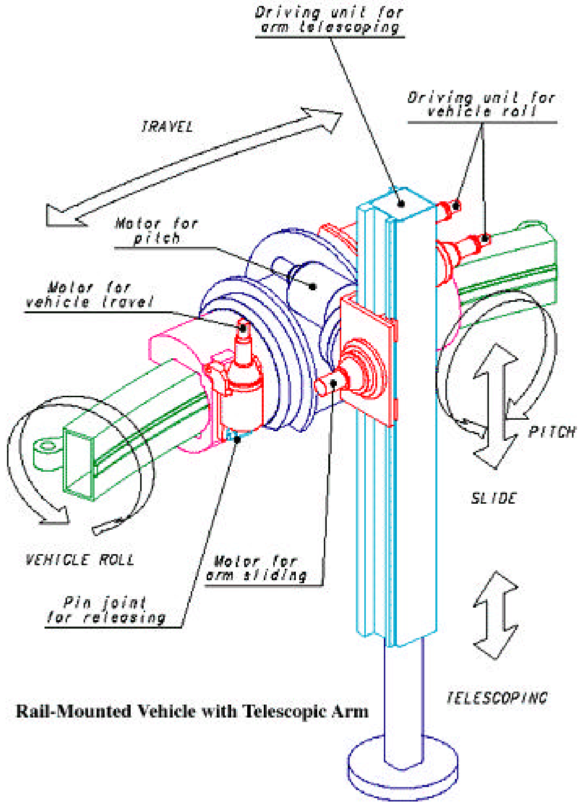


Figure 22 - Main Functions of Blanket Maintenance Vehicle

Figure 23 - Blanket Maintenance Facility

L7. Divertor Remote Handling

This project (Figures 24 and 25) was devoted to demonstrating that divertor remote maintenance operations were feasible and could be done in the required time. In ITER, such maintenance includes replacing and refurbishing all components 3 times during the first 10 years of operation and about 5 times overall, replacing and refurbishing 16 single faulty components during the machine life, positioning the high heat flux components so the maximum step between those on adjacent cassettes would be under 4 mm and so that the maximum variation around the whole torus would be within 10 mm, locking and securing the supports, making water pipe connections, assembling electrical connectors, and handling port plugs, and replacing all cassettes in less than 6 months and replacing a single cassette in under 8 weeks.

Two test facilities - a Divertor Test Platform (DTP) and a Divertor Refurbishment Platform (DRP) - were set up at the Brasimone Research Centre (Italy). The DTP was used to simulate at full scale all handling operations inside the vacuum vessel, including removal/replacement through vessel ports. Radial and toroidal movers were developed and used to demonstrate the required movements and positioning, and jacking and attachment tools were also developed.

Tests on the DTP confirmed the ITER maintenance concept, its integration inside the vessel, accuracy of cassette positioning, adequacy of nominal gaps and tolerances, and payload capabilities. Certain improvements were investigated to reduce costs and to implement lessons learned in the early tests to improve man-machine interface, sensors, and time, as well as to improve sliding components and to investigate rescue scenarios if components become jammed. Realistic estimates of intervention time are now possible. The total in-vessel time (i.e. excluding pipe cutting/welding, port handling and cask operations) to install 15 cassettes (one quarter of the divertor in the 1998 ITER design) is 32 hours.

The DRP simulates the most critical operations to be undertaken in the hot cell. The assembly and disassembly of high heat flux components (HHFCs) is simulated with prototype tools. Only those parts of the mock-up which are critical for HHFC mounting have been machined accurately. Tests show that the remote measurement system can be operated accurately enough (0.01 mm) that components can be correctly machined to fit. A target has been installed on the cassette with the required accuracy, and procedures have been streamlined to shorten the time taken.

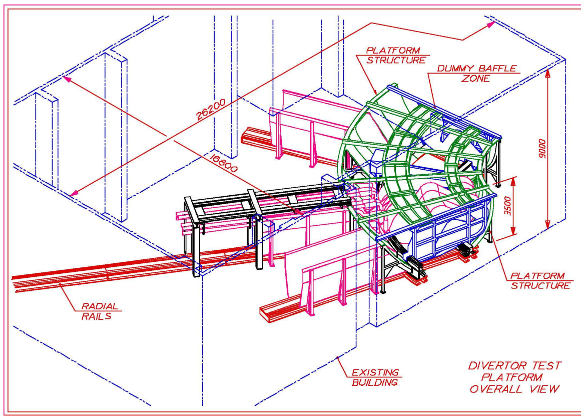


Figure 24 - Schematic of Divertor Remote Maintenance

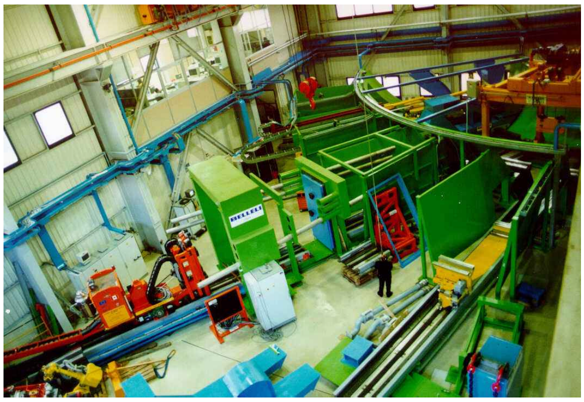


Figure 25 - Divertor Remote Handling Test Platform

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[2] ITER Physics Basis, Nuclear Fusion 39, 12, 2137-2664 (1999)

[3] ITER Joint Central Team and Home Teams (ed. Y. Shimomura), ITER Technology R&D, Fusion Engineering and Design, 55 (2001) Issues 2-3

[4] R.Aymar for the ITER Joint Central Team and Home Teams, ITER-FEAT – The Future International Burning Plasma Experiment Overview, 18th IAEA Fusion Energy Conference, Sorrento, Italy, October 2000, Paper ITEROV/1

[5] M.Huguet and the ITER Joint Central Team and Home Teams Key Engineering Features of the ITER-FEAT Magnet System and Implications for the R&D Programme, 18th IAEA Fusion Energy Conference, Sorrento, Italy, October 2000, Paper ITEROV6/1

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Participation in ITER

As mentioned above, the Home Team contribution to ITER has been made by all those working on ITER technology R&D and design tasks, and the Home Teams themselves are defined by those making the contributions. It is therefore not possible to provide a definitive list of those who have contributed in the Home Teams to the development of the ITER design, without omitting names and giving offence. Instead of this a list of organisations that have participated in these tasks for the Home Teams is given below.

Furthermore, any list of Home Team staff would not include those contributing to ITER as part of the voluntary physics programmes of the Parties, nor would it include the support staff of the Home Teams. However, the lack of a list of staff for Home Teams should in no way detract from the contribution of the Home Team participants.

The Home Teams were led by the following:

European Home Team:	R. Toschi (to December 1998) then K. Lackner
Japan Home Team:	S. Matsuda (to July 1998) then T. Tsunematsu (to July 2001) then M. Mori
Russian Federation Home Team:	O. Filatov
United States Home Team:	C. Baker

It is possible to determine those who have been assigned to the Joint Central Team and International Team, either for long term secondments, as visiting home team personnel, or as support staff. These are listed below.

Secondments

Europe (including Canada): Aymar, R., Barabaschi, P., Bareyt, B, Bartels, H-W, Benfatto, I., Bessette, D., Bosia, G., Boucher, D., Bruzzone, P-L., Cardella, A., Casci, F., Chiocchio, S., Costley, A., Dalle Carbonare, G., De Kock, L., Di Pietro, E., Dietz, J., Drew, M., Elio, F., Federici, G., Ferrari, M., Gambier, D., Girard, A., Gordon, Ch., Green, B., Haange, R., Hemmings, R., Hemsworth, R., Huguet, M., Ibbott, C., Ingala, L., Iseli, M., Janeschitz, G., Jong, C., Kveton, O., Ladd, P., Lyraud, A., Mann, J., Martin, E., Matera, R., Mills, M., Mitchell, N., Moledina, M., Mondino, P-L., Peridis, M. , Portone, A., Poucet, A., Raeder, J., Sannazzaro, G., Sborchia, C., Shaw, R, Sironi, M., Sovka, J., Spears, W., Tesini, A., Tivey, R., Vayakis, G, Verrecchia, M., Walker, C., Woodward, C., Wykes, M.

Japan: Ando, T., Ebisawa, K., Fujisawa, N., Hattori, Yu., Hiroki, S., Honda, Ts., Honda, T., Horikiri, H., Hoshi, Yu., Iida, F., Iida, H., Iizuka, T., Inoue, T., Ioki, K., Ishimoto, K., Ito, K., Itoh, M., Kataoka, Y., Kawai, Sh., Kitamura, K., Kobayashi, N., Kodama, T., Kondoh, M., Kuribayashi, T., Maruyama, S., Matsuhira, N., Matsukawa, M., Matsumoto, H., Matsumoto, Y., Matsunobu, T., Miki, N., Mita, Y., Mizoguchi, T., Mochizuki, E., Mohri, K., Mori, S., Moriyama, K., Nagashima, T., Nakamura, H., Nakashima, Y., Nishikawa, A., Odajima, K., Oikawa, A., Okada, H., Okuno, K., Onozuka, M., Osaki, F., Osano, K., Saji, G., Sato, K., Shibanuma, K., Shimada, M., Shimizu, K., Shimomura, Y.,Sugihara, M., Suzuki, Y., Tachikawa, N., Tajima, Y., Takahashi, Y., Takigami, H., Tanaka, S., Yamada, M., Yamaguchi, K., Yamamoto, S., Yonekawa, I., Yoshida, H., Yoshida, K., Yoshimura, K.

Russian Federation (incl. Kazakhstan): Antipenkov, A., Balasanov, Y., Barabash, V., Baulo, V., Britousov, N., Bykov, V., Chatalov, G., Chudnovski, A., Chuyanov, V., Gerasimov, S., Gribov, Y., Igitchanov, Yu., Ivanov, V., Kalinin, G., Kalinin, V., Kashirski, A., Kavin, A., Khripunov, V., Kostenko, A., Krasnov, S., Krivchenkov, Y., Krylov, A., Kukushkin, A., Lelekhov, S., Mitrishkin, Y., Morozov, S., Mukhovatov, V., Muraviev, E., Polevoy, A., Putvinski, S., Roshal, A., Rozov, V., Sadakov, S., Stepanov, B., Tanchuk, V., Topilski, L., Utin, Yu., Zapretilina, E.

United States: Abramovich, S., Ahlfeld, C., Baker, D., Bourque, R., Bowles, E., Burgess, T., Bushnell, C., Coombes, R., Dilling, D., Edmonds, P., Gallix, R., Gauster, W., Gohar, Y., Hager, R., Hanada, M., Hechler, M., Heckendorn, F., Holland, D., Iotti, R., Johnson, G., Johnson, L., Kahle, F., Koonce, J., Lindquist, W., Little, R., Loesser, D., Lousteau, D., Makowski, M., O'Connor, G., O'Toole, J., Parker, R., Perkins, F., Piec, Z., Piet, S., Post, D., Puhn, F., Raffray, R., Remsen, D., Rosenbluth, M., Santoro, R., Schaubel, K., Schleicher, R., Sheldon, R., Smith, P., Stoner, J., Stoner, S., Swift, R., Thome, R., Van Fleet, J., Vieira, R., Wesley, J., Williamson, D., Wong, F.

Visiting Home Team Personnel (if not included above)

Europe (including Canada): Albanese, R., Ambrosino, G., Bayetti, P., Bettini, P., Buende, R., Caroli, C., Cavinato, M., Ciscato, D., Heinemann, B., Helary, J-L., Holloway, C., Jakeman, R., Lister, J., Malaquias, A., Maschio, A., Moshonas, K., Ortolani, S., Petrizzi, L., Pironti, A., Pletzer, A., Porcelli, F., Pugh, J., Sauter, O., Tampucci, D., Taylor, N., Testoni, P., Thompson, E., Vivaldi, F., Wasastjerna, F., Zolti, E.

Japan: Araki, M., Fujisawa, N., Kuroda, T., Mori, S., Murasaka, E., Ohara, Y., Omori, J., Sato, S.

Russian Federation (incl. Kazakhstan): Alexeev, A., Arneman, A., Chatil, N., Chegodaev, A., Daniel, N., Danilov, I., Dlugach, V., Egorov, K., Epinatiev, E., Grigoriev, S., Ivashkin, A., Karsikov, Y., Karulin, N., Kavin, A., Khairoutdinov, R., Khomyakov, S., Kokotkov, V., Komarov, V., Korolkov, M., Koutchinski, V., Kuzikov, S., Kuzmin, E., Laboussov, A., Lukash, V., Makhankov, A., Malkov, A., Mitin, D., Nemtsova, A., Panasenkov, A., Panin, A., Pustovitov, V,. Razdobarin, G., Schipakin, O., Serebrennikov, D., Sorin, V., Sysoev, G., Vdovin, V.

United States: Wong, F., Vieira, R., Gertsch, P., Hager, R. Thomson, S., Nerem, A.

Support Staff (initial locations)

Garching: Andres, J., Barthel, T., Binder, M., Boselli, D., Bracco, S., Braun, B., Bruno,L., Campbell, J., Chodura, E., Chowanetz, J., D'Agata, E., Dirnberger, J., Dittl, P., Dowling, H., Duglue, D., Eisenreich, S., Eller, A., Esser, B., Ferres, C., Frediani. S., Friedrich, G., Fuss, S., Giegerich, M., Gorenflo, H., Gori, O., Gusic, C., Härtling, A., Hartmann, L., Haussmann, A., Heidl, H., Hurzlmeier, H., Igitchanova, E., Kalafat, A., Kerschl, P., Knebel, H., Krombholz, D., Kühn, I., Kuhn, M., Lehnert, S., Lightowlers, D., Lodato, A., Marrs, R., McHugh, G., McKehon, J., Meissner, P., Merz, G., Munro, P., Olm, L., Petzold, B., Plenteda, R., Polleichtner, R., Rutten, G., Ruvutuso, G., Sakkakibara, H., Santa, P., Scharl, G., Schittenhelm, M., Schnelle, B., Seidel, R., Solano, E., Spiegel, D., Steinbacher, T., Strohmayer, G., Taylor, A., Valenza, D., Vaslu, D., Vollmann, T., Walther, A., Weinfurtner, H., Wellmann, W., Wilhelm, B., Zolothukin, O

Naka: Asakura, Y., Avenall, P., Beckett, B., Cottham, P., Haneishi, Y., Haskell, H., Hirano, K., Hirayama, Y., Horikiri, M., Ito, T., Izumi, K., Junger, E., Kawakami,H., Kikuchi, K., Koga, Y., Komuro, Y., Kosaku, Y., Kuge, S., Kurosawa, M., Lockley, D., Long, M., Matsumoto, S., Matsuura, Y., McDonough, M., Messer, P., Miyagawa, Y., Motohashi, R., Murray, S., Nagayama, T., Neisch, R., Nemoto, K., Ohsaki, K., Osanai, T., Oshika, A., Ouchi, M., Pendleton, J., Puica, C., Robison, M., Shoji, K., Sibbald, M., Sugai, Y., Suzuki, T., Tabe, K. Takahashi, N., Tan, H., Terakado, Y., Terayuma, S., Terunuma, S., Tomita, F., Tomita, M., Umikawa, J., Warschewski, E., Winters, J., Yamada, Y., Yamaguchi, K., Yamane, Y., Yamauchi, T

San Diego: Acks, M., Anderson, C., Arrowood, L., Basile, A., Berman, G., Bradshaw, B., Cabrera, V., Carroll, N., Choate, C., Clelland, D., Cohen, D., Conroy, S., Dale, W., Ellefson, R., Endres, T., Erlenborn, M., Friend, M., Gantzel, K., Garb, M., Gardiner, S., Golubchikova, K., Gregory, P., Hansen, B., Henderson, W., Huerta, R., Jiron, C., Johnson, K., Jongeward, K., Kirschner, J., Kostrubala, A., Kotula, C., Lamoise, P., Lee, B., Lippert, C., Magnuson, J., Miller, L., Monserratt, E., Odedra, A., Ostermann, J., Park, R., Purdey, J., Putvinskaya, N., Sabado, M., Scharff, M., Tiernan, M., Yarborough, D

ITER EDA Technology R&D Contributors

EU

EFDA – European Fusion Development Agreement
C/o Max-Planck Institut für Plasmaphysik
Boltzmannstr, 2
D-85748 Garching bei München
Germany

ABB TRC - Sweden
Accel GmbH - Germany
AECL (Atomic Energy of Canada LTD.), Chalk River Laboratories – Canada
Alcatel - France
Alstom Belfort – France
Ansaldo Industria - Italy
Ansaldo Ricerche - Italy
Ansaldo Superconduttori – Italy
ATEA - France
Atmostat - France
Atominstitut der Österreichischen Universitäten , Wien - Austria

Babcock Noell Nuklear GmbH – Germany
Belgatom - Belgium
Belleli Energy S.p.A. – Italy
Bodycote HIP Ltd – UK
Bodycote Powdermet - Sweden
Böhler Edelstahl AG - Austria
Bruker – Germany

CEA Cadarache – France
CEA Fontenay-aux Roses - France
CEA CEN Grenoble – France
CEA Saclay – France
Cefival – France
CFFTP - Canada
CIEMAT Madrid – Spain
CLFA Arceuil - France
CNR Milano – Italy
Comex Nucléaire – France
Consorzio CREATE – Italy
Consorzio RFX – Italy
Co-ord3 - Italy
CRPP-EPFL - Switzerland
CRPP Fusion Technology – Switzerland
CSC - Italy
Cybernétix - France

DCN Indret – France
De Beers Industrial Diamond Division Ltd, Charters - UK
Department of Electrical and Computer Engineering, National Technical University of Athens –Greece
Department of Engineering Physics and Mathematics, Helsinki University of Technology –Finland
Dubna Datskov – Russian Federation
Dunlop - UK

ECN Petten – Netherlands
ENEA Bologna - Italy
ENEA Brasimone - Italy
ENEA Casaccia - Italy
ENEA Frascati – Italy
ERM/KMS Brussels - Belgium
Europa Metalli – Italy
European Fusion Engineering and Technology (EFET) EWIV - Germany

FIAT AVIO – Italy
FN - Italy
FOM-Instituut voor Plasmafysica “Rijnhuizen”-Netherlands
Forschungszentrum Karlsruhe – Germany
Forschungszentrum Jülich – Germany
Framatome ANP - France
Fraunhofer Institut für Angewandte Festkörperphysik, Universität Freiburg - Germany
Fravit – Italy

GEC Marconi - UK
Gradel – Luxembourg

Habia – Sweden
Helsinki University of Technology - Finland

ICIT – Romania
Idrosapiens - Italy
IESL / Foundation for Research and Technology Hellas (FORTH) - Greece
ILF Fraunhofer Institut Aachen - Germany
Imperial College – United Kingdom

Institut de Soudure – France
Institut für Plasmaforschung, Universität Stuttgart - Germany
Institut für Höchstfrequenztechnik und Elektronik, Universität Karlsruhe – Germany
Institute of Plasma Physics – Czech Republic
Institute of Solid State Physics, University of Latvia, Riga – Latvia
Instituto Superior Técnico - Centro Fusão Nuclear – Portugal
IVF - Sweden

JET Joint Undertaking – United Kingdom
Joint Research Centre (JRC), Ispra – Italy
JRC Petten - Netherlands

Katholieke Universiteit Leuven – Belgium
Kind GmbH – Germany
Kinectrics Inc - Canada
Kraftanlagen - Germany
Kvalitetsproduktion AB – Sweden

L'Air Liquide – France
LPMI, University of Nancy – France
LTC - Italy

Max-Planck Institut für Plasmaphysik Berlin - Germany
Max-Planck Institut für Plasmaphysik (IPP) Garching - Germany
Mécaéro - France
Mécachrome - France
Messer Griesheim - Germany
Metso Powdermet Oy - Finland
Motala Verkstad - Sweden

National Institute for Laser, Plasma and Radiation Physics, Magurele - Romania
National Institute of R&D for Physics and Nuclear Engineering “Horia Hulubei”, IFIN-HH, Bucharest - Romania
NFM - France
Nitek Corporation - Canada
NNC Ltd.- United Kingdom
NRG - Netherlands

Österreichisches Forschungszentrum Seibersdorf – Austria
Oxford Instruments Ltd. – United Kingdom

PCI - France
Phoenix - UK
Plansee - Austria
Politecnico di Torino - Italy
Proform - France

Risø National Laboratory - Denmark
Ritter Starkstromtechnik GmbH - Germany
RTM-Istituto per le Ricerche di Tecnologia Meccanica ed Automazione - Italy

SCK/CEN Mol - Belgium
Siemens AG – Germany
Slovak Academy of Sciences - Slovakia
Snecma - France
Spaggiari - Italy
SWEMA - Sweden

Tampere University of Technology - Finland
TCS - France
Techmeta - France
Technische Universität Dresden - Germany
Technische Universität Hannover - Germany
Technische Universität München - Germany
Technische Universität Wien – Austria
Tecphy - France
TEKES – Finland
Thales Electron Devices - France
TWI Cambridge - United Kingdom

UKAEA Fusion Culham - United Kingdom
Universita’ di Bologna - Italy
Universita’ di Napoli - Italy
Universita’ di Padova - Italy
Universita’ di Reggio Calabria – Italy
Universita’ di Udine - Italy
Universite’ de Montreal (Tokamak de Varennes) - Canada
University of Lapenraanta - Finland
University of Liverpool - UK
University of Manchester - UK
University of Saskatchewan - Canada
University of Sussex - UK
University of Stuttgart - Germany
University of Toronto - Canada
University of Twente – Netherlands
Usinor Industeel Le Creusot – France

Vacuumschmelze – Germany
Verdict Aerospace - UK
VR – Sweden
VR Studsvik – Sweden
VTT Manufacturing Technology - Finland

WEKA - Switzerland

Japan

Japan Atomic Energy Research Institutes (JAERI)
Depertment of ITER Project
801-1 Mukouyama, Naka-machi
Naka-gun, Ibaraki-ken, 311-0193
Japan

Universities and Institutes(*):
Doshisha University
Hokkaido University
Ibaraki University
Iwate University
Keio University
Kyoto University
Kyushu University
Mie University
Nagoya University
National Institute for Fusion Science
Okayama University
Osaka University
Shizuoka University
The University of Tokushima
The University of Tokyo
Tohoku University
Tokyo Institute of Technology

Toyama University
University of Tsukuba
Yokohama National University

(*) The above Universities and Institutes have contributed to the ITER EDA Technology R&D through research contracts with JAERI. The review on the technical conditions and items has been conducted at the ITER/EDA Research Collaboration Review Committee managed by National Institute for Fusion Science.

Industries:
Atago Bussan Co., Ltd.
Fuji Electric Co., Ltd.
Fuji Idek Co.
Fujikura Ltd.
Furukawa Co., Ltd.
Hitachi Cable, Ltd.
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Hitachi, Ltd.
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Nippon Oil Co.
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The Furukawa Electric Co., Ltd.
Tokyo Tungsten Co.
Toshiba Corporation
Toyo Tanso Co.
Ube Industries, Ltd.

Russian Federation

D.V. Efremov Scientific Research Institute of Electrophysical Apparatus,
Institute of Electrophysical Apparatus
St. Petersburg 189631 Russia

Russian Research Centre "Kurchatov Institute" (Nuclear Fusion Institute), Moscow
Research and Development Institute of Power Engineering, (Technical University), Moscow

Sverdlovsk Branch of Research and Development Institute of Power Engineering
A.A. Bochvar Scientific Research Institute of Inorganic Materials, Moscow
Nuclear Physics Institute of M.V. Lomonosov, Moscow State University
Skobelzin Nuclear Physics Institute, Moscow State University
State Research Center «Institute of Theoretical and Experimental Physics», Moscow

St. Petersburg State Technological Institute (Technical University)
St. Petersburg State University,
St. Petersburg State Technical University
Troitsk Institute for Innovation and Fusion Research
State Scientific Center "Institute of Physics and Power Engineering", Obninsk

Scientific Research Institute of Atomic Reactors, Dimitrovgrad
Engineering Centre of Nuclear Equipment Strength, Reliability and Lifetime, Moscow
All-Russian Design and Research Institute of Complex Power Technology, St. Petersburg
V.G.Khlopin Radium Institute, Research Industrial Enterprise "KIRSI", St. Petersburg
I.I. Polzunov Central Boiler Turbine Institute, St. Petersburg
Institute of Physical Chemistry of the Russian Academy of Sciences, Moscow

N.E. Bauman Moscow State Technical University
Moscow Physical Engineering Institute
Institute of Atomic Energy, National Nuclear Center of the Republic Kazakstan, Almaty
Institute of Theoretical and Experimental Physics, Kazakhstan State National University, Almaty
Central Research Institute for Shipbuilding Technology, St. Petersburg
All-Russian Scientific Research Institute for Cabling, Moscow

Central Researsh Institute of Structural Materials "Prometey", St. Petersburg
Ioffe Physical-Technical Institute, St. Petersburg
All-Russian Institute of Light Alloys, Moscow
State Enterprise "Red Star", Moscow
Scientific Industrial Center "MEGAIX", All-Russian Institute of Experimental Physics, Sarov, Nizhniy Novgorod region

Scientific Industrial Enterprise Central Scientific Research Institute of Machine-building Technology, Moscow
Experimental Machine-building Plant of "Energy" Rocket Space Corporation Korolev, Moscow region
Scientific Research Institute of Elastomer Materials and Products, Moscow
Russian Institute of Chemical Technology, Moscow
Gosatomnadzor, Moscow

Institute of Applied Physics of the Russian Academy of Sciences, Nizhni Novgorod
Institute of General Physics of the Russian Academy of Sciences, Moscow
Institute of Biophysics, Moscow
Industrial and Marine Medicine Institute, St. Petersburg
Institute of solid-state Physics of the Russian Academy of Sciences, Moscow

Institute of Applied Mathematics, Moscow
Research and Production Corporation "TORIY", Moscow
Izhorskie Zavody, St. Petersburg
Special Design Bureau "Kompensator", St. Petersburg
Scientific and Production Center "ECHO+", Moscow

Scientific and Production Association "Kompozit", Moscow
Ulba Metallurgical Plant, Ust' Kamenogorsk, Kazakhstan
JSC "Expocabel", Pilot Cabling Plant, Podolsk
Cabling plant JSC "Kirskabel", Kirsk, Kirov region
JSC "VNIIMetmasch", Moscow

Podolsk Machinbuilding plant "ZIO", Podolsk
"Istok", Fryasino
"Digazkron", Moscow
Scientific-research and Design Institute of Assembly Technology, Moscow
Central Research Institute of the Machine Manufacturing Technology, Moscow

Industrial Amalgamation "Atommash", Volgodonsk
Electrophysical Equipment Plant, St. Petersburg
Scientific Research Institute of SIA "Looch", Podolsk
"Spetssplav" Firm, Moscow
Institute of Non-Ferrous Metals, Moscow

Institute "Graphite", Moscow
Institute of Applied Chemistry, Moscow
St. Petersburg Institute of Nuclear Physics
SIA "Ritm", St. Petersburg
Shipbuilding Plant "Severnaya Verf", St. Petersburg

PELIN Laboratory, St.-Petersburg
Scientific and Production Enterprise, St. Petersburg
Central Air-Hydro-Dynamic Institute (CAHI, Zhukovsky, Moscow region)
Scientific Industrial Enterprise "ADRON", Institute of Theoretical and Experimental Physics (SIE "ADRON" ITEP, Moscow)
Enterprize "Kamov" (Moscow region)
Scientific Institute of Aviation Technologies
Moscow Radio Technical Institute
GYCOM Ltd. N.Novgorod
Institute of Beryllium, JSCo "Kompozit", Moscow
RSC "ENERGIA", Moscow region

USA

Argonne National Laboratory
Brookhaven National Laboratory
Columbia University
General Atomics
Georgia Institute of Technology
Idaho National Engineering and Environmental Laboratory
Jackson State University
Lawrence Livermore National Laboratory
Lodestar
Los Alamos National Lab
Lockheed Martin
Massachusetts Institute of Technology
McDonnell Douglas Aerospace (now Boeing)
National Institute of Standards & Technology
Naval Research Laboratory
New York University
Oak Ridge Institute of Science & Education
Oak Ridge National Laboratory
Pacific Northwest Laboratory
Pitt Des Moines, Inc.
Princeton Plasma Physics Laboraotry
Raytheon Engineering and Constructors, Inc. (was Ebasco)
Rockwell International Corporation (Rocketdyne)
Science Applications International Corporation
Sandia National Laboratories
Savannah River Site
Stone & Webster
TSI Research
University of California, Berkeley
University of California, Davis
University of California, Los Angeles
University of California, San Diego
University of Illinois
University of New Mexico
University of Texas
University of Washington
University of Wisconsin
Varian (now CPI)
Westinghouse Electric Corporation

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