FUSION
FOR
ENERGY

Part

B

**Fusion for Energy**

*The European Joint Undertaking for ITER and the Development of Fusion Energy*

# Technical Activity Report 2007-2008

FUSION FOR ENERGY

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The European Joint Undertaking for ITER and the Development of Fusion Energy
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# Table of Contents

Chapter

1

# Introduction to Fusion Energy Research and Technology

## Fusion Energy

Fusion reactions power the sun by converting light nuclei into heavier ones under the extreme pressure (340 billion atmospheres) and temperature (16MºC) due to gravity. The p+p reaction is the most important one. In these reactions about half of a percent of the hydrogen mass is converted into energy, in accordance with Einstein's well-known $E = mc^2$ equation relating mass and energy.

Mastering fusion energy on earth requires the development of other processes. Among all possible reactions, the fusion of deuterium (D) and tritium (T), i.e. D-T fusion, which produces helium and neutrons (Figure 1), is the easiest to achieve and has been chosen for future fusion power plants. Tritium is 'bred' inside the reactor by the reaction of lithium with a neutron.

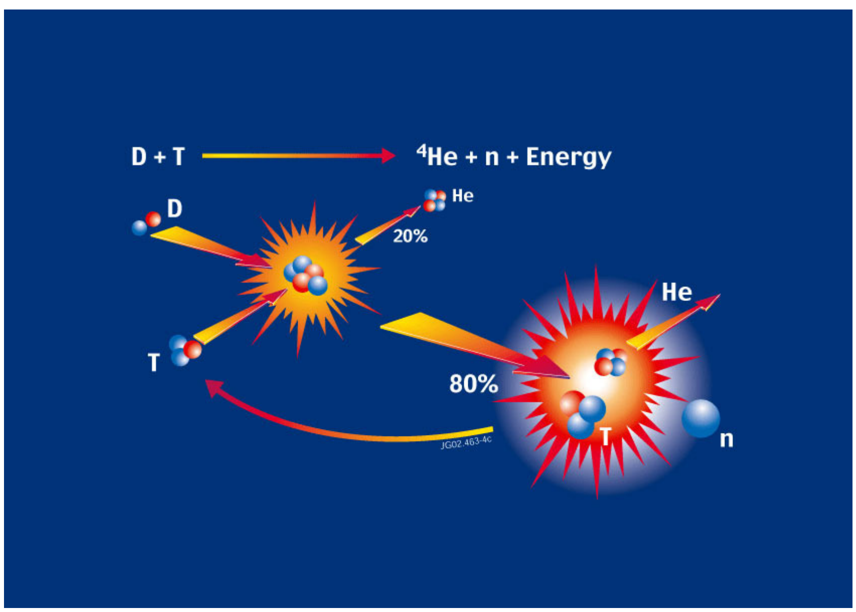


Figure 1 - The D-T Fusion Reaction

In order to fuse, the atomic nuclei have to be given sufficient energy (i.e. velocity) to overcome their mutual electrostatic repulsion when they collide. This kinetic energy is provided by heating the fuel to very high temperatures (for D-T fusion the maximum efficiency is achieved in the range 100 –150 million degrees). At such temperatures the gaseous fuel is completely ionised, being in the form of a “plasma”. The plasma must not be allowed to come into contact with the walls of the reaction container, since some of the surface layer of the wall would melt and evaporate and the plasma would be immediately polluted and cooled, losing the conditions for fusion reactions to occur.

In the machine with magnetic confinement, the plasma is held in a doughnut shaped vessel, a “torus”, and kept away from the walls by strong magnetic fields. Two main types of machines follow this approach: tokamaks (Figure 2) and stellarators. In future reactors, the pressure in the core will be about 10 atmospheres. The magnetic field traps the helium ions produced by the D-T fusion reaction and so imparting their energy to the plasma and providing self-heating. On the other hand, being electrically neutral, the neutrons, which carry 80% of the DT fusion energy, cross the magnetic field and deposit their energy in the blanket surrounding the plasma.

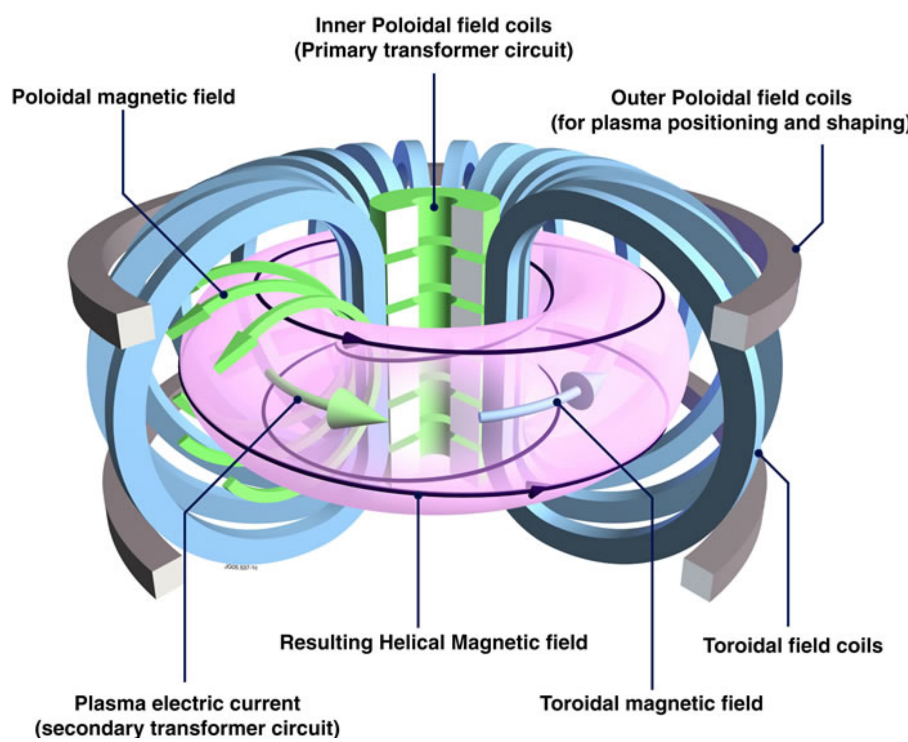


Figure 2 - Schematic of a Tokamak

Nuclear fusion can also be achieved on earth by 'inertial confinement', when a tiny pellet, filled with D-T fuel, is flash-irradiated with soft X-rays generated by many beams of high power lasers or atomic nuclei. Ablation of the pellet surface compresses the remaining pellet to very high pressure, comparable to that in the centre of the sun, such that large numbers of fusion reactions occur and the fuel is 'burned' whilst its own inertia keeps it in place.

Fusion research has seen enormous world-wide progress over the past decades. A major achievement was attained in the JET (Joint European Torus) tokamak, located near Oxford (UK), where, in 1997, controlled D-T fusion reactions released a fusion power of 16 MW (Figure 3). In the same series of experiments, plasma self-heating by the DT fusion reactions was observed.

The next step in the development of fusion is an international project, called ITER, aiming at producing fusion power in the hundreds of megawatt range, demonstrating the scientific feasibility and a number of key technologies necessary to make fusion a viable energy source.

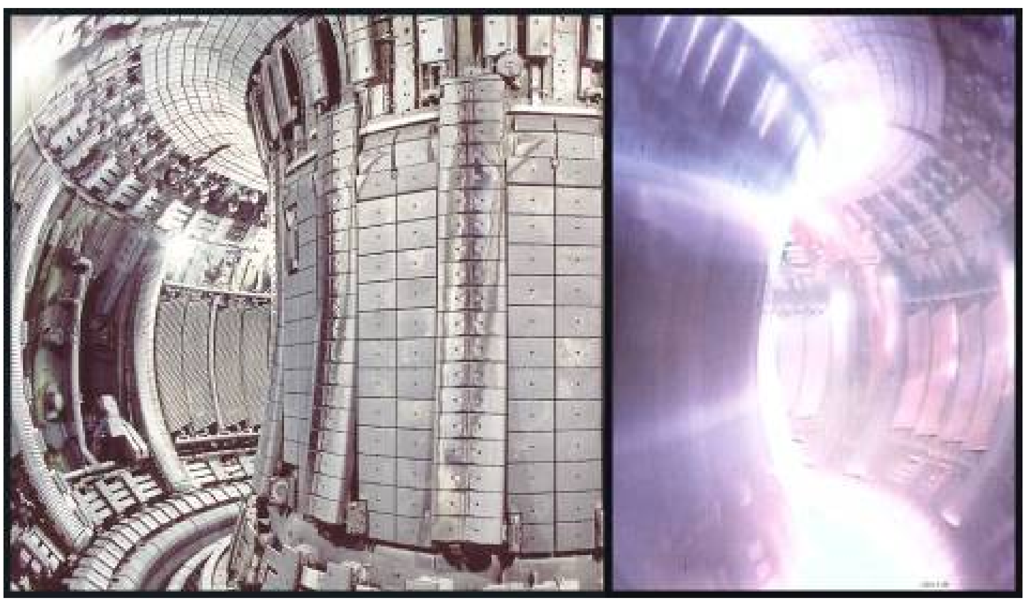

Figure 3 - JET vacuum vessel with and without plasma

The merits of fusion are the large availability on earth of the basic fuels (deuterium and lithium), the lack of production of greenhouse gas emissions, a very low impact on the environment with no long-lasting radioactive waste, and the inherent safety of the reactors, where no meltdown or runaway reactions are possible.

# European Fusion Programme

**Introduction**

In Europe, fusion research takes place in a great number of national research institutes and universities. At European level fusion research and development has been part of the Community research programmes since the inception of the Euratom Treaty in 1957. It has been recognised as one of the flagships of the European Research Area (ERA).

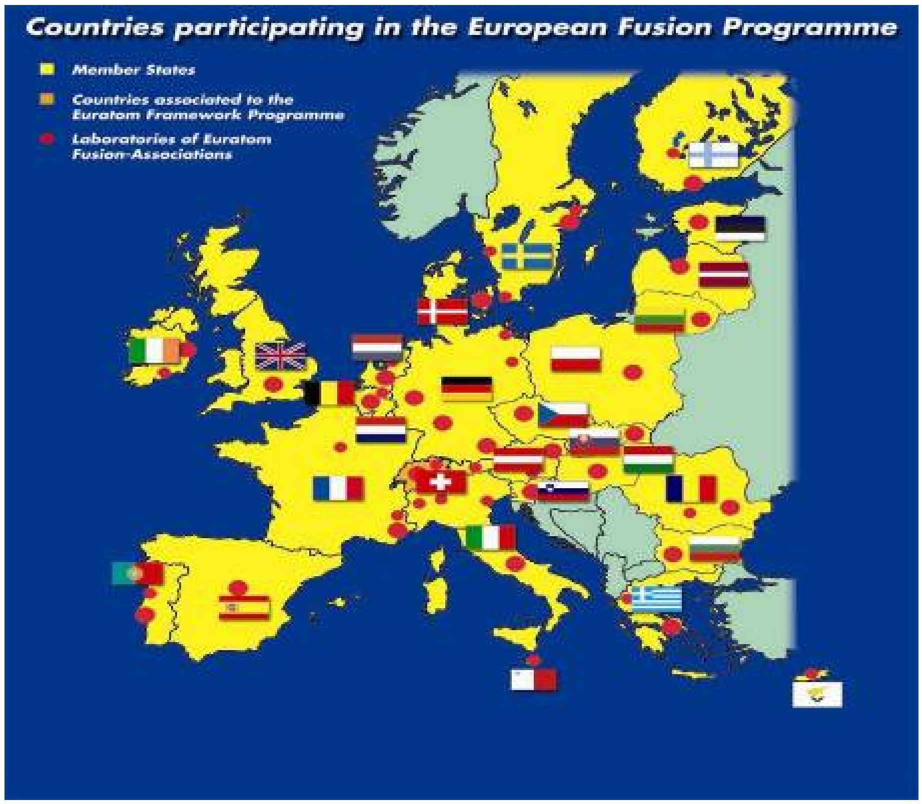


Figure 4 - Map showing the Euratom Associations

The main mechanism of participation in the programme is the “Contract of Association”. Each state, or organisation within a state, concludes a contract with the European Commission (representing Euratom), creating a “Euratom Association”. This contract specifies the work programme to be carried out within the overall Work Programme for fusion, promotes mutual access to major facilities, and provides the mechanism for funding from Euratom.

The European Union has established a fully integrated and co-ordinated programme in fusion research with participation by the majority of Member States and associated third states. The programme is managed by the European Commission, which is assisted by a consultative committee of

Member State representatives in the implementation of the programme (the "CCE-FU").

**Seventh Framework Programme (FP7)**

Under the Seventh multi-annual Framework Programme of the European Atomic Energy Community (Euratom) for nuclear research and training activities (2007 to 2011) often called "FP7", the specific programme for fusion energy aims to develop the knowledge base for ITER and its implementation, as a major step towards the creation of safe, sustainable, environmentally responsible, and economically viable prototype reactors, including:

- Realisation of ITER - site preparation, establishing the ITER Organisation and the European Joint Undertaking for ITER, management, technical and administrative support, construction of equipment and installations and support to the project during construction;
- R&D in preparation of ITER operation - physics and technology research to exploit the facilities and resources in the fusion programme and JET, assessing specific key technologies, consolidating ITER project choices, and preparing for operation;
- Technology activities in preparation of a demonstration fusion power station;
- R&D activities for the longer term - further development of improved concepts for magnetic confinement schemes, theory and modelling for understanding of the behaviour of fusion plasmas and coordination of Member States' research on inertial confinement;
- Human resources, education and training in view of the immediate and medium term needs of ITER, and for the further development of fusion;
- Infrastructures - construction of the international fusion energy research project ITER will be an element of the new research infrastructures with a strong European dimension.

Under FP7, fusion energy research is financially supported with an overall allocation of EUR 1 947 million on the basis of the following instruments:

- the Contracts of Association, between the Commission and Member States or Associated Countries or entities within those states;
- the European Fusion Development Agreement (EFDA);

- the European Joint Undertaking for ITER and the Development of Fusion Energy;
- international agreements between Euratom and third countries covering activities in the field of fusion energy research and development, in particular the ITER agreement;
- any other multilateral agreement concluded between the Community and associated organisations, in particular the Agreement on staff mobility;
- cost sharing actions with Coordination and Support Actions.

**European Fusion Development Agreement (EFDA)**

The European Fusion Development Agreement (EFDA) is a framework contract between Euratom and the national European fusion research institutions (Euratom Fusion "Associations"). EFDA focuses on research coordination with two main objectives: to prepare for the operation and exploitation of ITER and to further develop and consolidate the knowledge base needed for overall fusion development and in particular for DEMO, the first electricity producing experimental fusion power plant being built after ITER..

EFDA is based in two locations where Close Support Units (CSU), responsible for one or more of EFDAs activities are hosted. The EFDA-CSU Garching is located in Garching, near Munich (Germany), and is hosted by the German Max-Planck Institut für Plasmaphysik. The EFDA-CSU Culham is hosted by the UKAEA laboratory in Culham (UK), home of the Joint European Torus facilities. The planning and supervision of the activities carried out under EFDA is the responsibility of the EFDA Steering Committee in which all parties are represented. It appoints the EFDA Leader and the Associate Leader for JET, upon proposal by the Commission. The EFDA Leader and the Associate Leader are assisted by staff working at the CSUs.

EFDA was originally signed in 1999 and has been extended several times. Implementing Agreements relating to activities in specific EFDA areas are concluded separately, in particular for the collective use of JET.

**JET**

The JET Joint Undertaking was established in 1978 to construct and operate the Joint European Torus (JET) in Culham, UK, and in its time was the largest project within the European nuclear fusion programme. It started operating in 1983 and was the first fusion facility in the world to produce significant fusion power with a deuterium-tritium experiment in 1991. In the mid-1990s JET underwent several major upgrades to address physics and technology issues of relevance to ITER. During 1997 JET carried out three months of experiments using a range of deuterium-tritium fuel mixtures. The results were of major significance and three new world records were set: 22 MJ of fusion energy in one pulse, 16 MW of peak fusion power and a 65% ratio of fusion power produced to total input power.

The JET Joint Undertaking, was superseded on 31st December 1999 and ownership was transferred to the UK Atomic Energy Authority (UKAEA). The overall implementation and co-ordination of further scientific exploitation is now carried out under EFDA with collaboration from many Euratom Associations and scientists from around the globe.

With its divertor configuration, plasma size, heating, current drive and diagnostic systems, tritium, beryllium and remote handling capabilities, the JET device remains a unique facility that can access a wide range of operating regimes in experimental conditions closest to those expected within ITER. In addition, a number of upgrades are underway to study wall materials and radio frequency heating among others.

The JET Implementing Agreement serves as a single framework agreement between Euratom and its Associates for all the scientific and technical tasks carried out with regard to JET by the Associates. The exploitation of the JET facilities by the Associates is organised in a campaign-oriented manner. The facilities are operated by the UKAEA under a contract with Euratom, the JET Operation Contract.

## ITER

ITER – "the way" in Latin – will be the next major experimental tokamak facility. Its objective is "to demonstrate the scientific and technological feasibility of fusion energy for peaceful purposes". ITER is being constructed in the framework of an international collaboration between the European Union (including Switzerland), India, Japan, the People's Republic of China, the Russian Federation, the Republic of Korea, and the United States of America, under the auspices of the International Atomic Energy Agency (IAEA).

ITER, a fusion tokamak capable of generating 500MW of fusion power for about 7 minutes or 300MW for 50 minutes, began in 1985 as a collaboration between the then Soviet Union, the United States, the European Union and Japan. Conceptual and engineering design phases led to a commonly agreed detailed design in 2001. During these phases $650million worth of research and development by the ITER parties was spent defining the design and establishing its practical feasibility primarily through the construction of full-scale prototypes of key components (Figure 5). The ITER parties, with the Russian Federation replacing the Soviet Union in 1992, the United States opting out of the project between 1998 and 2003, the People's Republic of China, the Republic of South Korea and

India signed in November 2006 (Figure 6) the Joint ITER Agreement, entering into force in October 2007, after ratification by all signatories.

The leading fusion experiments such as JET (Culham, UK), JT-60 (Naka, Japan) and TFTR (Princeton, USA, closed in 1997) have provided the expertise in fusion physics and technology in preparation for ITER.

The smaller European machines in the EURATOM Associations have also contributed important data to define the ITER physics basis. The crucial next step in fusion research is to study the physics of burning plasmas and to demonstrate and test the key technologies for developing fusion as a safe and environmentally benign energy source. The ITER project is this next step, and will provide the physics and technological basis for the construction of a demonstration electricity generating power plant.

The overall ITER plant comprises the tokamak machine, its auxiliaries and supporting plant facilities. ITER (Figure 7; Table 1) has a vertically 'D' shaped plasma and a lower divertor. The divertor is the main area of contact of the plasma and is one of the most critical components in the machine as it controls the particles escaping the plasma, helium and the amount of impurities in the plasma and has to withstand high surface heat loads of up to 10MW/m$^2$.

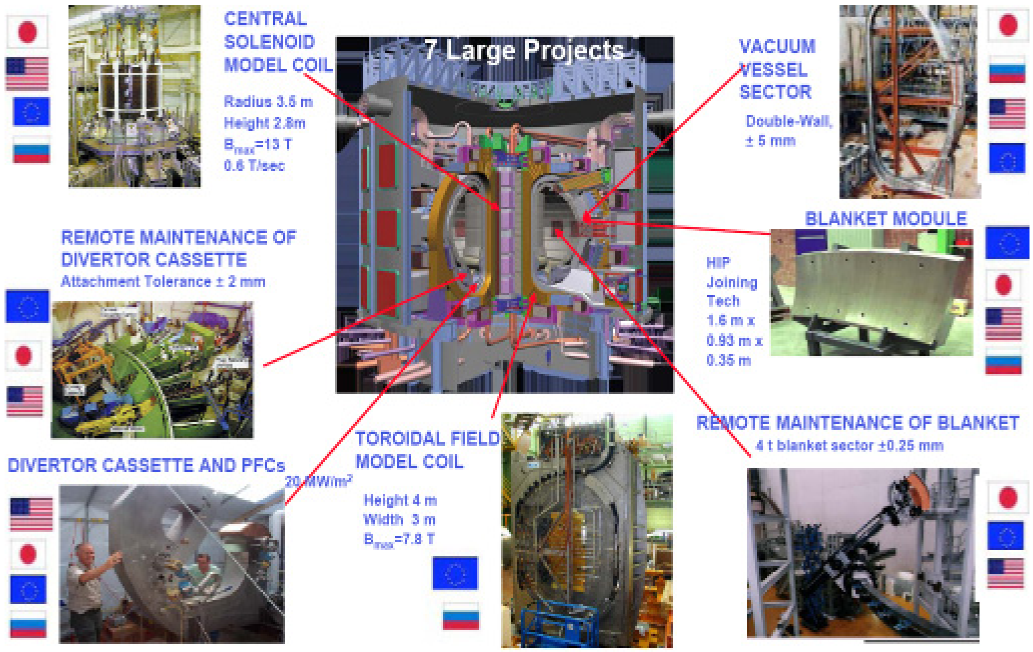


Figure 5 - The ITER Seven Large R&D Projects

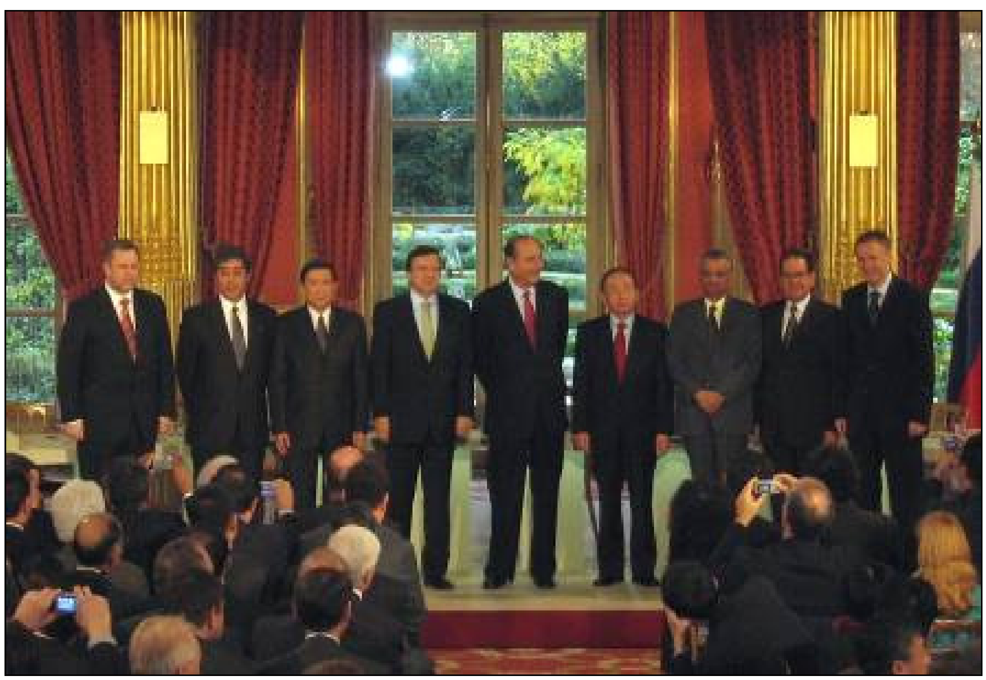

Figure 6 - The signature of the ITER Agreement (November 21st 2006)

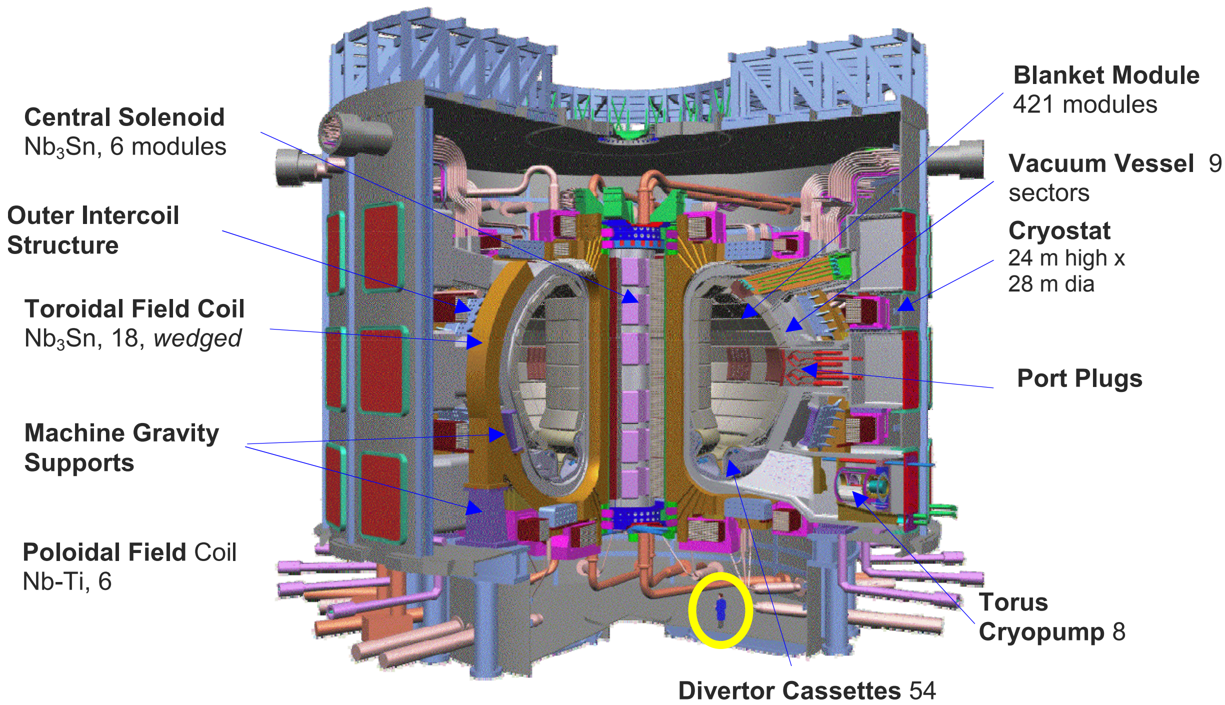


Figure 7 - The ITER Machine

| | |
|---|---|
| Major radius R | 6.2 m |
| Minor radius a | 2 m |
| Plasma current | 15 MA |
| Toroidal magnetic field | 5.4T |
| Fusion power | 500 – 700 MW |
| Ion temperature $T_i$ | 120 millions degrees |
| Fusion power gain Q* | 10 - 5 |
| Average wall loading from fusion neutrons | 0.57 MW/m$^2$ |
| External heating power (Initial phase) | 33 MW of Neutral Beam Injection<br>20 MW at the 50 MHz (Ion cyclotron )<br>20 MW at 170 GHz (Electron cyclotron) |
| Pulse duration | 500 s |
| Density n | $10^{20}$ m$^{-3}$ |
| Energy confinement time $\tau_E$ | 3.7 s |
| $n\tau_E T_i$ | 4 x $10^{21}$ m$^{-3}$s keV (corresponding to Q =10) |

* Q= generated fusion power/power injected into the plasma

Table 1 - The ITER Parameters

The ITER parameters (magnetic field, plasma current and plasma volume) have been chosen for a plasma performance (density, temperature, and confinement time) so as to obtain a power multiplication factor Q ~10; this is the factor by which the fusion power exceeds the input heating power (50MW) to the plasma (see Table 1). At Q ~ 10, the plasma self heating by DT fusion reactions is dominant and this is the reason for choice of this performance target. In a future reactor, for economic reasons, this factor will be typically 30 to 50.

External systems are also used to heat the plasma and to drive the plasma current. Ultimately, the external current drive will extend the nominal inductive burn duration of 7 minutes, using the transformer alone, up to 50 minutes or more. Plasma control is provided by the poloidal field coils (see Figure 7), control coils, pumping and fuelling and heating systems with feedback from diagnostic sensors.

ITER uses low temperature superconducting magnets for both the toroidal and the poloidal coils. These coils, which can generate a toroidal (see Figure 7) magnetic field of 5.3T on the plasma axis, hold the plasma inside the vacuum chamber and limit contact with the chamber walls. Access (ports) for the heating systems, diagnostics, test blanket modules, pumping and equipment to be used during the remote maintenance of the machine, are distributed around the vacuum chamber surface at three levels.

The inner surfaces of the vacuum vessel are covered with blanket modules which provide the shielding from the high energy neutrons generated by the fusion reactions and, with their coolant (water) remove the heat produced by the plasma. In future reactors, these modules will contain also lithium and neutron multipiers for breeding tritium which will then be injected into the plasma in a closed system.

In ITER, six Test Blanket Modules (TBMs) will be installed in three ports to test different reactor-relevant breeding blankets concepts. In particular, Europe will develop and test two TBMs based on helium-cooled lithium lead and helium-cooled pebble bed.

70K cryostat encloses the whole machine to allow a secondary vacuum for the superconducting coils, necessary to maintain their temperature of 4K by circulation of liquid helium.

The site of the ITER machine is Cadarache, in the South of France, where the activities to prepare the area where the buildings will be erected (Figure 8) have already begun. The ITER buildings form an integrated complex of 28 buildings extending over an area of about 50 hectares.


Figure 8 - The ITER project site lay-out: 3D graphic

# Broader Approach

**Introduction**

On the 5th February 2007, the Government of Japan and the European Atomic Energy Community (Euratom) concluded, in Tokyo, an agreement for the joint implementation of the Broader Approach Activities in the Field of Fusion Energy Research (usually referred to as “Broader Approach Agreement” or BA Agreement). The BA Agreement comprises the following three projects:

(a) the Satellite Tokamak Programme (STP);

(b) the Engineering Validation and Engineering Design Activities for the International Fusion Materials Irradiation Facility ( IFMIF/EVEDA);

(c) the International Fusion Energy Research Centre ( IFERC).

For the execution of this agreement Japanese Government appointed JAEA as “JA Implementing Agency” and, correspondingly, Euratom delegated Fusion for Energy (F4E), as the “EU Implementing Agency”.

Each project is subdivided into several work packages, assigned to the EU or JA, and detailed, for technical and management aspects, in the so-called

‘Procurement Arrangements (PAs), to be agreed and signed by the two Implementing Agencies.

The EU Voluntary Contributors (VCs) for BA, namely the Governments of Belgium, France, Italy, Germany, Switzerland, and Spain have pledged to contribute to one or more of the three BA projects, covering, with few exceptions, the Euratom commitments, in terms of personnel secondment, design, R&D, and the actual procurement of components, systems and services.

**Satellite Tokamak Programme**

The scope of the Satellite Tokamak Programme includes:

a) the participation in the upgrade of the tokamak experimental equipment JT-60U, owned by the Japanese Implementing Agency, to the JT-60SA Advanced Superconductive Tokamak. The construction phase includes design activities, manufacturing of components and systems, as well as integrated assembly;

b) the participation in its exploitation, to support the exploitation of ITER and research towards DEMO by addressing key physics issues for ITER and DEMO.

The construction and exploitation of the Advanced Superconducting Tokamak is conducted under the Satellite Tokamak Programme and the Japanese national programme.

The EU in-kind commitment comprises the toroidal field (TF) magnets (including conductor, TF coil cold test facility and current leads), the power supplies and controls, the cryostat, the cryogenic systems and the power supplies for the Electron Cyclotron Radiofrequency heating system.

**IFMIF/EVEDA**

This project covers activities involving Engineering Validation and Engineering Design, in order to produce:

(i) a detailed, complete and fully integrated engineering design of the International Fusion Materials Irradiation Facility (IFMIF);

(ii) all data necessary for future decisions on the construction, operation, exploitation and decommissioning of IFMIF; and

(iii) to validate the continuous and stable operation of each IFMIF subsystem.

Such design and technical data will be set out in a final design report, to be adopted by the BA Steering Committee, upon proposal by the IFMIF/EVEDA Project Leader, and made available for each of the Parties, to use either as part of an international collaborative programme or in its own domestic programme. The implementation of this programme includes:

(a) The establishment of the engineering design of IFMIF, consisting of:

  (i) a complete description of the IFMIF facility including its three main subsystems (the accelerators, the target facility and the test facility), the buildings (including the hot cells for post irradiation examination), the auxiliary systems and the safety systems;

  (ii) the detailed designs of the components, the sub-systems and buildings with specific regard to their interfaces and integration;

  (iii) a planning schedule for the various stages of supply, construction, assembly, tests and commissioning, together with the corresponding plan of human and financial resources requirements; and,

  (iv) the technical specifications for the components allowing calls for tender for the supply of items needed for the start-up of the construction.

(b) The establishment of the site requirements for IFMIF, and performance of the necessary safety and environmental analyses.

(c) The proposal of the programme and corresponding cost, manpower and schedule estimates for the operation, exploitation and decommissioning of IFMIF.

(d) A comprehensive program of validation R&D, in particular:

  i) designing, manufacturing and assembling the prototypical low energy part of one of the two IFMIF accelerators and the first high energy section including its radio frequency power supplies and their auxiliaries, and conducting its integrated beam testing operation;

  ii) designing, manufacturing and testing of scalable models to ensure engineering feasibility of the Lithium Target and the Test Facility;

  iii) constructing the buildings to house the prototypical accelerator and its auxiliary systems.

The EU in-kind commitment is concentrated on the design and R&D in support of the accelerator and the test facility, with specific contributions to the lithium target design and R&D.

**IFERC Programme**

The scope of the IFERC Programme covers R&D activities aiming at contributing to the ITER project and promoting a possible early realisation of DEMO - a future demonstration power reactor.

In particular the following lines of activities are foreseen:

(a) activities of the DEMO Design Research and Development Coordination Centre aiming at establishing a common basis for a DEMO design, including:

  i) seminars and technical meetings,

  ii) exchange of scientific and technical information,

  iii) DEMO conceptual design activities,

  iv) research and development activities on DEMO technologies,

(b) activities of the Computational Simulation Centre (CSC), including the procurement and exploitation of a super-computer for large scale simulation activities to analyse experimental data on fusion plasmas, prepare scenarios for ITER operation, predict the performance of the ITER facilities, and contribute to DEMO design, and

(c) activities of the ITER Remote Experimentation Centre to facilitate broad participation of scientists into ITER experiments, including the development of remote experimentation techniques for burning tokamak plasmas, to be tested on existing machines, such as the Advanced Superconducting Tokamak.

The primary EU in-kind contribution is the super-computer of the CSC.

Chapter

# 2

# Technical Activities in 2007 and 2008

## ITER

### Design Review

Following the signature of the ITER agreement and the establishment of the ITER Organisation (IO), a Design Review of ITER[1] was launched at the end of 2006 by the IO to address a large number of design issues that were recognised by the fusion community to be still unresolved and to update the baseline for the construction project in line with the evolution of the project and R&D.

Europe, first with EFDA and most recently with Fusion for Energy (F4E), has played an important role in the ITER Design Review, contributing from the very beginning to the identification of open problems (see for example report[2] ) and during the review in the proposition of solutions.

The Design Review was performed by 8 working groups (see below), which were chaired by scientists and engineers from outside the ITER team with a co-chairman from the IO.

- WG-1: Design requirements and physics objectives
- WG-2: Safety and licensing
- WG-3: Buildings
- WG-4: Magnets
- WG-5: Vacuum Vessel

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[1] The ITER Design Review (ITER_D_24FMKP v1.0)

[2] M. Gasparotto and the EFDA CSU Garching Team - Outstanding Technical Issues to be addressed by the ITER Design Review Working Groups, December 15, 2006 (EFDA_D_254JCH v1.0)

- WG-6: Heating
- WG-7: Tritium
- WG-8: In-Vessel Components

The working groups consisted of approximately 150 members comprising leading experts drawn from the worldwide fusion community. 80 Professional Person Years (PPY) were contributed from the ITER members (majority from EU) in order to perform more complex studies, design analysis and design work to support the working groups.

It must be said that more than a design review, this was a categorisation of issues. By mid-2007 a total of around 500 issues on the ITER design were registered in the database ("Issue Cards"). In some cases, specific Design Change Requests (DCRs) were formulated, albeit in some cases, rather than design solutions they represented a more detailed description of the problems that appeared in the Issue Cards.

These issues were later reviewed and prioritised, and effort was concentrated on the 13 areas of major technical risk, which had been endorsed by the ITER Scientific and Technical Advisory Committee (STAC[3]).

These included, in particular:

- in-vessel water-cooled copper coils for plasma vertical stabilisation and suppression or mitigation of plasma Edge Localised Modes (ELMs);
- improvement of the plasma shape control capability and extension of the operating space, mainly through an improvement of the poloidal field (PF) coil system (e.g. increase of dimensions of PF6 with the possibility of sub-cooling for an increased current carrying capability);
- reinforcement of the vacuum vessel supports mostly in the lower port areas to withstand higher loads especially during asymmetric vertical displacement events (AVDEs);

[3] The STAC-2 has made suggestions for the investigation of 13 issues in its report, which the IO and the DAs thoroughly studied in the months that followed. In almost all cases, the IO came out with specific proposals for design optimisation solutions.

- reviews of the existing design of the blanket and the first-wall, in which thermal load specifications and reliable remote replaceablity are still uncertain;
- selection of armour material for the first divertor. The STAC endorsed the IO's plan not to start with a tungsten divertor, but instead to begin with carbon targets and to expedite physics and engineering R&D on W divertor targets[4];
- cold test facilities for the final tests of superconducting coils. The strategy proposed by the EU to test each Toroidal Field (TF) winding pack before insertion in the coil cases, and each PF coil at a temperature range of 4-10 K with a level of current sufficient to check the joint resistance would limit the risks of coil failure;
- proposal for the integration of the test blanket module (TBM) into the ITER Research Plan;
- design of the hot cell and, in particular, the area devoted to treatment and intermediate repository of radwaste (mostly class B radwaste, i.e. higher activity and longer life radioactive waste).

The outcome of the work on the 13 issues is in most cases satisfactory, given the constraints on the scope of the work and the timescale in which it had to be done. It must be noted that the main contributions in support to the IO have been provided by the EU and US Domestic Agencies.

However, in several cases the work is not yet complete and in certain areas the design is at an early stage and the requirements are still under definition. Many of the technical solutions that impact the viability of the concept directly need considerable engineering analysis effort and/or R&D effort, and their lead time could have an impact on the schedule. These include the solutions adopted for the in-vessel coils, and the design of the blanket and the first/wall. As far as the internal coils are concerned, it must be noted that the design being considered for ITER is rather innovative and unconventional and that there are some issues that must be addressed by R&D (e.g. the water cooling, the remote maintainability). Similar concerns can be extended to the design of the first wall and blanket shield where the solutions adopted or being considered need to be investigated and fabricability, assembly and remote maintainability need to be demonstrated.

[4] F4E is developing a proposal that postpones the decision to use carbon or tungsten to start operations until the R&D on tungsten targets has been completed and experimentation on JET with metallic plasma facing components will provide the necessary data. This proposal has the advantage that the use of W from the very beginning would avoid the loss of operation time due to a divertor change-out and provide the fastest approach to DT plasmas.

The additional areas where further work is needed are:

- the limiters/first-wall strategy to withstand transient loads during off-normal transients and plasma start-up;
- the development of workable techniques for leak localisation for cryostat and in-vessel components; and
- the overall strategy and the demonstration of methods to control the in-vessel tritium inventory and radioactive dust.

### Cost Assessment Exercise

A task force was established by F4E early summer of 2008 to:

(i) conduct an independent assessment of the resources for the EU in-kind contributions to ITER;

(ii) understand the basis for these estimates, judge their overall credibility and assess the main causes of significant cost growth relative to previous cost estimates; and

(iii) identify uncertainties, high-risk budget areas and potential risk mitigation strategies.

The work of the Task Force was carried out in three months by a group of experts selected on the basis of their technical expertise and broad knowledge in project management. For each of the 16 procurement packages assigned to the EU, the design, the requirements and cost were updated by F4E from the original 2001 status either through a detailed industrial cost assessment or as a result of the F4E internal analysis in collaboration with experts in national fusion laboratories.

By applying a methodology that is deemed to be adequate for a project of this size and complexity, which includes the use of risk cost adjustment factors and cost estimate accuracy ranges, the following conclusions were drawn:

- Significant difference in the cost estimates between 2001 and present;
- Previous cost estimates were based on an insufficient Project Definition Level (PDL) and an incomplete assessment of the fabrication processes;
- On average more than half of the differences between the 2001 estimate and the present estimate can be attributed to internal

factors i.e. factors that are associated with project evolution and are under the project's control (e.g. design and requirement changes, recent results of R&D and assessment of manufacturability, change of QA, testing and inspection requirements) and the rest to cost increase is due to external factors, i.e. factors that are entirely outside of the control of F4E e.g. cost of raw materials and labour increase and interface multiplication among parties. The major impact comes from 5 drivers: Buildings account for about 28% of overall cost growth, magnets (17%), vacuum vessel (9%), in-vessel components (blanket and divertor) (13%), diagnostics (15%) and H&CD Systems (9%).

It must be noted that the present estimates have some limitations: absence of contingencies for the "pricing" of contractual conditions by potential tenderers for liabilities beyond usual practice, warranties, insurance; absence of contingencies for unforeseen events during assembly or commissioning that could require a partial or complete re-fabrication of components; absence of contingencies for maintaining the time schedule in the face of delays in R&D or procurement, and in some cases, profit margins of companies.

These estimates were then reviewed by an Ad-Hoc Group, established by the Governing Board of F4E, in order to provide an independent assessment of resources estimated by F4E. The key findings of this AHG are included in the Toschi Report and can be summarised as follows:

(i) the methodology used by the F4E Task Force is appropriate;

(ii) the AHG agrees with the analysis of the F4E Task Force on the difference in the cost estimates between 2001 and present;

(iii) the AHG recognises that the previous cost estimate was based on an insufficient PDL and an incomplete assessment of the fabrication processes;

(iv) the new resource estimates are up-to-date and the best currently available; and

(v) it also noted the absence of a cost uncertainty assessment in the 2001 cost estimate.

To reduce the level of uncertainty of the current estimates and to contain the cost, the following recommendations were made:

(i) increase the project definition level;

(ii) alternative technical fabrication solutions including R&D and prototyping activities;

(iii) further reliable industrial estimates to improve the overall cost estimate accuracy;

(iv) revisit the current aggressive schedule and explore the optimum trade-off between cost, risk and schedule, with due regard to potential deferrals; and

(v) advocate changes to the procurement sharing in order to minimise complex cross-party interfaces, reduce duplication of R&D and facilities (including possible joint tendering).

### Site and Buildings (WBS 6.1 and 6.2)

**Introduction**

At the end of 2007, the Arrangement for site preparation between The European Commission (EURATOM) and CEA/AIF (Agence ITER France) was signed. It includes the following main activities:

- Earthworks and construction works for ITER platform levelling;
- Design and Construction of Annex Buildings (with the exception of the ITER Head Quarters, to be delivered by F4E);
- Design and construction of the temporary ITER Head Quarters.

In addition, the formal decision to establish a F4E office for the ITER "Site and Buildings" activities at Cadarache was taken by the Governing Board in October 2008.

**ITER Platform Leveling**

Following the clearance and deforestation of the ITER site, which was completed at the end of 2007, the main earthworks commenced in January 2008 and were approximately 90% completed by the end of 2008 (see Figure 9 and Figure 10). 1.700.000 $m^3$ of rocky material was blasted and a total of 2.300.000 $m^3$ of soil was handled. Due to bad weather and a huge unforeseen increase in the quantity of material to be blasted, an additional 4 months are required to complete the platform levelling. In addition, activities related to the heavy haul road, storm basins and the fence infrastructure were carried out during 2008.

**ITER Headquarters**

On Joint Work Site 2 (JWS 2) the temporary Headquarters of the ITER Organisation was constructed (see Figure 11). The contract commenced in January 2008 and the building was handed over to the ITER organisation (IO) in November 2008.

Figure 9 – ITER Platform beginning 2008

Figure 10 – ITER Platform end of 2008

Figure 9 and Figure 10 – ITER Platform before and after levelling

Figure 11 - Temporary ITER Headquarters Nov 2008

Figure 12 - Earthworks for Annex Buildings

In 2007 an Architect Competition took place for the design of the final Head Quarters Office building (Building 72) and other annex buildings (information centre, medical services, site access control building). The Architect Consortium started design works in January 2008 and completed the preliminary design in October 2008.

For the annex buildings the work related to excavation and earth preparation was launched (Figure 12).

**Pre-Architect Engineer (pre-AE)**

A pre-AE Procurement Arrangement was signed in April 2008. This arrangement includes the agreement to carry out a study to define all the requirements from IO with respect to all the buildings. The main deliverables of this work are:

- System Requirement Documents (SRDs) for each of the 28 buildings and for the Tokamak Complex
- Design Description Documents (DDDs)
- System Interface Control Documents (SICDs)
- Drawings - 2D and 3D in Catia
- Cost estimate
- Schedule

A preliminary report was received in March 2009.

F4E considers that this report is sufficient to launch the tender of the Architect Engineer. The final inputs will be received in September 2008 before the signature of the Architect Engineer contract. In the framework of the pre-AE contract, a workshop on Value Engineering was organised in September 2008. Possible cost savings were identified in the areas of:

- Re-configuration of the Assembly Hall (Figure 13)
- Space usage in the Tokamak Complex
- Relocation of the Hot Cell Building
- Optimisation of excavation and ground support structure

A contract will be launched in 2009 to analyse such possibilities in detail.

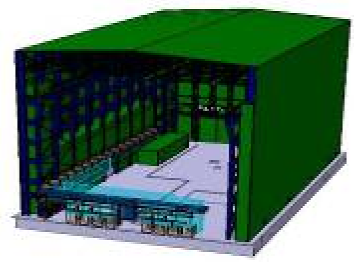

Figure 13 - Assembly Hall

**Poloidal Field Coil Winding (PF-coil) Building**

The PF coils will be built on the ITER site due to their large dimensions (see Figure 14). The construction of this building is on the critical path and should be completed in 2011. The main features are:

- Area of the building 250m x 37m
- 2 cranes (bridge or gantry) located inside the building
- Special temperature and dust control

The PF coil building PA was signed in November 2008.

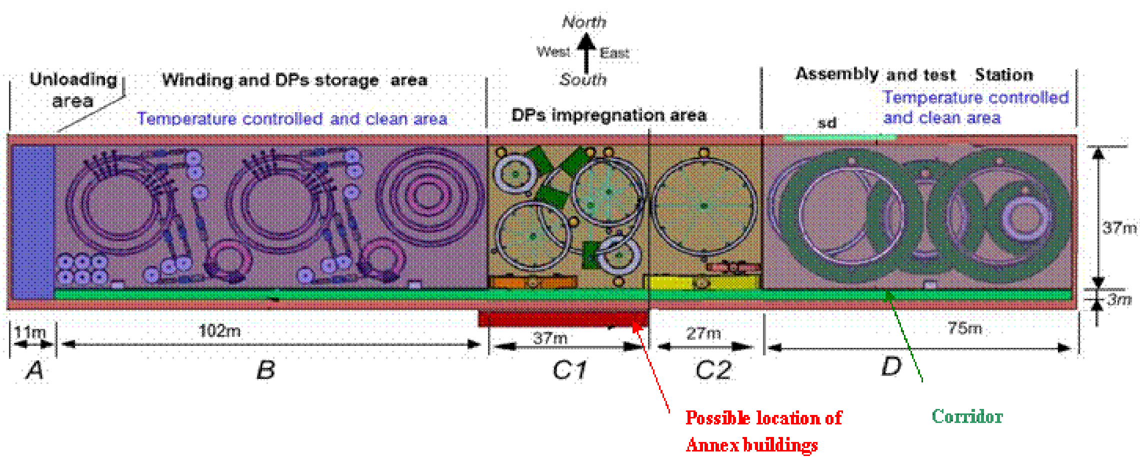


Figure 14 – Schematic of the PF coils construction areas

## Magnets (WBS 1.1)

The Magnet Group of F4E has started the procurement of the ITER magnet components, with the first call-for-tenders for 10 Toroidal Field winding packs and 27 superconductor lengths to be provided by Europe.

Following the signature of the Procurement Arrangement for the TF conductors in December 2007, the PA for the TF coils has been successfully reviewed and signed on 20 June 2008. Procurement strategies and preliminary risk assessments have been developed. The PA for the PF conductors has also been prepared, while reviews of the PF coil draft PA and the one for PF coil winding building, signed by the Site & Building Group in November 2008, have been carried out.

The call-for-tender for 62 tons of Cu strand started in mid-March and was completed in December 2008, while the one for 95 tons of $Nb_3Sn$ strand started in July. Pilot quantities of strand from four potential suppliers and short cables made by two suppliers according to the latest ITER design (option 2) have been produced (Figure 15) for three new TF conductor qualification samples to be tested in the Sultan facility, plus another sample for the so-called “back-up solution”. A 14-month long grant for testing of conductor samples in Sultan has been awarded in October 2008.


Figure 15 - New TF conductor sample - option 2

A contract for the production of the TF & PF conductors has been drafted, pending bi-lateral agreements with the USA Domestic Agency for jacketing of their TF lengths and with the Russian Federation Domestic Agency to exchange NbTi cables for jacketing of all PF1 & PF6 conductor lengths.

The procedure for the selection of suppliers of the TF winding packs started in October, together with the preparation of tender documents for the procurement of two radial plate prototypes. Bi-lateral discussions started with JADA to establish synergies for the TF coil manufacture and reduce schedule, costs and risks.

Several R&D activities started under the EFDA technology programme have been completed: a TF coil cross section mock-up has been successfully impregnated with the high radiation resistance newly developed cyanate-ester & epoxy resin blend (Figure 16); the PF Conductor Insert has been successfully assembled and tested in June-August 2008 in the CSMC facility at Naka (Japan) and confirmed the design performance of the proposed conductor for the ITER PF coils (Figure 17); pre-compression ring mock-ups and base material specimens have shown that the selected glass-epoxy composite can reach adequate structural performance against relaxation, creep and sufficient ultimate strength (Figure 18).

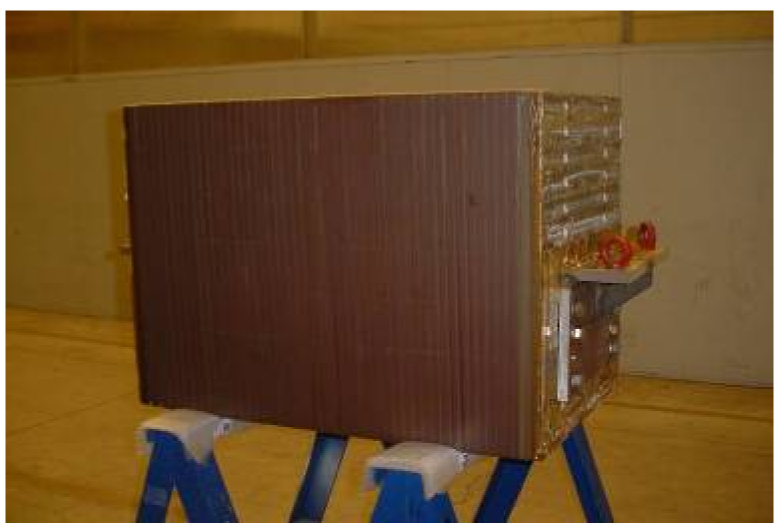

Figure 16 - Mock-up with cyanate-ester resin

Figure 17 - PF Insert before installation in Naka

Figure 18 - Pre-compression ring mock-up after rupture

Other relevant R&D on welding and non-destructive testing techniques for conductor jackets and coil structures is in progress. Out of the 32 EFDA technology tasks and contracts which were transferred to the F4E Magnet Group, 17 were completed in 2008.

During 2008 the F4E Magnet Group engaged in reviews of the detailed manufacturing schedule and costs of the EU magnet components.

### Vacuum Vessel (WBS 1.5)

Early in the year ITER decided to add the ELM/VS coils (see chapter on Design Review) to the baseline design. These coils are mounted directly onto the inner shell of the Vacuum Vessel and extended the work required before starting the Procurement of the 7 Sectors to be delivered by Europe.

A Call for Expression of Interest for VV Sectors, issued in the middle of the year, led to a successful qualification from 4 European companies and 3 Consortia. A technical information meeting at the end of December was attended by most of the Qualified Companies.

Figure 19 - Sector Mock-up

The VV mock-up using conventional welding was completed successfully (see Figure 19), managing to achieve the required tolerances. At the same time, the Electron Beam method progressed (see Figure 20), showing very low distortions and it will be used for the inboard part of the sectors. Several tasks were underway to develop improved manufacturing techniques, including explosive forming. F4E made contributions to critical R&D for the Assembly of the sectors, including a successfully demonstration to deliver the shielding gas to the rootside of the VV sector field weld and a hydraulic weld/cut robot (see Figure 21).

Figure 20 - Electron Beam Welded Sector Mock-up

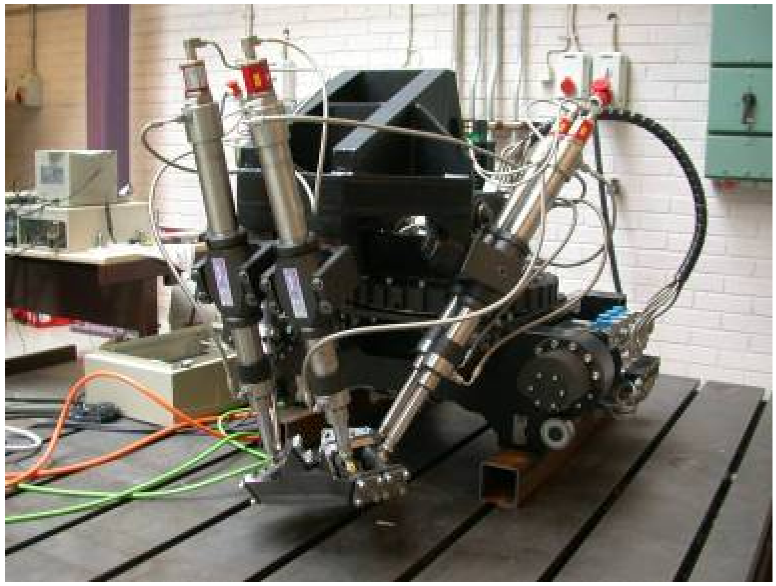

Figure 21 - Sector Welding Robot

### Blanket and Divertor (WBS 1.6 and 1.7)

**Blanket area**

With a signature of the Procurement Arrangement (PA) for the ITER Blanket system foreseen in 2011, the Blanket activities in the In-Vessel group concentrated on support of ITER design activities, such as the First Wall (FW), Shield and manifold systems, and on the continuation of key R&D issues.

Figure 22 - FW panel prototype

For the manufacture of the ITER FW, the joining of the Beryllium (Be) armour material to the Copper Chromium Zirconium (CuCrZr) heat sink alloy is the most delicate fabrication operation. This joint, although exceeding the present ITER FW design requirements, is at present the limiting factor of the FW performance, with a detachment of Be tiles at about

3 MW/m$^2$. Therefore, the development of the Be/CuCrZr alloy joining by Hot Isostatic Pressing (HIPping) has continued at CEA Grenoble to further increase the engineering margins in view of the likely future increase of the ITER FW design requirements. This includes optimization of the joining conditions, manufacture and testing of FW mock-ups and preparation of neutron irradiation experiments.

FW mock-ups were thermal fatigue tested at ENEA Brasimone and successfully achieved 30,000 cycles at 0.6 MW/m$^2$ without any indication of failure. These mock-ups will be HHF tested at higher values to assess the performance limits. The HIP joining of the Be tiles on a fourth full-scale FW panel prototype (Figure 22) was successfully completed by AMEC NNC Ltd. The first EU-DA FW qualification mock-up was successfully tested at the electron beam test facility of Sandia National Laboratory. (USA), in the framework of the ITER FW qualification programme, for the required 12,000 cycles of 96 sec at 0.7 MW/m$^2$ plus the subsequent 1000 cycles at 1.4 MW/m$^2$. Additional tests were performed to assess the limit of the Be/CuCrZr HIP joints and tests were performed for 200 cycles at 1.7 MW/m$^2$, which was the maximum value acceptable for the test facility to keep the Be temperature below the allowable limit. No indication of failure could be observed. The mock-up will be sent back to EU for subsequent ultrasonic testing and HHF testing.

The EU FW qualification test facility (BESTH) at the Nuclear Research Institute (NRI, Czech Republic) for testing medium scale FW mock-ups has been completed and commissioned. Thermal fatigue testing of the two first ITER FW qualification mock-ups, one from the USDA and one from the EUDA (Figure 23), was completed. Both mock-ups were tested for the required 12,000 cycles of 5 minutes at 0.62 MW/m$^2$. Ultrasonic testing of the Be/CuCrZr interfaces performed after thermal fatigue testing did not reveal any indication of failure on the EU mock-up. The mock-up was sent to the JUDITH2 test facility at the Forschungszentrum Juelich (Germany) for the second part of the qualification programme at higher heat flux.

Figure 23 - EU FW Qualification mock-up after test at NRI

**Divertor area**

The main objectives of the divertor activities during 2008 were to complete the qualification of the EUDA for the divertor procurement and to start the preparation of the Procurement Arrangement with the IO for the procurement of the Inner Vertical Target.

The manufacture of the divertor qualification prototypes, which started in 2007 in accordance with the ITER procurement schedule, was completed by the two selected companies, namely Plansee SE and Ansaldo Ricerche SpA. These prototypes include all the key features of the inner vertical targets and have Carbon Fibre Composite (CFC) and tungsten (W) armoured plasma facing component (PFC) parts. Three qualification prototypes had to be manufactured (Figure 24): one full monoblock design and one mono-flat tile design by Plansee plus one full monoblock design by Ansaldo. All prototypes passed the acceptance criteria during manufacture and were sent to the TSEFEY-upgrade electron beam test facility at the Efremov Institute of St Petersburg (Russian Federation) to be high heat flux (HHF) tested in the framework of the ITER divertor qualification test programme.  The testing should be completed during the first quarter of 2009.

Figure 24 - Divertor Qualification Prototypes

More than 100 divertor mock-ups with artificial and calibrated defects manufactured by Plansee and Ansaldo were ultrasonic tested at CIEMAT, SATIR infrared tested at CEA and finally HHF tested in the AREVA-FE200 test facility. The outcome of the above tests constitutes the core of a database used for the definition of the divertor PFC acceptance criteria.

The development of repair techniques for CFC and W monoblocks is still in progress. It includes the manufacture of both reference and repaired CFC and W monoblock mock-ups. These mock-ups will be HHF tested to assess the thermal fatigue performance of the repair techniques.

The assembly and hydraulic tests of a full-scale divertor cassette (one cassette body plus dummy target and dome components) has continued at ENEA Brasimone. The prototypes were integrated (Figure 25) and are being hydraulically connected at the Divertor Refurbishment Platform facility. They will then be hydraulically tested in the ENEA CEF1 hydraulic test loop.

Figure 25 - Integrated divertor cassette

At the Forschungszentrum Juelich, destructive examination of the vertical target near full-scale prototype was completed and the thermal fatigue testing of small-scale divertor components has continued in the JUDITH test facility.

In the framework of the qualification programme of alternative CFC grades for the PFCs, the mock-up fabrication has been completed at Plansee and Ansaldo, their thermal fatigue testing will be launched in 2009. Also, small-scale mock-ups have been manufactured at ENEA Frascati by using an alternative tungsten grade.

Tentative procurement strategies for the Divertor system have been developed and a preliminary risk assessment has been performed.

### Remote Handling (WBS2.3)

During 2008, the Remote Handling Group of F4E worked in the following areas:

- definition of the overall schedule and programme of activities for F4E RH, in agreement with the ITER schedule milestones and in compliance with the F4E financial constraints;

- implementation and completion of all those already running design and test activities, related to the European contribution to ITER RH, which were launched under EFDA;

- start up of the new tasks to be performed in preparation of the Procurement Arrangements that will come in some years from now, in harmonisation with what done so far (EFDA tasks) and according to the overall schedule; and

- definition of the human resources needed for bringing the RH group to its full capacity.

In particular, the following achievements are reported:

- a Grant has been placed with TEKES (Finland) in order to operate the Divertor Test Platform 2 (DTP2) facility (see Figure 26 and Figure 27) basically from end of 2008 to first half of 2010. The facility is now equipped with a DIV RH prototype, namely the Cassette Multifunctional Mover (CMM) that was delivered in October 2008, and is being/will be used to test the divertor cassette in-vessel handling sequences, in conditions closer and closer to those of ITER (control room, remote operations, virtual reality etc.). The operational scenario will be enriched during 2009 with the use of a Manipulator Arm combined with tooling for cassette locking system and a sliding table for further use of it on board of CMM. The Grant placed covers also the design and specification of the next DTP2 upgrades/extensions in view of further industrial procurements. These activities are in support/preparation of the future PA.

Figure 26 - Installation of the Cassette Multifunctional Mover prototype in the test stand in DTP2 - Finland

- beneficiaries have been selected for another two Grants, related to Transfer Cask System (TCS) and In-Vessel Viewing System (IVVS) design and preparation of future tests in ad-hoc facilities(in the IVVS case, the design is supported by the use of an existing laboratory device for support tests). These activities will be important in terms of arriving to a viable conceptual design of both systems, and in defining the requirements for a test programme with ITER-relevant prototypes, before arriving to the PA.

- EFDA tasks have been completed in the following areas: DIV RH (in particular those preparing DTP2 operation covered by the Grant mentioned above); Neutral Beam Remote Handling conceptual design; Neutral Beam Duct Liner conceptual design (including design study of RH devices); Transfer Casks studies (control system, rescue scenarios); Blanket First Wall conceptual design; studies on leak detection and localisation.


Figure 27 - Manipulator arm in a test stand in DTP2

**Vacuum Pumping and Fuelling (WBS 3.1)**

The ITER machine requires high vacuum for its operation. The cryopumps are the key components for producing and maintaining the vacuum in the torus, cryostat, and Neutral Beam Injectors. They contain 4K (-269° C) cooled surfaces and/or sorption agents for helium trapping, and in this way all gases are effectively pumped. Their operation is based on the well known observation that gases and vapours are bound and held on cold and charcoal coated surfaces. In ITER, the cryopumps will be fed with supercritical helium at very low temperatures of approximately 4K, via a distribution system incorporating Cold Valve Boxes (CVBs) and cryolines. There are 8 torus, 2 cryostat and 2 NBI cryopumps.

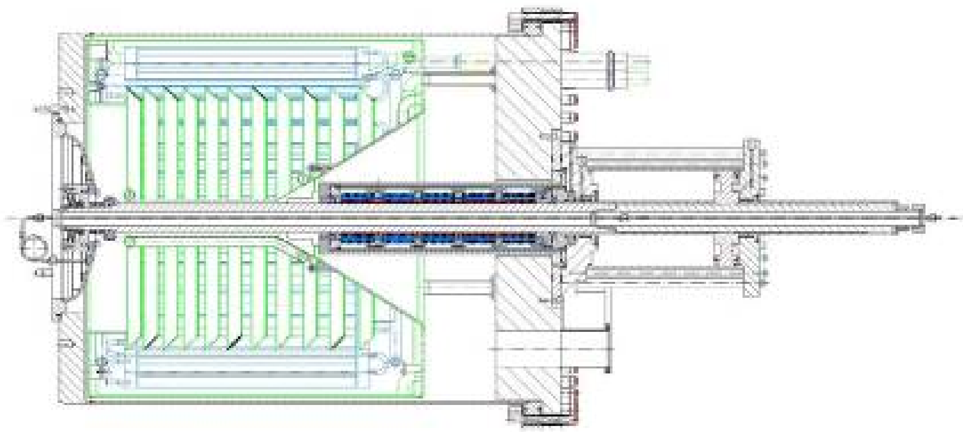

Figure 28 - Cross section of torus or cryostat cryopumps

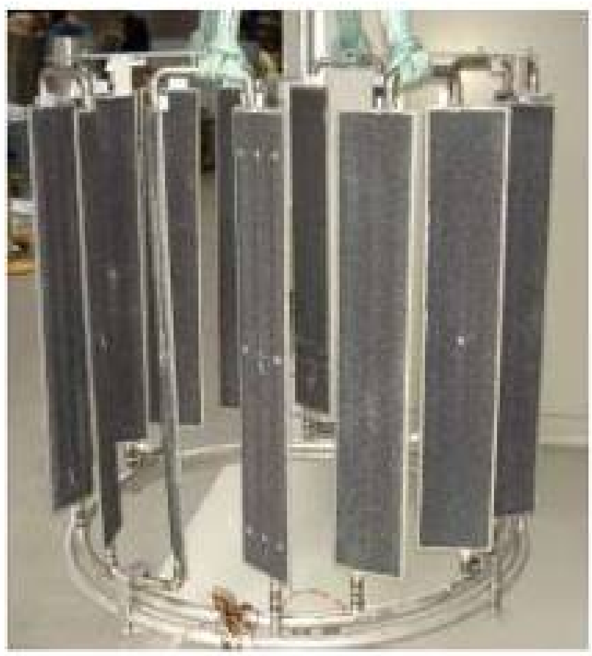

Figure 29 - Cryopanels of the model pump

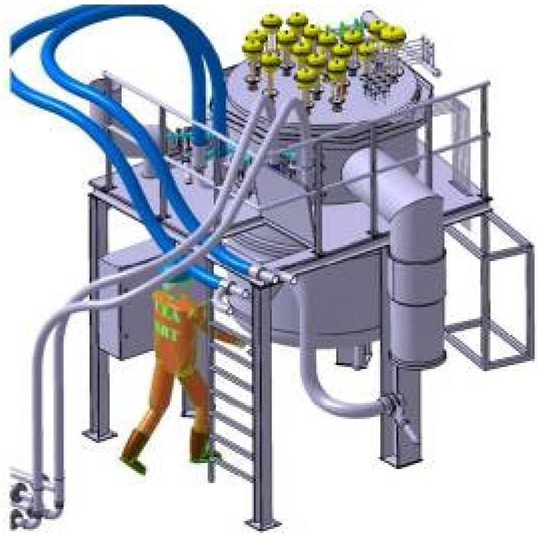

Figure 30 - View of a Cold Valve Box

The design of the torus and cryostat cryopumps (Figure 28), now in its final stages, is a novel and very demanding engineering challenge, which, in addition to the pumping surfaces, incorporates an integral vacuum valve. For this reason, a half-scale model pump has been produced, demonstrating effective and reliable pumping performance. Figure 29 shows a key part of this model pump, namely the pumping surfaces of the cryopanels coated with charcoal which acts as the adsorption agent. Following the completion of the final design of the torus cryopump, build-to-print drawings and detailed specifications will be produced. This will allow the manufacture and extensive testing of a Pre-Production Torus Cryopump (PPC) in order to define with a high degree of confidence, the procurement packages for the in-time delivery of the pumps to ITER.

Similarly, the detailed designs of the CVBs and cryo-jumpers (special pipes for the supply of cryogens) are progressing (Figure 30) in order to fully specify the relevant procurement packages.

**Tritium Plant (WBS 3.2)**

The tritium plant is essential for the operation of the tokamak; especially after the initial hydrogen phase, since tritium will be produced in D-D reactions and is one of the two fuel elements during the D-T phase. The main functions of the tritium plant are: i) to process all tritiated gas streams independently of their origins, ii) to produce the gas streams required for fuelling at the flow rates and isotopic compositions specified by the operational requirements and iii) to detritiate all gaseous waste streams to the very low levels required before their release into the environment.

The EU is committed to procure two subsystems of the Tritium Plant: i) the Water Detritiation System (WDS), where tritium is transferred from water molecules into hydrogen and ii) the Hydrogen Isotopes Separation System (HISS) where from the various injected hydrogen mixtures the specific isotopic streams are produced as required for operations.

In the Tritium Plant area, the work in support of ITER IO for the production of the documentation required for design and safe operation is on-going and has resulted into various contributions. At the Tritium Laboratory Karlsruhe, the ITER relevant scale Isotope Separation System test loop (Figure 31) and the Water Detritiation System experimental loop (consisting of an electrolyser specifically adapted for tritium compatibility and of an 8 m high Liquid Phase Catalytic Exchange -LPCE- column) were operated extensively in isolated modes and are now ready for combined/integrated operation studies (TRENTA facility) to allow full optimization of the two systems. In this way the otherwise expected chronic tritium releases by HISS can be drastically reduced.

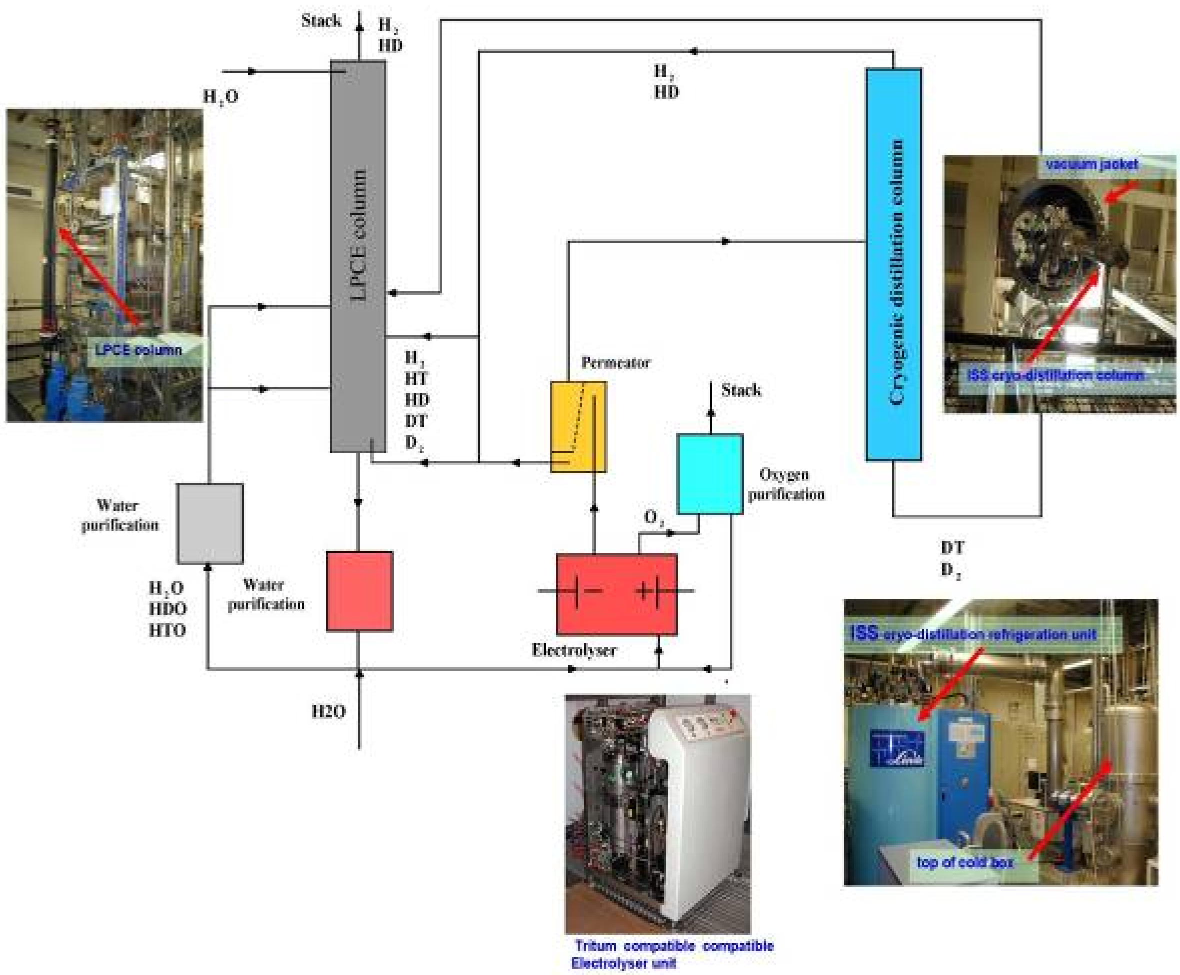


Figure 31 - Schematics of the TRENTA facility showing integration of WDS ISS

### Cryoplant (WBS 3.4)

In ITER and future fusion devices the correct functioning of very important components such as the superconducting magnets and cryopumps relies on the supply of cryogens at the requested temperatures and flow rates by the cryoplant. The ITER cryoplant will be one of the largest and most complicated ever built due to the high variable heat loads and the complex distribution system.

During 2007 and 2008 the cryogenic design activities were organized in two main tasks covering the cryoplant design (CRYO2) and the cryogenics system functionality (CRYO3). The design activities focused on the conceptual design and layout of the cryoplant. The objective was to analyze the cryoplant conceptual design, its technological choices and main sub-systems, define the layout and site integration requirements for installation, commissioning and operation and, finally, review cryoplant procurement schedule and cost assessment. This task was performed in close collaboration with CEA through two industrial support contracts with the two big European cryogenic systems suppliers (Air Liquide and Linde). Both contracts started early 2007 and were closed at the end of 2007 and beginning 2008. The conceptual design of the cryoplant and the technological choices for its main sub-systems were proposed, as well as alternative solutions and improvements. The complete layout of the cryoplant building (Figure 32) including space, handling, transport and logistics, utilities, storage and operational requirements was identified. Finally, the procurement schedule and cost assessment of the cryoplant system was validated by industry.

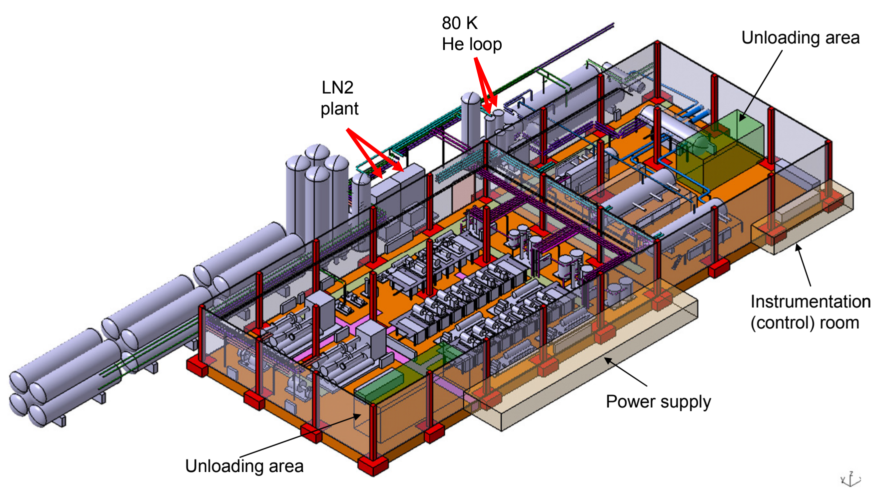


Figure 32 - Proposed layout for the ITER cryoplant system

The activities related to the cryogenics system functionality focused on functional modes analysis, the development of a project plan and the detailed design of a cryogenic test loop. The CRYO3 task was performed by a contract with CEA, starting in February 2007 and finishing in September 2008. The outcome of this task has been very positive and has provided ITER with excellent tools to continue and finalize the definition of the ITER cryogenic system. A wide range of operational, transient and abnormal modes were analyzed, producing associated process flow diagrams for the overall system. A baseline document for the Project Plan was produced including:

- Introduction and present status of the ITER cryogenic system project;

- Project description and analysis based on technical and managerial issues;

- A preliminary risk analysis for the ITER cryogenic system;

- A Quality Control analysis;

- A schedule and resources loaded plan to develop the project.

- Finally, a detailed design for a cryogenic test loop (Figure 33) has been proposed for the study and validation of the mitigation strategy of the ITER variable heat load. Test program, planning and budget have also been defined.

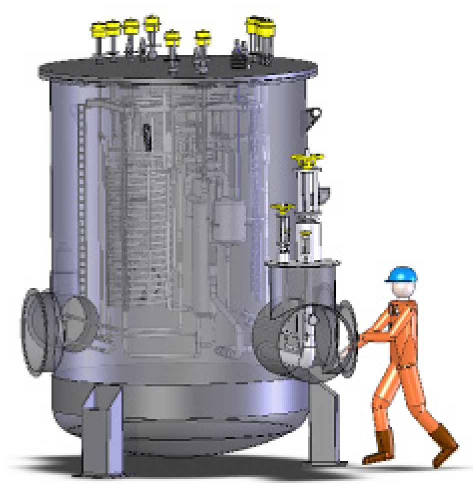

Figure 33 - Artists impression of the proposed cryostat for the cryogenic test loop

Future activities on cryoplant will concentrate on:

- R&D and support activities for ITER: cryogenic test loop, numerical flow-dynamic calculations and dynamic simulator of the cryogenic system;

- R&D in support of cryoplant design review for the optimization of cryoplant design and update of process flow diagrams and functional modes analysis;

- Preparation of cryoplant procurement package: cryogenic manual, preparation of technical specifications, study on $LN_2$ plant, 80 K loop and gas management systems.

### Power Supplies (WBS 4.1 and 4.3)

The ITER project comprises a complex electrical power supply system, which has to supply both the tokamak, the Heating and Current Drive (H&CD) systems, the ancilliaries and services for buildings. Since the supply to the tokamak for the poloidal coils and the additional heating system varies very rapidly, it generates the need for significant "reactive power compensation" to comply with the requirements of the electricity company.

The main systems (Tokamak amd H&CD) require a total active power of 500 MW on the grid. Furthermore, a Reactive Power Compensation and Harmonic Filtering system of 750 MVAR will assure compliance with grid operator's requirements for maximum reactive power consumption.

The auxiliary systems include all buildings with their associated Heating, Ventilation and Air-conditioning (HVAC) systems, the cryoplant and the water-based cooling system. In total, 120 MW are required from the grid, of which half will be for the Cooling Water system and one fourth for the cryoplant and cryo-distribution. Safety loads will be supplied from the emergency power supply diesel-generators sets with a total active power of 14 MW.

### CODAC (WBS 4.5)

CODAC is the COntrol and Data ACquisition system of ITER. It comprises the computer systems, networks, data acquisition and instrumentation. The overall architecture and the managerial organization are indicated in Figure 34. The CODAC system, itself, consists of the computers, networks and the interfaces to the physical plant. This is the responsibility of and funded directly by the ITER IO. The rest of the control and instrumentation, in the plant systems, is the responsibility of the suppliers of the plant itself and is specified by the plant Responsible Officers and is funded partly out of the Procurement Arrangements and partly form the IO.

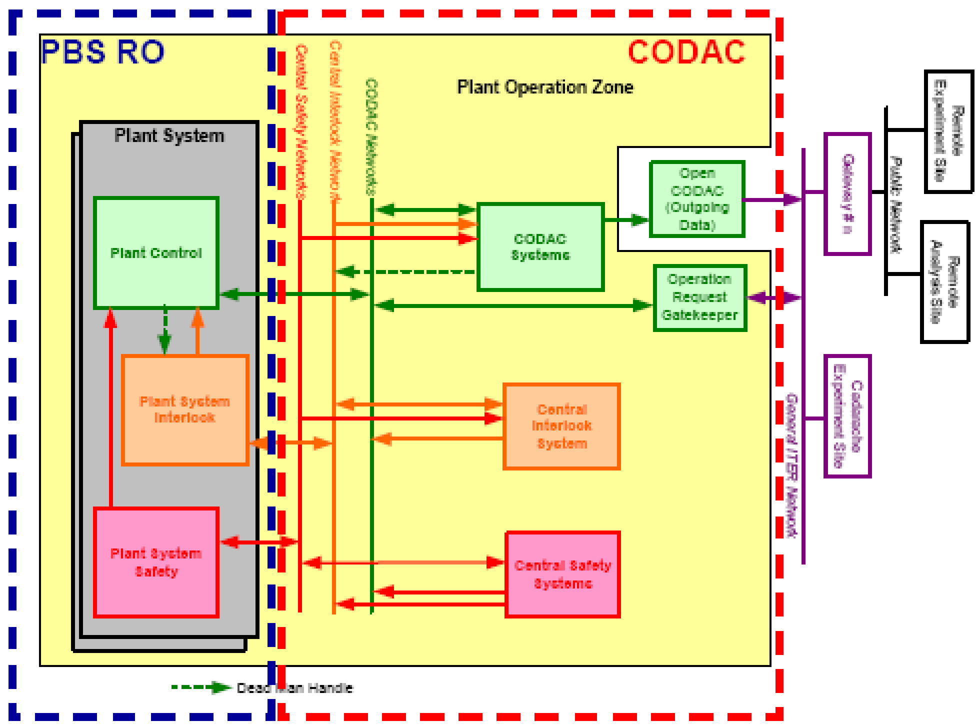


Figure 34 - Control and data acquisition architecture, showing the breakdown in management responsibility between CODAC (red) and the plant system RO (blue). The “CODAC system” comprises the blocks in green and the networks are indicated by the vertical red, amber and green lines.

The EU has been closely involved in the preparation of the structure of the control and acquisition system, the specification of the individual elements of the system and the development of the CODAC standards, both through individual participation on the ITER site and participation in the various working groups that support the central team. In addition, EU industry has played a significant role through contracts placed by the IO. The split in responsibility requires careful management in order to ensure that the plant control and instrumentation complies with ITER standards, is maintainable, economic and “future proof”. To this end, the overall system will be based on standard, industrial components, wherever possible.

As well as assembling demonstration systems, the IO has been preparing a catalogue of standard components and the “Plant Control Design Handbook”. This latter describes the overall system in detail and forms the basis for the application of ITER standards in this area. The relationship between suppliers of plant systems and CODAC will be defined by this handbook. On 27-28 October 2008, a colloquium was held both to discuss the Plant Control Design Handbook and the EU procurement strategy, with EU industry, Fusion Associations and members of the IO CODAC team. F4E participated by presentations of its procurement strategy and case studies of plant systems for which it is responsible. The meeting provided useful feedback to both the IO and F4E, which has helped to develop a coherent strategy for the management of ITER Control and Instrumentation and its procurement.

### Heating and Current Drive Systems (WBS 5.1, 5.2 and 5.3)

**NB system**

Europe continues to provide essential support to IO in all the activities related to the Neutral Beam Heating and in the implementation of the design changes decided during the 2007 ITER Design Review. The activities during 2008 were mainly focused on the design of the components of the first ITER injector and of the sub-systems of neutral beam test facility on experiments on high voltage vacuum insulation. The procurement of the ITER NB injectors (Figure 35) must be accompanied by a substantial R&D effort centred on the establishment of a full scale test facility (Figure 36).

The negotiations amongst the Parties and ITER Organization (IO) for the establishment of the test facility in Padua are proceeding in parallel to the technical design activities aimed at progressing the preparation of procurements. In 2008, the design of the NB test facility was essentially finalised. In particular, EU provided substantial support to IO for the design review of the ion source test facility (Figure 37) which was successfully completed. In addition to the progress of the electrical and mechanical design of all main components, specific efforts have been devoted to perform the design modifications requested by the ITER International Team. The first NB Procurement Arrangement covering the power supplies was prepared in co-operation with IO and it will be signed during the first half of 2009.

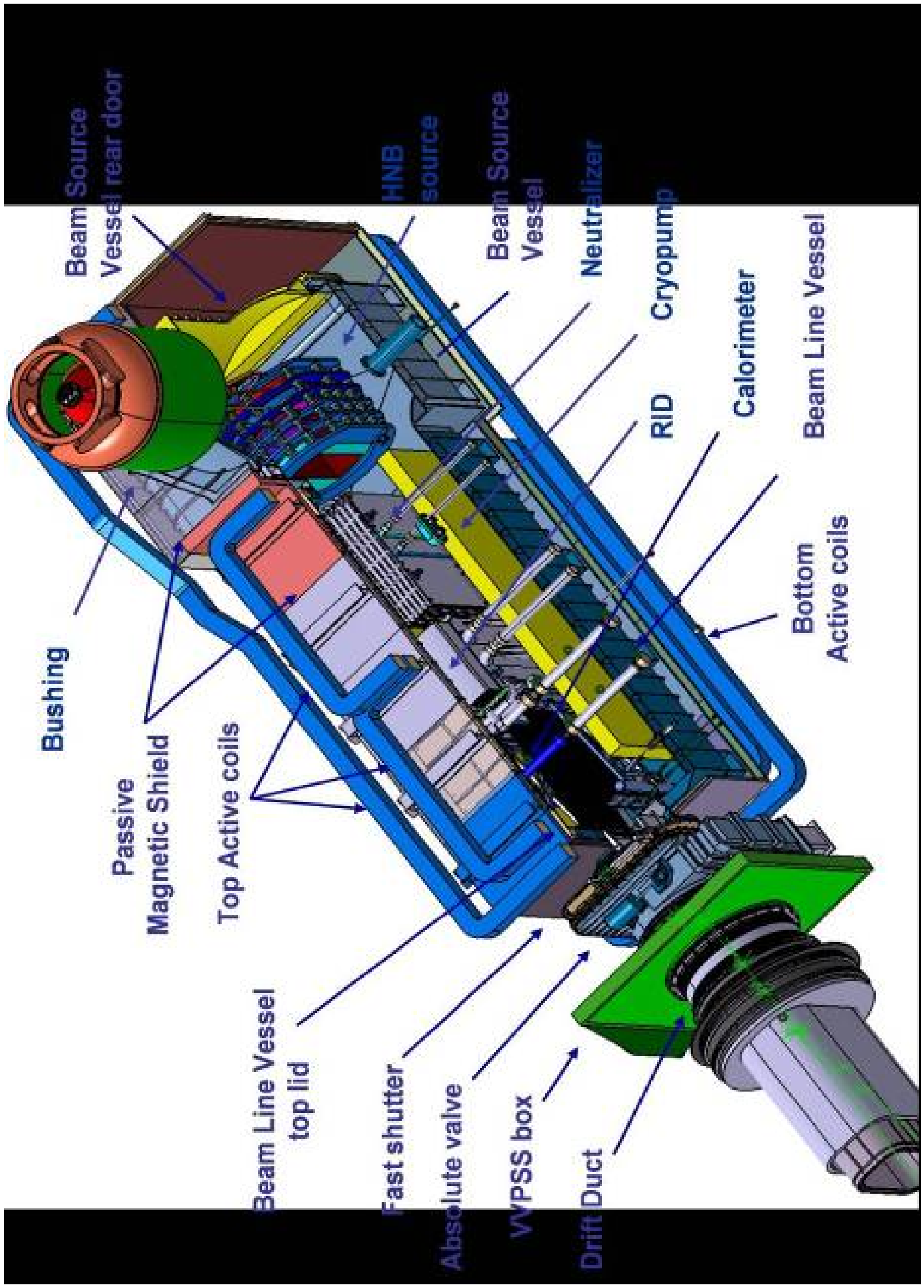


Figure 35 – ITER Heating and Current Drive Neutral Beam Injector

Figure 36 – Artists Impression of the Neutral Beam Test Facility Buildings

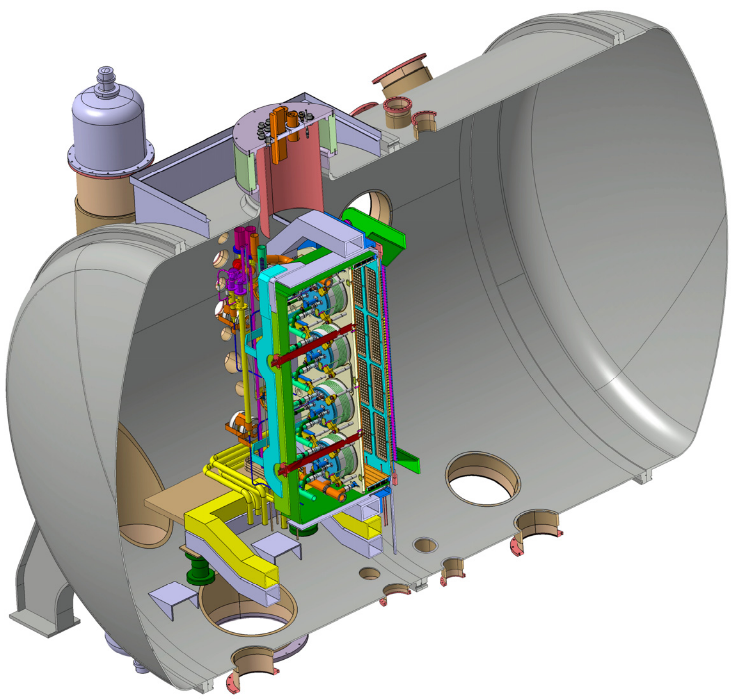

Figure 37 – Ion Source Test Facility: Full scale ion source and vacuum vessel

**EC Power Sources and Power Supplies**

The first European 2 MW gyrotron prototype for ITER, based on coaxial cavity geometry, was tested at CRPP-Lausanne, in the period December 2006-September 2008. It was proved that the operation was stable at 170 GHz in the desired mode and 1.4 MW of RF power was achieved for a few milliseconds, a good agreement was found between the experimental results and the model predictions, the collector power handling capability was successfully proved (2.4 MW for 2s, and at 2MW using 10s pulses).

All new infrastructures and services of the European Electron Cyclotron Test Facility established at the CRPP (Switzerland) under EFDA support operated successfully (Figure 38). The main reasons limiting the performance of the first gyrotron prototype were identified and modifications were proposed for implementation in the refurbished gyrotron. New electron gun, beam tunnel and mode converter system were designed and are expected to substantially improve the gyrotron RF performance.

Figure 38 - EC Test Facility in CRPP-Lausanne and gyrotron under test

The short pulse pre-prototype coaxial cavity gyrotron at the FZK, Karlsruhe, was tested at the end of 2008 and beginning of 2009, for the first time with the nominal magnetic field at the cavity, and with some improved internal components. An output power of up to 1.8 MW was achieved in single mode operation (going from ~71 to 91kV or equivalently from ~0 to 1.8MW

RF power), Figure 39. Additional tests are planned in the near future to check the performance of the gyrotron with a much improved design of the mode converter system. The contract for the gyrotron refurbishment will be placed in 2009.

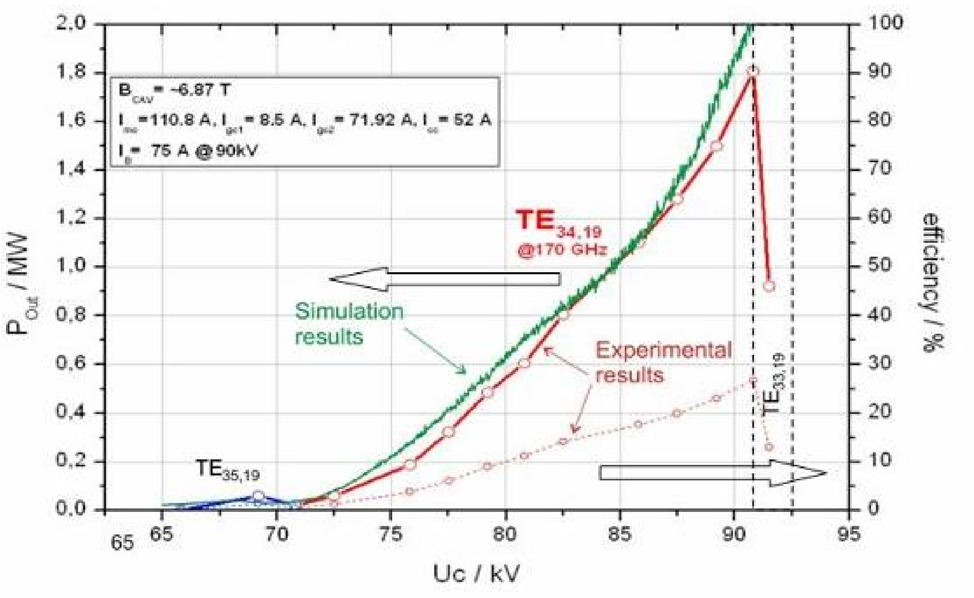


Figure 39 - Output power versus beam voltage as obtained from recent experiments with the FZK pre-prototype coaxial cavity gyrotron

The main power supply for the EC test facility (Figure 40), procured under an EFDA contract, was successfully factory tested and then installed and commissioned at CRPP at beginning of 2009.

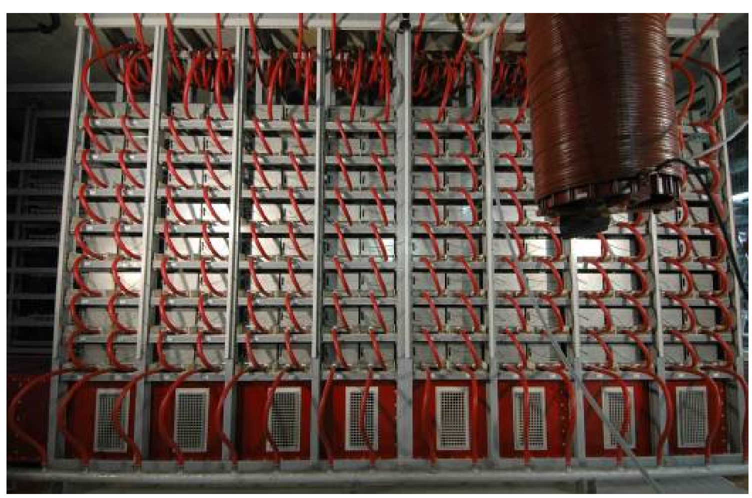

Figure 40 - Gyrotron main power supply installed at CRPP, Lausanne

**IC Power Sources**

The design and prototyping testing of the EC Upper Launcher has continued, and the final optical configuration as been selected and analyzed. Integration of mm-wave components into the port plug structure has been accomplished. The design solutions developed for the EC launcher for the Blanket and the port plug structure have been proposed to ITER as a possible generic Upper Port plug Design. A model of the latest Upper Launcher design is shown in Figure 41.

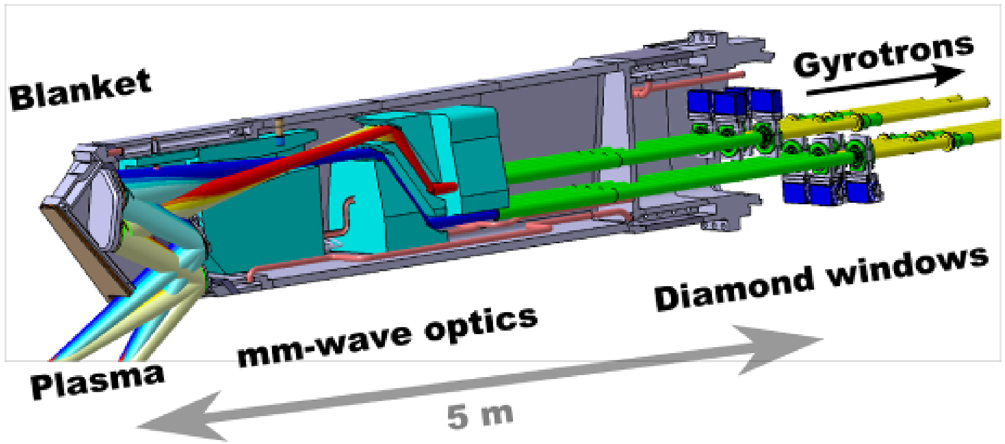


Figure 41 - The EC Upper Launcher for ITER

The 2007 ITER design review working group on Heating & Current Drive design guidelines of the ITER ICH antenna implied substantial changes in the design compared to the 2001 baseline. In the new concept, the matching system is external to the port plug. The new conceptual design, shown in Figure 42, also incorporates a removable Vacuum Transmission Lines (VTL) to allow RF windows and diagnostics to be removed in situ from the rear. 2008 has seen the preparation of the design and R&D activities that should start in 2009 and that are required for completing the antenna built to print design.

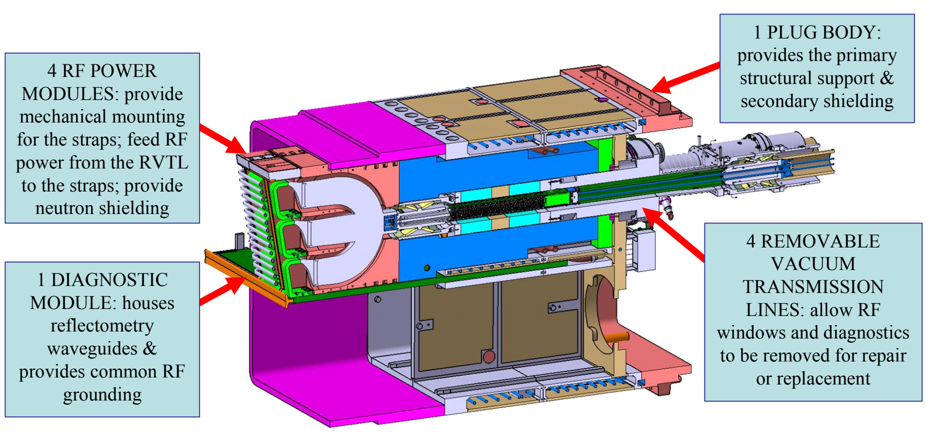


Figure 42 - The ICH Antenna conceptual design in 2008

## Diagnostics (WBS 5.5)

The diagnostics for ITER are measurement systems which will provide the data making possible not only the scientific and technological exploitation of ITER but also control of the plasma and protection of the machine from damage. F4E is responsible for coordinating the R&D, design and supply of more than a quarter of the ITER Diagnostics, providing some of the most important measurements for ITER operations (e.g. magnetic fields; plasma density and temperature; plasma-wall gaps and interactions; radiated power; and helium ash density).

2008 saw the successful conclusion of several diagnostic design activities begun under EFDA Grants, as well as the launch of F4E's first Calls for Proposals in this area. These activities have been undertaken both in Fusion Associate laboratories across Europe and in industry. Progress has been made on many fronts and credible conceptual designs now exist for most systems.

Diagnostics mounted on the vacuum vessel, such as magnetic field sensors and radiated power detectors (or 'bolometers') are among the most critical since they will be installed early in the ITER construction schedule, must withstand the most severe environmental conditions and cannot easily be maintained or replaced.

In some cases, technologies new to the fusion environment have had to be exploited. For example high-frequency magnetic field sensors made of stacked ceramic plates with printed metal tracks (so called 'low-temperature co-fired ceramic', or LTCC, technology - Figure 43) have been tested during 2008 as one way to avoid problems encountered with the manufacture of conventional 'wire-wound' coils. High-frequency magnetic sensors will be used on ITER for studying and controlling certain types of plasma instability, which can reduce the plasma performance and potentially result in damaging disruptions.

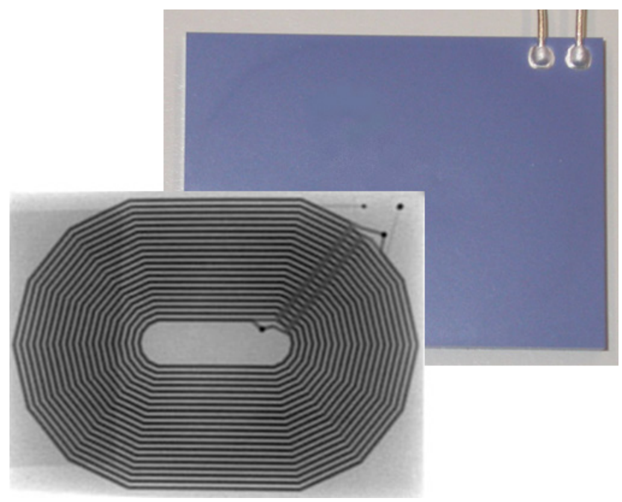

Figure 43 - LTCC high-frequency coil in the visible (background) and X-ray (foreground), showing internal conductor windings.

Another in-vessel diagnostic for which substantial progress was made in 2008 are the gauges used for measuring the gas pressure in the divertor region, where hot plasma meets material surfaces and is neutralised to form deuterium, tritium and helium gas. Pressure gauges are fundamental tools on all tokamaks, but engineering them to survive in the harsh ITER environment and to meet the ITER measurement requirements is a significant challenge. An existing design used on the ASDEX Upgrade tokamak has, however, shown considerable promise. Improvements suggested by 3D Monte Carlo and finite element modeling of the gauge operation have now been tested on a prototype. The adapted gauge seems capable of measuring pressures up to 30 pascals at a magnetic field of 6 teslas, exceeding the ITER measurement requirements Figure 44.

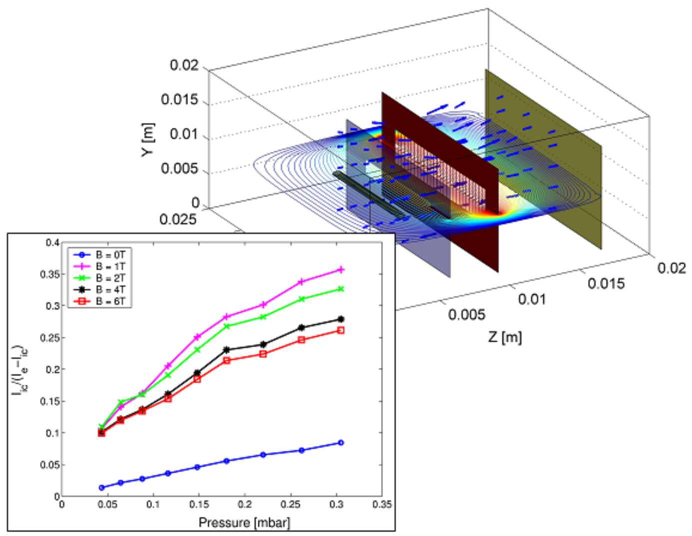


Figure 44 - Results of 3D modelling of 'ASDEX-type' pressure gauge and test results on adapted prototype showing good response over the required range, at ITER-like magnetic field.

The last year has also seen the culmination of several studies related to design and integration of diagnostics based in the ports of the ITER vacuum vessel, such as the core plasma LIDAR Thomson scattering system; the visible/IR wide-angle viewing system; the radial neutron camera; and the core plasma charge exchange recombination spectroscopy system. The studies focused on demonstrating feasibility; identifying critical components; and addressing the overall design to the level needed for a Project Plan. As well as addressing diagnostic-specific issues, they included development, optimization and corresponding engineering analysis of the port-plugs (Figure 45): water-cooled stainless steel structures housing the diagnostics and radiation shielding, and closing the ITER vacuum.

One of the principal challenges addressed was to show how complex, multi-component diagnostic systems could be successfully integrated with the radiation shielding, in a way which allowed for the possibility of maintenance whilst ensuring that neutron radiation behind the port-plug was reduced to acceptable levels. As if not difficult enough for a single diagnostic, some of the port plugs for which F4E is responsible contain up to nine separate diagnostic systems. Nevertheless, feasible conceptual designs now exist for all the port based diagnostics.

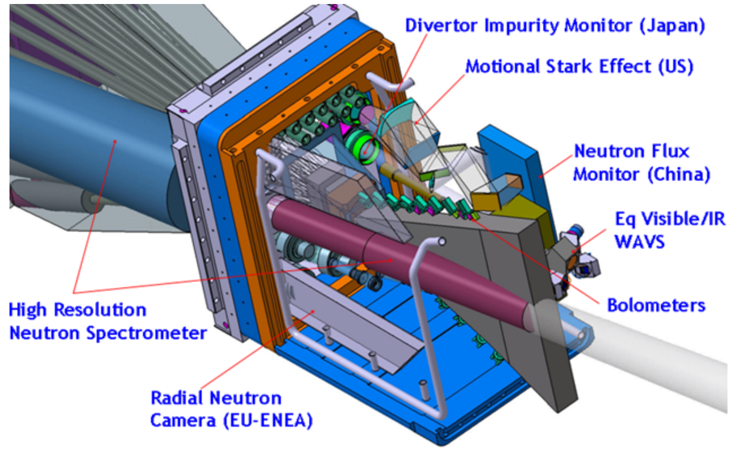


Figure 45 - Cut-away of equatorial port-plug with radiation shielding removed, showing conceptual integration of many complex diagnostics systems in a single port. The result resembles, in size and complexity, a space satellite. There are 18 diagnostic port plugs on ITER, 5 of which will be supplied by F4E.

Many other technical challenges, of course, remain. Most notably, perhaps, ensuring sufficient lifetime for optical components facing the plasma, which may be eroded by energetic particles or deposited with material eroded from nearby surfaces, and ensuring the compatibility of components with the ITER radiation environment. In both cases, a programme of R&D is on-going. Future F4E Grants will continue to address open R&D issues, such as these, whilst progressing further the detailed design of diagnostic systems, in preparation for the procurement phase.

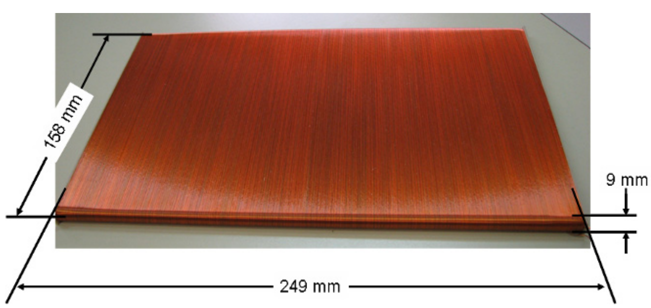


Figure 46 - Prototype ex-vessel magnetics sensor. These large coils are wound using conventional technologies but with great precision to achieve the necessary accuracy in the elongated geometry.

Towards the end of 2008, the first F4E 'Call for Proposals' related to diagnostic systems was launched, aimed at qualifying prototypes for the ex-vessel magnetics sensor coils (Figure 46). These include large magnetic field sensor coils, which mount on the exterior of the ITER vacuum vessel and act as a backup for coils inside the vessel, and so-called 'Rogowski' coils, which provide one of the most important measurements on ITER: the plasma current. The Grant has now been awarded, to a consortium of the CRPP, CEA and ENEA-RFX Fusion Associates.

### Materials Engineering relevant to Test Blanket Modules

Structural materials development and qualification for breeding blankets has two main objectives (i) to validate materials and joining processes for ITER TBM and (ii) to further improve properties under and after neutron irradiation and key fabrication technologies for DEMO breeding blankets. The materials studied are 9% Cr steels EUROFER and ODS-EUROFER.

Progress has been made in the qualification of EUROFER. The existing data base of physical and mechanical properties and different joining processes of un-irradiated and irradiated material has been broadened.

After completion of most irradiation campaigns launched under EFDA up to DEMO relevant doses and post irradiation examinations some conclusions on critical issues can be drawn, in particular on irradiation embrittlement. Under neutron irradiation the transition temperature between brittle and ductile material behavior (DBTT) is shifted towards higher temperature. Comparing irradiation campaigns on EUROFER steels at different irradiations temperatures and differently heat treated, there is enough evidence for a saturation of the DBTT at DEMO relevant doses of about 70 dpa (Figure 47)

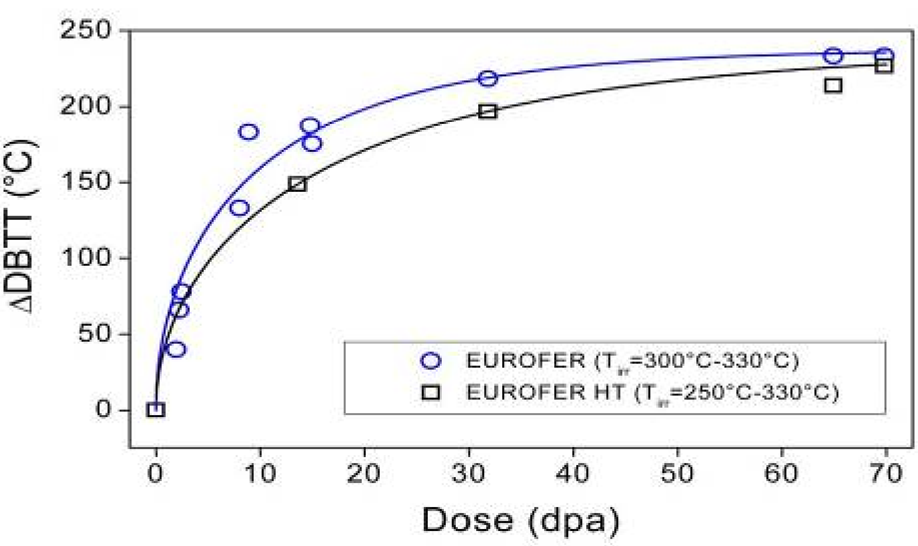


Figure 47 - Dependence of irradiation induced embrittlement on neutron dose: Change in ductile to brittle transition temperature (DBTT) with increasing irradiation dose for EUROFER steel at low irradiation temperatures and for two different heat treatments.

For high temperature DCLL TBM specific SiC/SiC ceramic composites materials are developed for, both, good thermal and very high electrical isolation as flow-channel inserts. As the first of its kind worldwide, 11 plates of 100 square cm² have been successful at industrial scale fabricated by MT Aerospace.

### Test Blanket Modules

Future fusion reactors will need to re-generate the tritium (T) consumed in the D-T reactions and to extract the thermal power generated by the plasma under economically sound conditions for electricity production. These functions shall be ensured by a so-called tritium breeder blanket covering the inner side of the vacuum vessel and directly facing the plasma. For several years, Europe has been developing two tritium breeder blankets concepts that will be tested in ITER under the form of Test Blanket Modules (TBMs) located in an equatorial port of ITER (Figure 48):

i. the Helium-Cooled Lithium-Lead (HCLL) concept which uses the eutectic Pb-15.7Li (enriched in $^{6}Li$) as both tritium breeder and neutron multiplier,

ii. the Helium-Cooled Pebble-Bed (HCPB) concept with lithiated ceramic (enriched in $^{6}Li$) pebbles as tritium breeder and beryllium pebbles as neutron multiplier.

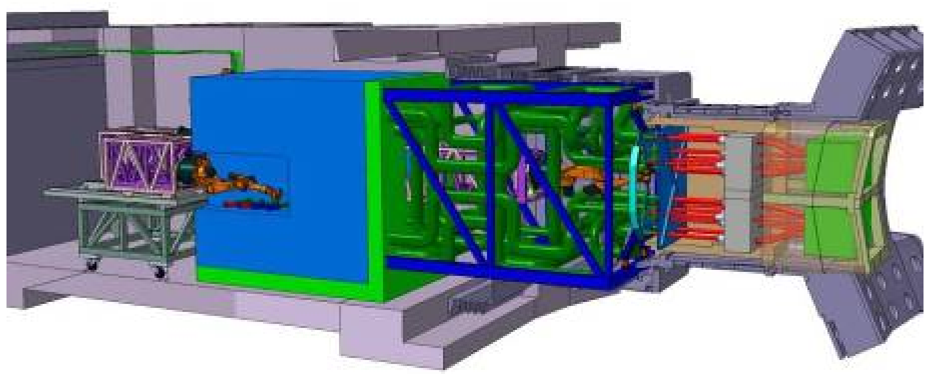

Figure 48 – HCLL and HCPB Test Blanket Modules (TBM) to be tested in an ITER equatorial port

Both concepts are using a 9Cr-WVTa Reduced Activation Ferritic Martensitic steel as structural material, the EUROFER, and pressurized Helium technology for efficient heat extraction (300-500°C, 8 MPa). Each TBM consists mainly of a EUROFER steel box containing the tritium breeder and neutron multiplier materials as well as a series of heat extraction plates that are directly in contact with these materials. An internal stiffening grid provides mechanical resistance and segregates the volume into cuboids containing breeder/multiplier materials and cooling plates. All the EUROFER steel structures are actively cooled by circulation of pressurized Helium in internal channels. A manifold system located at the back of the TBM ensures the distribution/collecting of Helium to/from the various parts of the TBM structures, in a way that optimizes the temperature of TBM materials according to their function. The tritium released by the breeder material is transported via a slowly circulating Helium purge stream or Pb-15.7Li flow through the external detritiation units where the tritium is recovered.

The fabrication of the EUROFER box of TBMs requires the development and qualification under nuclear standards of advanced fabrication technologies. The fabrication of heat extraction panels (first wall, cooling plates) is mainly obtained by diffusion bonding of plates with specifically grooved channels (Figure 49 and Figure 50). Recent developments demonstrated the feasibility of fabrication by implementation of Hot Isostatic Pressing (HIP) or combined YAG laser welding / HIP (CEA patent) on reduced-size mock-ups. Thermo-mechanical cycling tests have demonstrated the resistance of a cooling plate under PbLi environmental conditions and thermal loadings (3,000 cycles) as expected in ITER. The assembly of the heat extraction panels and structural plates to form the TBM box is performed by welding. Here again, developments have been carried-out to optimize and qualify welding processes for their use with EUROFER steel and to minimize the overall deformation of the box. Processes studied go from classical multi-pass TIG to more advanced ones like high energy single-pass YAG laser or hybrid laser/MIG (CEA patent). Figure 51 shows a mock-up demonstrating the assembly of 11 mm thick EUROFER plates using a single-pass 8 kW YAG laser process. The thick plate assembly inside is the clamping tool which will be removed after welding. The process has been optimized to minimize deformation and to avoid damaging the internal plate cooling channels located close to the weld. Future development contracts will now address the fabrication of larger size mock-ups and their qualification according to (nuclear) Codes & Standards.

Many experiments have been performed to study, optimize and model the performance of the TBMs in fields like thermo-mechanics of structures or pebble beds (Figure 52), magneto-hydrodynamics of flowing liquid metal under magnetic field (Figure 53), ceramic breeder and Beryllium pebbles irradiation performances in fission reactor, tritium permeation barrier efficiency (in-situ developed oxide or Al-based coatings) to limit tritium diffusion toward the TBM coolant, filling of TBM mock-up with pebbles to optimize the packing factor, etc. In addition, test facilities allowing future testing of TBM prototypes up to full-scale parameters are under construction.

Figure 49

Figure 50

TBM first wall (Figure 49) and cooling plate (Figure 50) EUROFER mock-ups fabricated by diffusion bonding. These components feature internal cooling channels for heat extraction by pressurized Helium, 300-500°C, 8MPa.

Figure 51 - Assembly mock-up of the TBM internal stiffening grid obtained by single-pass YAG laser welding of 11mm thick plates

On-going and future development contracts will allow completing performance simulation by experiments and modelling, detailed design of ancillary systems (helium cooling system, tritium extraction system and coolant purification system) and TBMs support equipment, their integration in the ITER plant and the definition of a TBM testing plan compliant with the ITER research and operational plan.

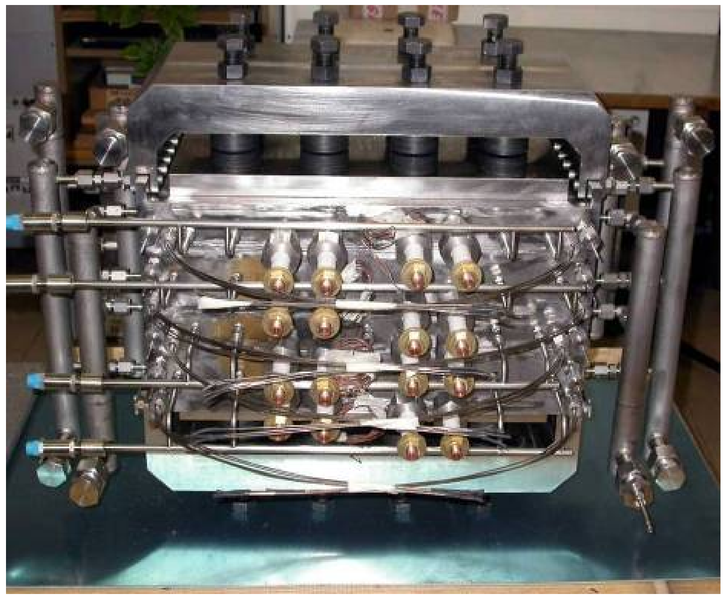

Figure 52 – External view of the HEXCALIBER mock-up containing piled-up layers (not visible) of lithium ceramic / beryllium pebbles and cooling plates for investigation of thermo-mechanical behavior of the HCPB TBM.

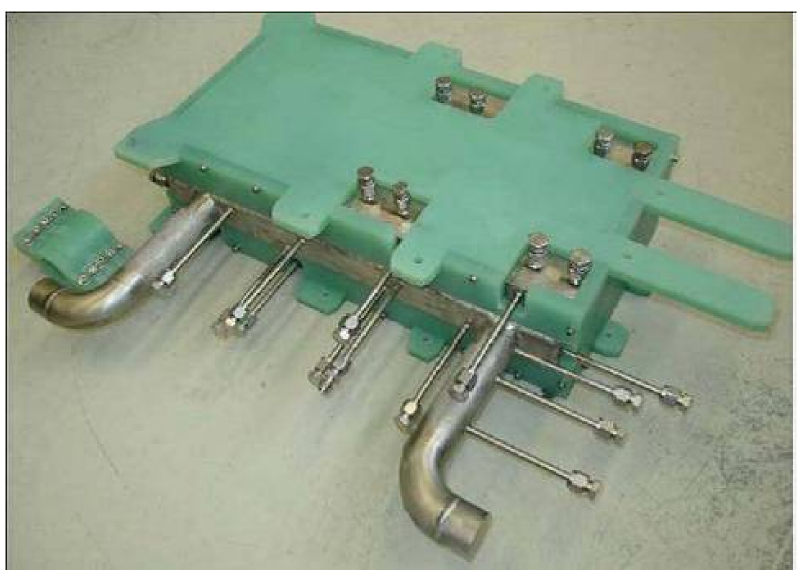

Figure 53 – Half-scale mock-up of a section of the HCLL TBM for measurement of magneto-hydrodynamic performances (liquid metal velocity profiles, pressure drop) in the MEKKA test facility.

F4E's Plasma Engineering activities in 2007-08 include participation in the ITER technical activities, the coordination of the design of the Electron Cyclotron Upper launchers and Ion Cyclotron Antenna, as well as analysis for the evaluation and optimization of critical design aspects of the ITER device.

In the area of ITER design analysis and optimization, a detailed assessment of the ITER operational space was performed, taking into account the recent modifications to the machine design regarding the poloidal field coils and the divertor geometry. The analysis considers also the headroom needed by the control system in terms of coil current. . The overall results indicate that the control of 15MA plasmas in ITER is likely to be adequate in the range of *li* 0.7-1.0. The resulting operational space is shown in Figure 54.

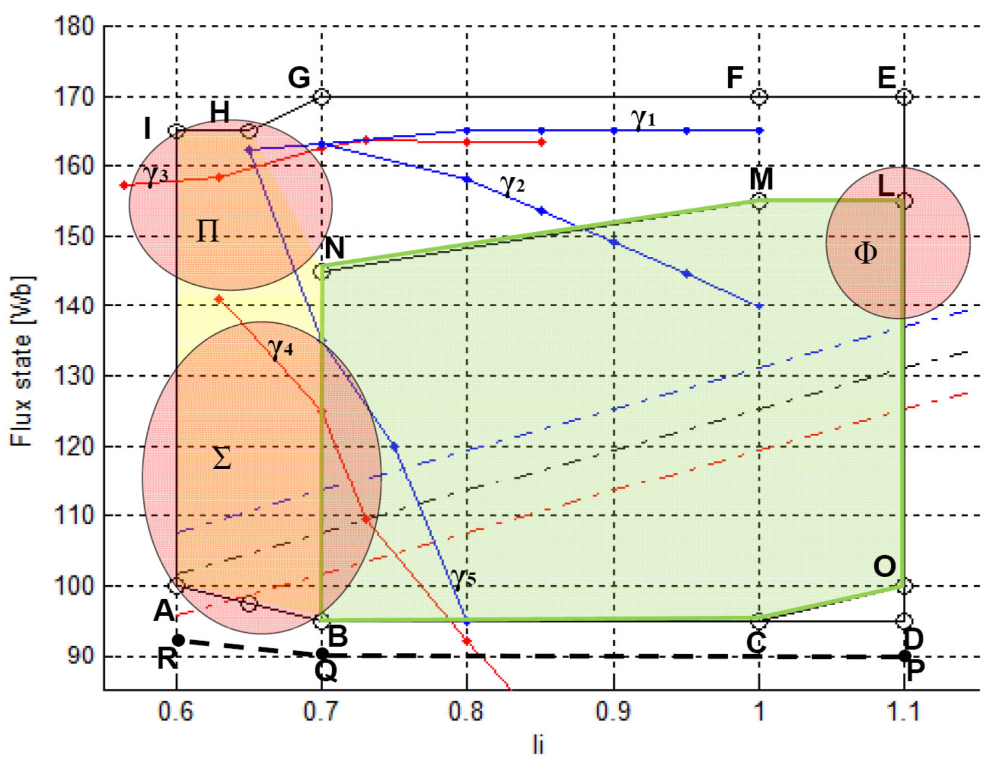


Figure 54 – ITER 15 MA operating envelope in the *li*-flux state domain

In support of the specification and design of the in-vessel coils system, a study of the Vertical Stability of the ITER plasmas was performed, including the implications of the EU proposal of modifications to the Vessel internal shell. The results of the study confirmed that in-vessel coils are the only option for ensuring a reliable VS stabilization system and, for the case of the EU modified Vessel, the requirements in terms of current and voltage rise by 20%.

The issue of ripple-induced fast ion losses was also investigated in detail. 3D magnetic field and ITER wall geometry was implemented in the ASCOT code. Fast ion losses due to the Toroidal Field ripple are found to be acceptable for the reference plasma scenarios. On the other hand, island formation may occur due to the local ripple created by the massive ferromagnetic Test Blanket Modules. These islands could induce anomalous fast ion losses not presently accounted for. This area is the subject of investigation for 2009-2010.

### Safety

Two types of activities have been performed in support of ITER licensing: R&D and safety analyses.

**R&D in support of ITER licensing**

Activities have been undertaken that are aimed primarily at completion of that part of the R&D programme needed to provide the elements to support and complete the ITER Preliminary Safety Report (RPrS). In addition the necessary long term R&D activities have been defined in more detail. Typical examples are:

- Dust management inside the vacuum vessel (production, mobilization, diagnostic and removal) for which F4E has participated in working groups to define a dust inventory control strategy and to outline a R&D programme for dust diagnostics and validation of removal techniques;

- Combined hydrogen/dust explosion models development and validation. Significant new results have been obtained (Figure 55) to improve the database for explosion of Tungsten/air and Hydrogen/Tungsten/air mixtures, and the case of Beryllium dust explosion has been addressed by defining a new activity inside the F4E Work Programme;

- Demonstration of the feasibility of prevention and/or mitigation of in-vessel hydrogen/dust explosions has taken benefit of Limiting Oxygen Concentration experiments (Figure 56) and of the validation of computer code models versus experiments.

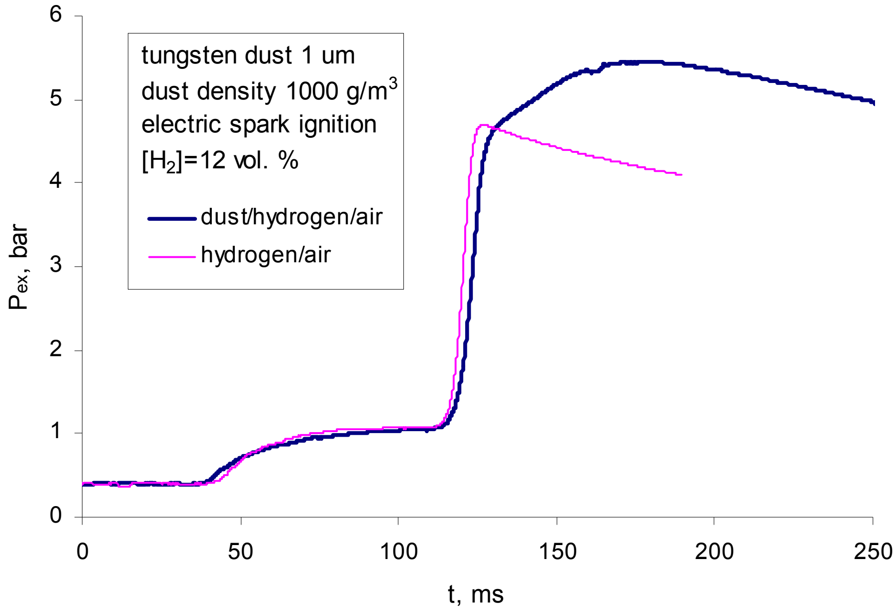


Figure 55 - Dustex experiments - combined hydrogen/tungsten dust/air explosion

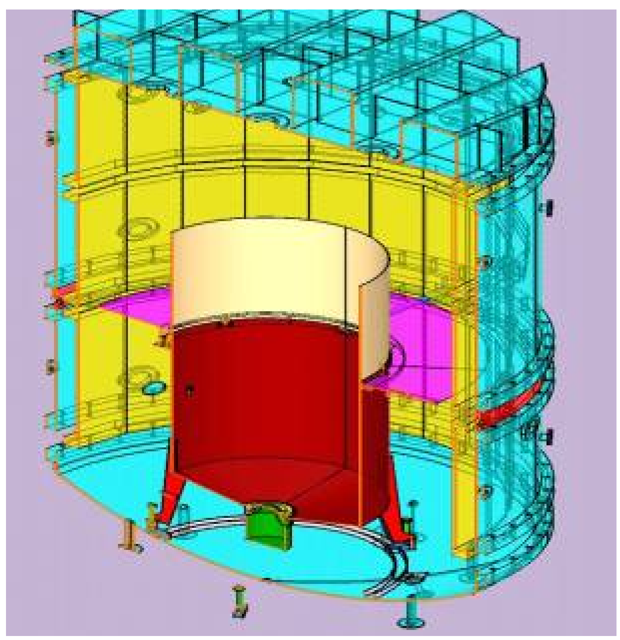

Figure 56 - $H_2$/Dust explosion mitigation, MISTRA Facility

**Analyses in support of ITER licensing**

A number of activities have been performed covering urgent technical support for ITER-IO for the rapid definition of some critical aspects of the design, the support to ITER in the form of 'second review' of the text of the Preliminary Safety Review and preparation and participation to the meetings with the French Safety authorities. Typical examples are:

- Analyses in support of the RPrS have continued by taking into account the design evolution and the feedback from the licensing authority. Emphasis has been put on occupational safety (e.g. ALARA application based on design evolution) and on more detailed calculation of some accident sequences;

- First safety analyses for the EU Test Blanket Modules have been performed to approach the level requested by the licensing authority;

- Demonstration of operating experience from existing fusion machines (Tore Supra and ASDEX-U;) in support of ITER design solutions and projected performance has continued;

- ITER upgrade (neutron fluence 1.0 MWa/m$^2$ instead of 0.3 MWa/m$^2$, full tungsten first wall instead of Beryllium) has been assessed from the safety point of view.

### Analysis and Codes

**EDIPO Project**

EDIPO is a high-field superconducting magnet under construction at Babcock-Noell in Würzburg (Germany). Once completed, EDIPO will be among the world largest high field dipoles and it will be used, primarily, to test the ITER superconducting magnet production. During 2008, the dummy DC coil winding was completed and heat treated. The AC coils were also manufactured as well as the test wall and outer cylinder (Figure 57).

Figure 57 - EDIPO dummy coil after reaction heat treatment (left) and its containment steel cylinder 35 mm thick (right)

**European ITER Site Studies (EISS5)**

On January 2008, the Demande d'Autorisation de Création (DAC), including the Rapport Preliminaire de Sûreté (RPrS), was delivered by the ITER Organization (IO) to the French Authorities. It was the culmination of a joint effort between EFDA and ITER IO where Europe has developed the ITER site studies in a wide range of technical aspects, many of them never considered before for a fusion facility.

### Materials and Fabrication relevant to ITER

The work in the Materials & Fabrication field in 2008 aimed at qualification of material manufacturing, material properties and the joining of different materials. Extensive tasks are carried out in order to ensure that the materials and joints, foreseen in the EU contribution to ITER, are qualified. Significant work was also carried out to develop and qualify alternative materials and manufacturing routes to have more options and candidates to choose from for the procurement of ITER components.

The work includes irradiation campaigns, thermal fatigue testing and assessment of material data (Figure 58). The EU mock-up passed the qualification programme of First Wall panels that is aimed at evaluating Cu-alloys (CuCrZr), stainless steel 316L(N)-IG, beryllium and the joints between the materials. Work was carried out on the Shield Blanket Modules that was aimed at optimizing the powder-HIP manufacturing route in order to have a cost effective solution.

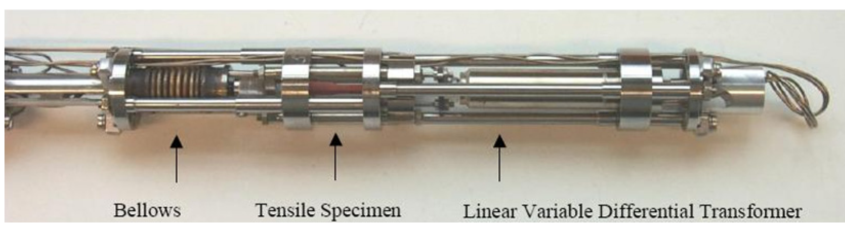


Figure 58 - The final assembly of a complete test module before installation in the irradiation rig in TW5-TVM-SITU2.

The work on Divertor materials continued with qualification of carbon fiber composites (Figure 59) and tungsten. A corrosion study was carried out that elucidated that the effect of copper impurities on 316L(N) during ITER operational conditions is limited and the choice of material was good. The Divertor material specifications were amended in collaboration with ITER.

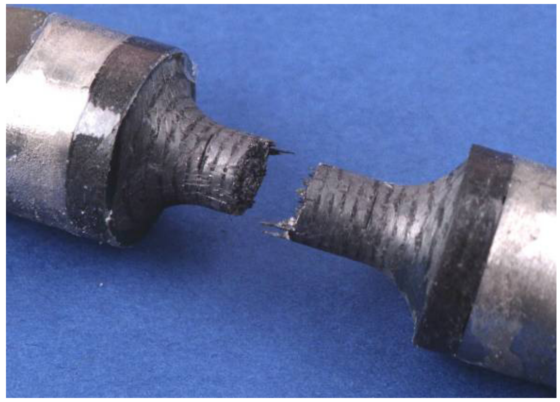

Figure 59 - Carbon fiber composite specimen from the task TW5-TVM-CFCQ2 showing a fracture surface in the low conductivity, needled, direction

An assessment of leak detection techniques for the Vacuum Vessel (VV) was completed, which is a concern during the assembly of the VV sectors. Laser based spectroscopy techniques appear most promising to achieve high precision.

## Nuclear Data

Two tasks associated with Nuclear Data have been completed:

- The European Activation Fusion library has been validated and improved based on feedback from statistical analysis and integral validation analysis covering energies up to 60 MeV. In particular Cromium-52 cross sections for the interesting range of energy have been evaluated and Manganese and Tantalum cross sections have been revised using Monte Carlo calculations with MCNP. Methods for the calculation of covariance have been extended to energy beyond 20 MeV for tungsten and oxygen.

- The construction of the TBM-HCLL mock-up has been completed and the irradiation and performance of the neutronics experiments and necessary measurements like Tritium Production Rate (TPR), neutron and gamma ray flux and spectra have been performed (Figure 60). The evaluation of the results is in progress.

Figure 60 - TBM-HCLL mock-up experiments

# Broader Approach

## Satellite Tokamak (JT60-SA)

The period from signature of the BA agreement (5th May 2007) to the end of 2008 has been primarily utilised to:

- build-up the technical teams in EU and JA;
- revise the overall machine design for cost reduction;
- set-up a common quality management system and a number of management tools.

In the EU, the preparation of technical specifications, for the components included in the EU in-kind contribution, has progressed and a detailed schedule for procurement implementation has been developed.

**Project Integration Activities**

Many project integration activities have been realised during the reporting period including:

1. Integrated Project Team (IPT) and staffing: the structure of the IPT, consisting of the Project Team, the JA Home Team and the EU Home Team has evolved, with an increasing number of staff members in the three Teams. The organisational structure of the Project Team (PT) was better defined and its staffing reached 7 members (including part-time members). The JA Home Team (JA HT) increased to about 130 persons, made available, on a full-time basis, by JAEA. The EU Home Team (EU HT) increased to about 50 persons including 11 full-time members at the F4E Antenna in Garching, and full-time/part-time members among the EU Voluntary Contributors.

2. Infrastructure and management tools: Fusion for Energy (F4E) implemented and managed the JT-60SA Document Management System (DMS). The “Common Management and Quality Programme” (CMQP) for the JT-60SA Project was submitted for information to the 3rd Steering Committee (SC-3) meeting.

3. Integrated Design Report (IDR): The IPT devoted itself to the development of the Integrated Design Report (IDR), along the lines indicated by the Steering Committee (SC). The IDR, consisting of the Re-baselining Report, the Plant Integration Document (PID), the Schedule, the Common Management and Quality Programme (CMQP) and the update of the in-kind contribution sharing and

formal valorisation (the so-called "value allocation"), was approved by the SC at its 4th Meeting at the end of 2008 (SC-4).

**Detailed Design and Procurement Implementation**

A substantial redesign, with the primary aim to reduce costs, was successfully completed and reviewed during 2008. In order to achieve the required savings, while maintaining the machine scientific mission, it has been necessary to introduce several design changes, both at system and at component level. The machine was first optimised, as a whole, defining and analysing the system-level requirements. This resulted in a lowering the overall aspect ratio, while decreasing the toroidal field and retaining the plasma current, as well as in the re-design of the PF system. The system-level optimisation was, then, followed by a number of detailed component-level design optimisations and component-to-component trade-off analyses. Table 2 outlines the main changes implemented, and the advantages, in terms of reduction of quantities or simplification of manufacturing.

Table 2 – Main changes implemented in Satellite Tokamak Project Re-baselining

| Component | Items | Changes |
|---|---|---|
| TF magnet | Amount of strand | Reduction by more than a factor of 2. |
| | Length of the conductor | Reduction from 30 km to 24.4 km. |
| | Number of turns | Reduction from 90 turns to 72 turns. |
| | Mass of the structure | Reduction from about 800 tonnes to about 300 tonnes. |
| | Amount of cooling channels in the casing | Drastically reduced. |
| | Nuclear heating as well as AC losses | Reduction by a factor of ~3. |
| PF magnet | Number of EF coils | Reduction from 7 to 6 coils. |
| | Total conductor length of EF system | Reduction by more than 10 km (from about 38 km to about 27 km). |
| Vacuum Vessel | Weight of vessel body including ports | Reduction from about 230 tonnes to about 150 tonnes. |
| | Thickness of inner/outer shell | Reduction from 24 mm to 18 mm. |
| | Vessel body shape | Simplification to a multi-arched shape in poloidal direction and a polygonal shape in toroidal direction (10 degree segments). |
| Cryostat | Shape of vessel body | Simplification from spherical shape to a cylindrical shape with single curvature, faceted shape. |
| | Double wall structure filled with boron doped concrete | Simplification to single wall structure and elimination of the boron doped concrete and the outer shell. |
| Cryogenic | Refrigeration load | Reduction of total refrigeration load (4.5 K equivalent load) from 15 kW to about 10 kW, as well as simplification of the layout. |
| ECRF | Number of units and pulse duration in the Initial Research Phase | Deferral: the nine 100s-units foreseen in the CDR[5] shall be actually installed in the Integrated Research Phase. In the Initial Research Phase, the ECRF system will start with two 5 seconds-units and two 100 second-units. |
| | Frequency of additional 5 units in the Integrated Research Phase | Frequency is presently considered as 110GHz. |

[5] CDR: Conceptual Design Report , basic technical document and original reference for the Broader Approach Satellite Tokamak Programme

**TF magnet (EU)**

**Design activities:**

1. The design of the TF magnet was developed, during 2008, achieving significant cost reductions. The design evolution of TF magnet was summarized in the Re-baselining Report and the main parameters of the technical specifications were included in the Plant Integration Document.

2. Subsequent further detailed design developments included:

   a. Improved design of the TF coils mechanical structure based on the mechanical connection of the TF casings to the outer inter-coil structures. Such design allows:

      - reducing the coils size and the "cold mass" in the test facility,

      - reducing the size of largest components for transportation,

      - enhancing flexibility during assembly, and

      - improving the sharing of the work packages among EU Voluntary Contributors.

   b. New design of TF magnet supports, with hinged support bars replacing flexible supports, with reductions of stresses induced during to machine cool-down and thermal losses through the supports.

   c. The design of the winding pack also progressed and was validated by detailed stress and thermal-hydraulics analysis. The new solution allows maintaining the required temperature in the TF conductors, after a plasma disruption (and consequent heat generation in the coil casings), by means of glass epoxy insulation, acting as a thermal barrier; this solution avoids the expensive TF casings cooling channels.

3. The qualification process for the TF conductor has continued and samples reproducing the baseline conductor have been successfully tested in the first part of 2009.

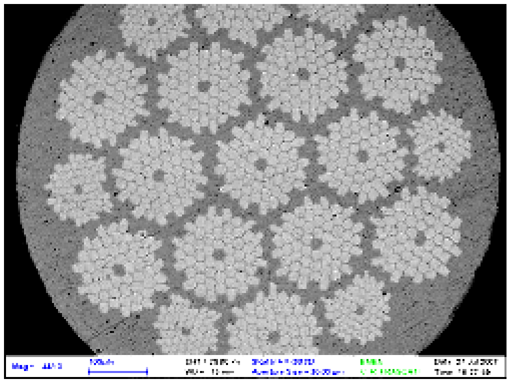

Figure 61 - Section of the JT-60SA TF conductor NbTi strand diameter 0.81mm. Effective Filament diameter 15 microns

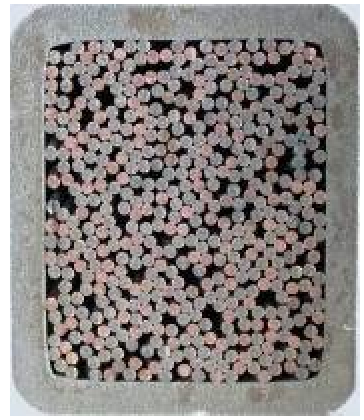

Figure 62 - Section of the JT-60SA super-conducting cable. 324 conductors carrying 25.7 kA at a peak magnetic field of 5.65 T.

4. The design of the facility, to be built and operated in the EU, for the test of the toroidal field coils, prior to shipment to Japan, has progressed with the definition of the test concept, preliminary sizing of the test equipment, preparation of the schedule for setting up the facility and performing the tests, draft technical specifications/test programme and agreement on the sharing among the EU Voluntary Contributors. Final site selection, in Europe, is underway, with two candidate sites (CEA Cadarache or CEA Saclay[6]).

5. The preparatory procurement activities for the High Temperature Superconductor Current Leads are underway.

[6] The site has been now chosen and the JT-60SA Coils Test Facility shall be located in CEA Cadarache

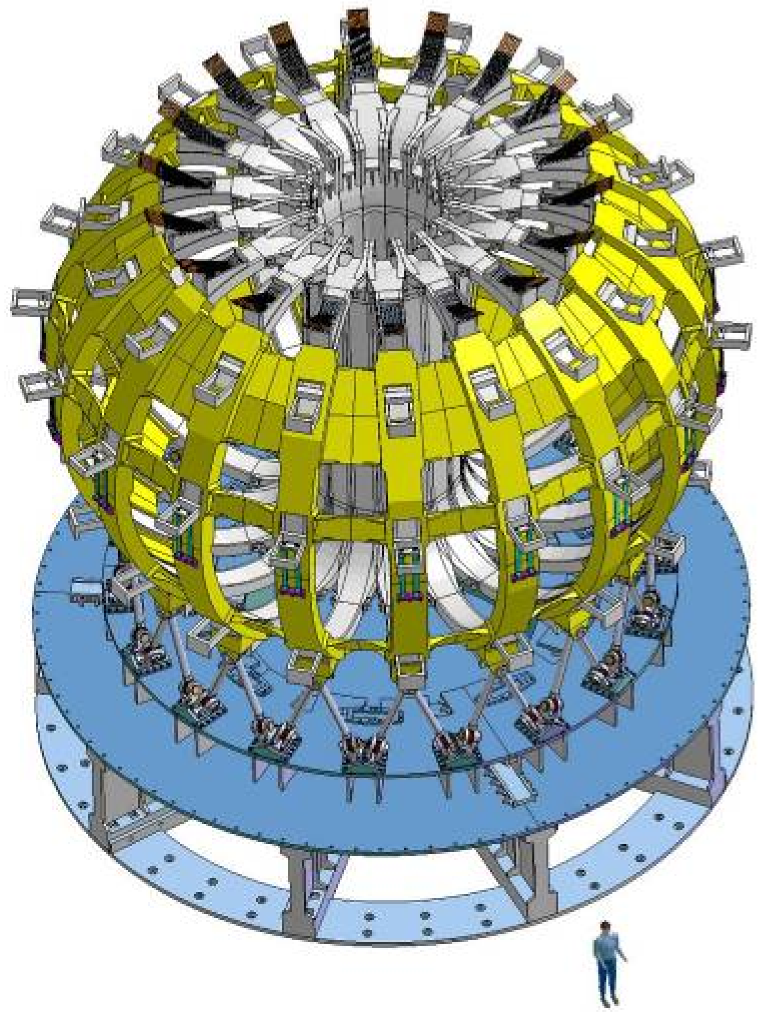

Figure 63 - JT-60SA - Toroidal Field Coils -18 coils using 24 km of helium-cooled Niobium-Titanium superconducting cables, carrying 25.7 kA, supported by a 190-tons stainless steel mechanical structure.

**PF magnet (JA)**

**Design activities:**

1. The design evolution of PF magnet was summarised in the Re-baselining Report and the technical specifications were included in the Plant Integration Document.

2. The detailed design of the PF magnets, consisting of the Central Solenoid (CS) and the Equilibrium Field (EF) coils, was completed, while the further development of some interfaces between the CS and TF coils is underway.

**Procurement activities:**

1. PF Conductor PA [Period: 4 October 2007 - 28 December 2012]

- Seven industrial contracts were awarded in March 2008.

- In these contracts, strand pre-production and procurement of strand manufacturing equipment is progressing. The hot extrusion and cold working for conductor’s jackets were also started.

- The experimental proof of the PF conductor performance, using the National Institute for Fusion Science (NIFS) facility, was satisfactorily achieved.

- The fabrication for Nb3Sn strand for the CS conductor, and NbTi strand for the EF conductor, were started with successful tests (in November 2008 and December in 2008, respectively), which allowed launching the mass production.

- The production of 200 m long dummy copper cables, reproducing the CS and EF conductors, was completed.

2. PF Coil Manufacturing Building PA [Period: 29 June 2008 - 30 October 2009]

- The PA for the supply of the PF Coil Manufacturing Building was concluded in 29 June 2008.

- Two contacts for the Winding Building and the Jacketing Building were awarded in June and August 2008.

3. PF Coil Manufacturing PA

- The PA for the PF Coil Manufacturing was submitted by JAEA to F4E for signature on 26 December 2008.

**Vacuum vessel (JA)**

**Design activities:**

1. The design evolution of Vacuum Vessel was summarised in the Re-baselining Report and the main technical specifications were included in the Plant Integration Document.

2. Design of the Vacuum Vessel Welding Building was developed by the JA HT and the launch of the PA is planned for March 2009.

**Procurement activities:**

1. Vacuum Vessel PA [Period: 28 January 2008 - 27 December 2013]

   - The PA for the supply of the Vacuum Vessel was concluded on 28 January 2008.

   - Two contracts were awarded in March 2008.

   - The detailed design of the Vacuum Vessel was performed indicating some minor interface issues with the TF coil remaining to be solved.

   - The manufacturing design and the welding R&D for fabrication of the Vacuum Vessel were conducted at the contractor's factories, focusing on the fabrication sequence, automatic welding equipment, welding jigs, qualification of weld processes (TIG and MAG, filler materials, tensile, bending fatigue tests and Charpy impact test), weld non-destructive examination, dimensional controls and tolerances.

   - The plate materials for Vacuum Vessel (SUS316L) were shipped to Naka site in October 2008.

**In-vessel components (JA)**

**Design activities:**

1. The new lower divertor design was developed, considering recommendations from the several detailed design reviews and plasma physics assessments.

2. The review of the divertor design, from the point of view of remote handling maintenance, was initiated in November 2008, taking advantage of a newly established JET-JAEA collaboration on the divertor design.

3. The detailed design report of the divertor components was prepared and the technical specifications for the PA for the divertor cassette were drafted.

4. R&D of the mono-block divertor plate samples was carried out in preparation of the procurement of the divertor targets. Twelve full-size divertor elements mock-ups were manufactured and were used for high-heat flux tests (achieving a heat removal performance higher than 15 $MW/m^2$) and for non-destructive test (based on thermographic imaging).

5. The R&D of the bolted armor tiles (Carbon Fibre Composites and graphite tiles) was also carried out to confirm the design and materials choice, with particular regard to heat transmission.

**Procurement activities:**

1. Raw materials for in-vessel components [Period: 28 January 2008 - 31 August 2012]

   - The PA for the supply of a first part of the raw materials for in-vessel components was concluded on 28 January 2008.

   - Three contracts for the supply of raw materials for divertor targets, domes and baffles were awarded in 2008, and industrial fabrication of Carbon Fibre Composites materials has started.

**Power supply and control (EU and JA)**

**Design activities:**

1. JT-60SA Power Supply (PS) system consists of existing units, presently operating on JT60, and new components. Among these ones, EU contribution includes: the Power Supplies (PSs) and Quench Protection Circuits (QPCs) for both the Poloidal Field Coils (PFCs) and the Toroidal Field Coils (TFCs), the Switching Network (SNU) for Central Solenoid (CS) coils 1-4 (which provides the voltage for plasma breakdown), the PS for plasma control coils and the PS to control Resistive Wall Modes (RWM).

2. The design evolution of the PS and control was summarised in the Re-baselining Report and the technical specifications were included in the Plant Integration Document.

3. A complete model of the JT-60SA poloidal circuit, including all the passive structures, was developed, and it was utilized to analyze the effects of the plasma disruption and of QPC intervention.

4. Two different complete models of the JT-60SA PS system, including the Motor Flywheel Generators (MFGs), were developed in order to verify MFGs compatibility with the JT-60SA electrical loads and thyristor converters performances.

5. Various technical solutions for the overall PS general scheme have been studied and the main characteristics of JT-60SA PFC and TFC PSs were defined, including basic assumptions for the operational procedure in case of fault. It was also confirmed that the harmonic currents stay within power grid specified limits.

6. A preliminary proposal for the layout of PS System was made by the JA HT, also on the basis of EUHT proposals, taking into account layout and cost constraints.

7. The studies on the conceptual design of the QPC were completed in the first months of 2008. A new design solution was devised and analyzed; it is based on a Hybrid Circuit Breaker composed of a mechanical switch, for conducting the continuous current, and, in parallel, a static circuit breaker for current interruption.

8. The preliminary design of the SNU was made by EU, including the Hybrid Circuit Breakers.

9. A critical revision of the first conceptual design of the PS for Resistive Wall Modes control was begun in the second semester of 2008.

**Cryostat (EU)**

**Design activities:**

1. The design evolution of the Cryostat was summarised in the Re-baselining Report and the technical specifications were described in the Plant Integration Document.

2. The details of the cryostat design were developed after the re-baselining of the JT-60SA project including::

   a. a new design of the cryostat base of larger diameter to accommodate the space requirement of the TF coils and VV supports; a solution was devised, subdivided in three 120-degrees sectors, and designed for final bolting and welding on-site and to stay within road transportation limits.

   b. a new design of the cryostat body was prepared, as a single wall structure externally reinforced with ribs, consisting of an assembly of single curvature segments simpler and less expensive to fabricate.

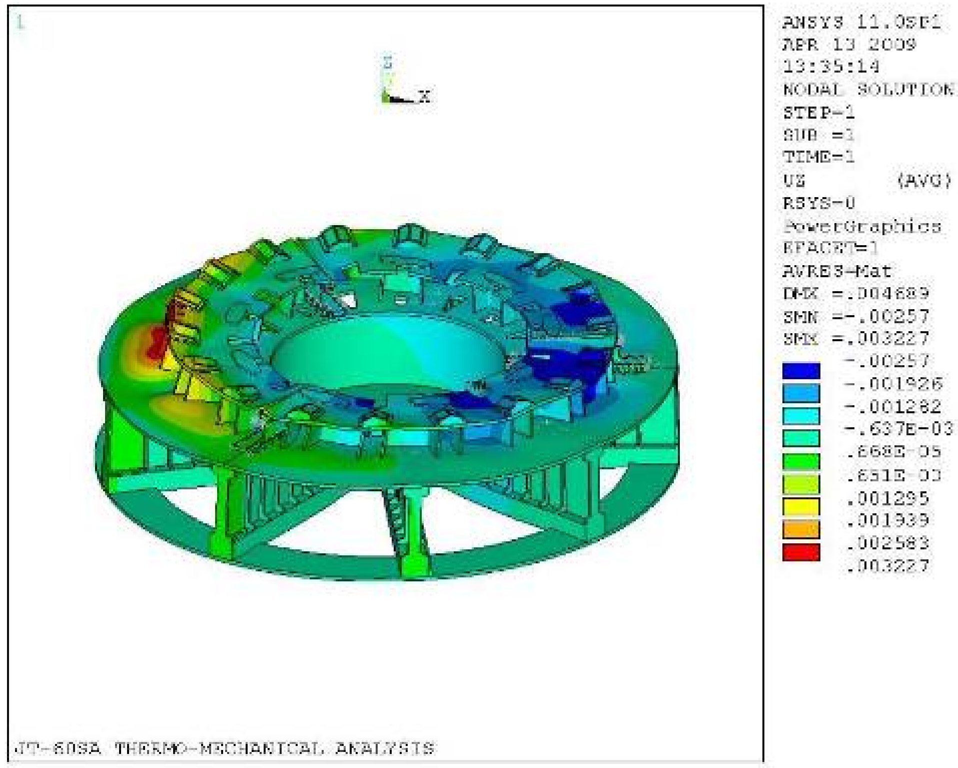


Figure 64 - Structural Verification of Cryostat Base under worst case load case: Temperature gradients, loads of supported structures (vacuum vessel and magnets, external pressure of 1 bar and reference seismic loads

**Cryogenic system (EU)**

**Design activities:**

1. The design evolution of the Cryogenic system was summarised in the Re-baselining Report and the technical specifications were included in the PID.

2. Design work on the cryogenic system progressed to confirm/detail/optimise the loads from all cryogenic users and to estimate the pressure drops of cryogenic circuits.

3. An industrial study was launched focused on:

   a. management of strongly variable loads during plasma operation,

   b. possible savings on capital cost and operation of the plant,

   c. utilities requirement and cryogenic buildings layout.

4. The flow diagram for TF coils, CS and EF coils was detailed. The design of coil terminal boxes and valve boxes allowed defining the

routing of cryolines and interfaces with the passages in the cryostat base. The share of procurement of the cryogenic distribution boxes was adjusted between EU and JA.. The new share allows a better management of the interfaces.

**ECRF (EU and JA)**

**Design activities:**

1. The design evolution of the ECRF was summarised in the Re-baselining Report and the technical specifications were described in the PID.

2. The re-design of the JT-60SA tokamak, and mainly the decrease of the magnetic field by a factor 1.15, has allowed a redesign of the ECRF system. As a consequence the EU procurement for ECRF has been resized to two complete sets of power supplies, powering two 110 GHz gyrotrons (provided by JA).

3. The new design and scope of procurement has been agreed with the EU Voluntary Contributor (Switzerland) and a partnership with the Swiss industry has been initiated in order to study in more detail, the optimal configuration to supply a depressed collector gyrotron of the triode type. By the end of 2008, exchange of information and data, based on the JT-60U gyrotrons operation, was started to define the preliminary power supplies specifications and to finalise the ECRF design details.

4. Design of the "linear motion antenna" also progressed. Calculations confirmed that the heat loads of RF loss on the mirrors are reasonably low, and can be removed by water cooling. Coverage of the beam injection angles and the beam radius were calculated as a function of curvature and the layout of the mirrors.

5. Validation of the improved gyrotron with the new mode convertor for 1 MW /100 s/ 110 GHz started and it is progressing smoothly at JAEA Naka site.

**Assembly**

**Design activities:**

1. A conceptual scheme for assembly was developed and included in the Re- baselining Report

2. An integrated CAD model of all the components placed in the Cryostat was developed, allowing the early identification interferences and assembly sequence problems.

3. A preliminary scheme for on-site metrology has been developed.

**Project Management**

The following key organisational activities were envisaged in the IFMIF/EVEDA Work Programme for 2008:

- Project Team build-up: establishment of a Project Team of 14 people by mid-2008 (10 professionals and 4 support staff).
- Quality Assurance: overall definition of organisation by mid 2008.
- Design Office: Start of implementation at Rokkasho by mid-2008.

At the end of 2008, the Project Team consisted of 8 professionals and 4 support staff.

The IFMIF/EVEDA Project Team features three systems groups, i.e. Accelerator, Lithium Target, and Test Facilities; each group consists of a group leader and a professional (none for Lithium Target so far). They coordinate for their systems, the progress resulting from the European and Japanese contributions.

The “Management and QA” group in the Project Team (a leader, an expert, and a support for IT) was put in place in April 2008, and the guidelines for the IT infrastructure and quality management were established.

The lack of professionals responsible for Design Integration, Design Office, Interface Management, and RAM (reliability, availability, maintainability) analysis, did not allow initiating the full scope of these activities in 2008.

**Test Facilities**

During the reporting period the design was concentrated on the design of the Test Cell architecture. The achievements can be summarised as follows:

- The reference Target and Test Cell (TTC) design has been defined and improved. For the purpose of precise and flexible positioning of “test modules” against the back-plate, a revised reference design of the TTC, together with design variants, was proposed. A merged TTC Concept (MTC) was introduced which combines the following advantages:
  - systematic decoupling of different functions such as handling heavy loads and positioning with stringent precision requirements;
  - facilitated inspection and disassembly by means of a cylindrical steel liner;

- flexibility in the choice of the environment, from high vacuum to atmospheric pressure;

- essentially independent movement and assembly of the "test modules".

- The High Flux Test Module (HFTM) has been modified to include reflector positions and, thus, to gain additional irradiation volume. Despite of severe flux gradients, the additional rigs provide cooling and heating characteristics comparable to those of the rigs in standard locations.

- The study of high flux test modules in experimental helium gas loops were advanced following two complementary routes. On laboratory scale, the instrumentation for deformation testing of the HFTM Single Rig mock-ups was tested in the Helium loop ITHEX-F[7]. For the large scale tests, the helium loop HELOKA-LP, at FZK[8], is going to be used; the loop is in advanced status of preparation, with all major components and piping elements installed. The process control system was fabricated, factory tested, and delivered.

- The nuclear analysis for the Target Backplate and for the High and Medium flux test modules (for which the McDeLicious code is the standard instrument) quantified the contribution to nuclear heating and structural damage of gamma radiation, in comparison to the one of neutrons. With an upgrade of the McDeLicious code, which, now, takes into account the generation photons in deuteron-induced nuclear reactions on lithium, it was found that gamma radiation will increase the nuclear heating up to 10 % in the Lithium Jet, the Backplate and the HFTM, whereas its contribution to the displacement damage and gas production near the d-Li source is negligibly small.

- For the in-pile testing of HFTM mock-ups, foreseen in the reactor of SCK-CEN[9], a first irradiation plan for 4 instrumented capsules (based on the IFMIF reference design) was defined and verified by thermal analyses.

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[7] IFMIF Thermo-Hydraulic Experimental Facility (ITHEX), at the Institute for Reactor Safety, Research Centre of Karlsruhe (Germany).

[8] FZK Forschungszentrum Karlsruhe, Germany.

[9] SCK-CEN Studiecentrum voor Kernenergie – Centre de Etude de l' Energie Nucleaire, Moll,Belgium.

- The CRPP EPFL/PSI[10], in the frame of the tests of a new design for the "test module" concept, developed a novel specimen geometry [11]for the creep fatigue test module.

- CIEMAT[12] performed design studies for the "test modules" for the validation of the liquid breeder components and for the study of tritium release. These studies were based on IFMIF testing specifications. Potential alternative concepts were also validated by thermal and neutronics analyses.

- As first diagnostic tools, providing key information on the neutronic parameters, a fission chamber and an ionization chamber were made available by CIEMAT.

**Lithium Target Facilities**

The IFMIF/EVEDA Work Programme 2008 indicated four fundamental priorities:

- the start of fabrication of the EVEDA Li Test Loop;

- the continuation of the development and qualification of diagnostic tools and the impurity traps and monitors at the Japanese Universities;

- the completion of the preliminary design of the two backplate solutions and their integration into the target assembly;

- the completion of the design and implementation of the purification system for Lifus 3 at ENEA Brasimone Centre and the start of operation of the loop.

The conceptual design phase of the EVEDA Li Test Loop was completed in 2008, allowing the start of detailed engineering and construction.

The collaboration with Japanese laboratories reached the following status in 2008:

- A High Speed Video System for surface imaging was tested at Osaka University.

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[10] CENTRE DE RECHERCHES EN PHYSIQUE DES PLASMAS – Ecole Polytechnique Fédérale de Lausanne – Paul Sherrer Institute

[12] CIEMAT, Centro de Investigationes Energeticas, Medioambientales y Tecnologicas, Spain

- Work on traps and monitors continued at University of Tokyo and Kyushu University and the construction of small circulation loops is ongoing.

Preliminary calculations of the effect of the heat deposition of the deuteron beam in the lithium film were accomplished by JAEA using analogy with models. Calculations showed that backplate curvature can be varied in a large range.

Following the preliminary neutron damage mapping of the backplate, calculated in 2007, the neutron irradiation damage, of the tightening mechanisms region was estimated (in terms of displacement rates) to provide data necessary for the backplate design (performance of the structural material and the anti-seizure coating).

Thermo-mechanical analyses gave the evidence of substantial stress levels and deformations experienced by the whole target due to dissimilar combinations of the structural materials. As a consequence, it was proposed to build the IFMIF target assembly completely in RAF steel.

The erosion/corrosion loop Lifus 3 was commissioned at ENEA Brasimone, leaving aside the purification system. A first erosion/corrosion test of 1,000 hours with AISI 316 and Eurofer 97 specimens was concluded. The start of the loop with the purification systems is shifted to 2009.

**Accelerator Facility**

In 2008, the most significant activities performed for the accelerator facility were devoted to the Accelerator Prototype. Their results can be summarized as follows:

- The change from the conventional to the superconducting solution for the drift tube linac (SC-DTL) was introduced. To this aim, a proposal from the EU Voluntary Contributors for the accelerator facility was submitted for evaluation to the Project Committee in March 2008. The major advantage is that larger beam diameters can be formed. This allows larger accelerating cavities, reduction of wall losses and higher stability of operation at reduced costs of operation. The design change was accepted by the Steering Committee in May 2008.

- The design of most of the sub-systems have experienced significant progress. The most advanced system is the injector, developed by CEA Saclay, for which the Critical Design Review (allowing manufacturing) has been successfully passed. Also, the design of the RF Quadrupole (RFQ) at INFN Legnaro reached a maturity sufficient for “freezing” the reference design in a Preliminary Design

Review. Both reviews were held with the participation of international experts.

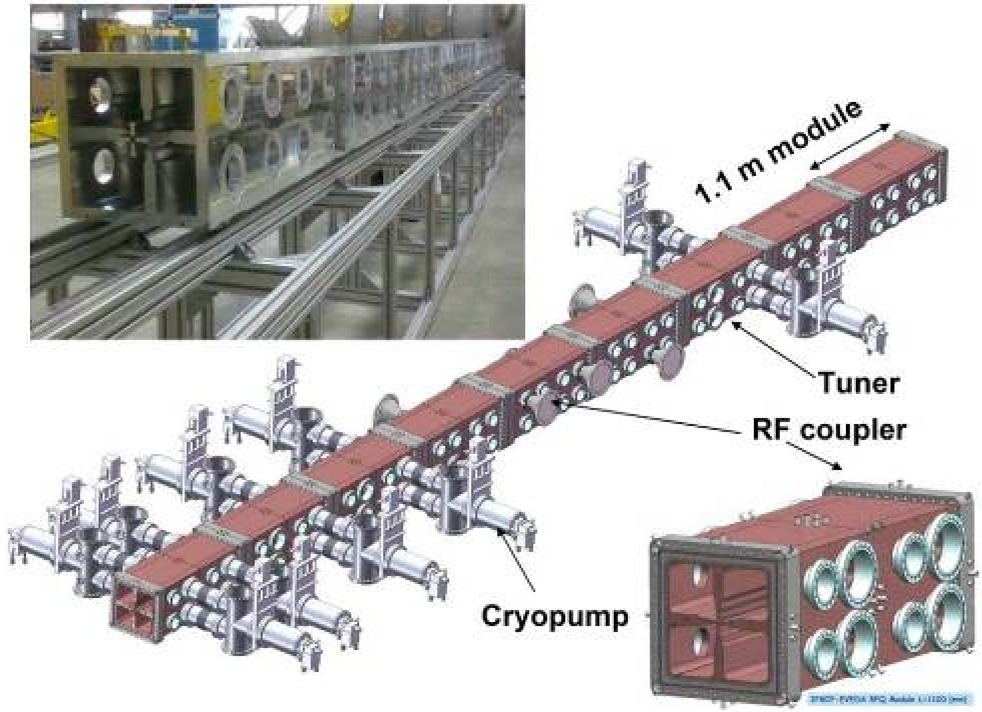



Figure 65 - IFMIF radiofrequency quadrupole (RFQ) unit forming the first accelerating element in the accelerator facility), full-scale aluminum mock-up (in insert) has been set up at INFN Legnaro to develop the tuning algorithms for optimising mode purity.

- Specific difficulties in the accelerator development tasks, due to delays in team and organization build-up, were early identified. A tight project control has been implemented based on a baseline schedule for the Accelerator Facility (AF) in order to identify, as soon as possible, any delays and to introduce corrective actions in the work plan, early in 2009.
- An accelerator specific safety working group was created in order to effectively tackle the safety issues which have been identified as potential project risks.
- The construction of the accelerator building in Rokkasho is progressing in schedule.

**Conventional Facilities**

In the frame of the preparation of the engineering design of IFMIF facility, substantial efforts were devoted to the First identification of Site Requirements and Site Design Assumptions and their categorisation according to the aspects concerning Land, Heat Sink, Energy and Electrical Power, Transport and Shipping, External Hazards and Accident Initiators, Infrastructure, and Regulations and Decommissioning.

### International Fusion Energy Research Centre (IFERC)

The period from signature of the BA agreement (5th May 2007) to end of 2008 has been primarily utilised:

- to specify the design and begin the construction of the buildings that will house the “International Fusion Energy Research Centre” (IFERC) in Rokkasho;
- to start the DEMO R&D activities, as agreed in the IFERC workplan, and specify and design the R&D facilities that will be available in Rokkasho;
- to hold a number of workshops to define the DEMO Design activities that will be jointly conducted, in a second phase, at the “DEMO Design Research and Development Coordination Centre”;
- to define a schedule for the implementation of the “Computer Simulation Centre” (CSC) and to specify the interfaces between the building and the super-computer and refine the procurement sharing;
- to establish a preliminary set of “user needs and requirements” to be used to defined the technical performance requirements and the benchmark test for the CSC super-computer selection.

As for the procurement, the JA Implementing Agency has launched several Procurement Arrangements (PAs) and related industrial contracts.

In the EU, the preparation of technical specifications for the components included in the EU in-kind contribution has progressed and detailed schedule for implementation has been developed. The DEMO R&D technology activities have progressed as planned and the start of formal signature of PAs is planned for 2009.

**Building construction activities**

The construction of the building of the IFERC complex has started and the common technical facilities are planned be installed during 2010.

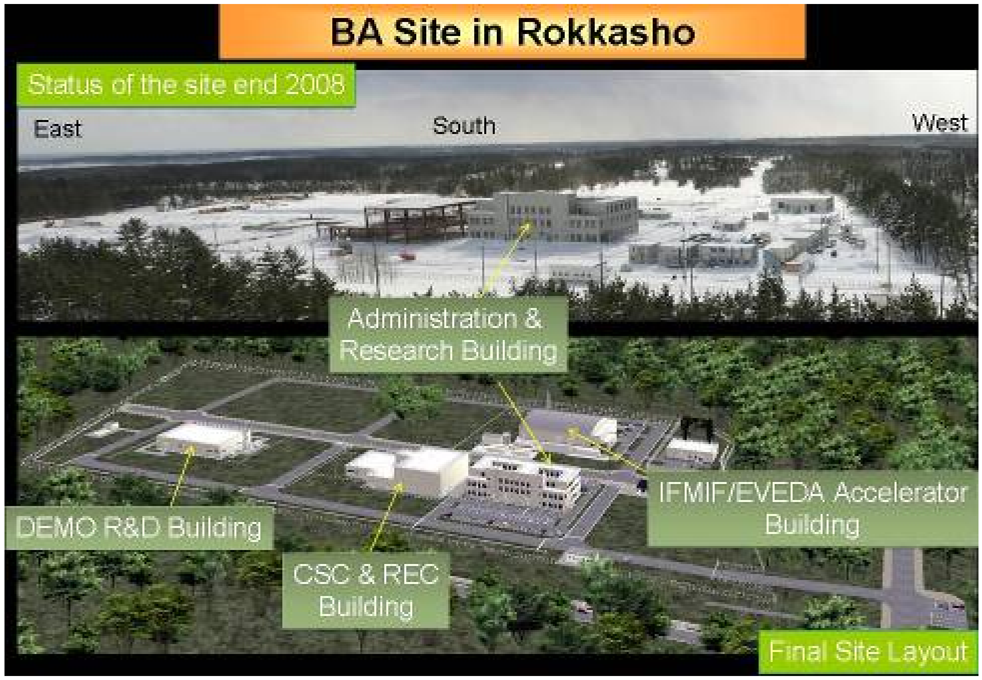



Figure 66 - Status of the BA Rokkasho Site (end 2008) compared to the final layout

**DEMO R&D activities**

The DEMO R&D activities have been conducted, as planned in the 2007 and 2008 IFERC Work Progammes, subdivided in the following main tasks:

- T1 - SiC/SiC Composites;
- T2 - Tritium Technology;
- T3 - Materials Engineering for DEMO Blanket;
- T4 - Advanced Neutron Multiplier for DEMO Blanket;
- T5 - Advanced Tritium Breeders for DEMO Blanket.

In addition, JA has carried out an activity for the preparation of the installation of R&D equipment in the Rokkasho Centre.

In the following paragraphs, the EU and JA main results, obtained in the DEMO R&D activities, are separately reported.

**Results: EU activities:**

**T1 - R&D on SiC/SiC Composites**

- R&D on mechanical properties of SiC/SiC composites: a progressive failure methodology for composite plates was implemented, by developing a "multi-scale" computer code.

- R&D on physical properties of SiC/SiC composites and ceramics: electrical conductivity decrease, associated to amorphization induced by electron irradiation, was observed to occur for hot-pressed (HP) SiC. Desorption and absorption measurements during electron irradiation were carried out, which will be continued in 2009. Surface electrical degradation of HP SiC after He bombardment was observed to occur. Measurement of thermal diffusivity from room temperature up to 500ºC was carried out, together with specific heat and density measurement; during 2009 the measurements will be extended up to 1000ºC. A detailed design of the experimental set up for erosion-corrosion investigation of SiC in Pb-17Li was carried out.

**T2 R&D Materials Engineering for DEMO Blanket**

- Optimization of fabrication technology: as for the manufacturing technology, an optimization of the candidate steels produced in EU, by applying a double final heat-treatment has shown promising results in terms homogeneity through the plates thickness.

- Irradiation effects on mechanical properties and microstructure: the activities were focused on the design of thermo-mechanical treatments (cold rolling and heat-treatment) for Eurofer97 plates which can simulate the irradiation hardening and strain-hardening reduction observed for irradiated tempered martensitic steels. Microstructural characterization of the degraded plates was partly performed in 2008, and will be finalized in 2009.

**T3 - R&D on Advanced Neutron Multiplier for DEMO Blanket**

Recent progress was made in the following areas:

- preparation of fabrication devices at FZK (ie, microwave sintering, wet ball milling, arc melting, etc);

- successful fabrication of large rods of Be-30.8 wt%Ti (about 30 mm in diameter);

- characterization of Be and beryllides, i.e., chemical composition, distribution of precipitation (PIXE, TEM with EDX, EELS) and crystalline structure (X-ray diffraction analysis) etc.;

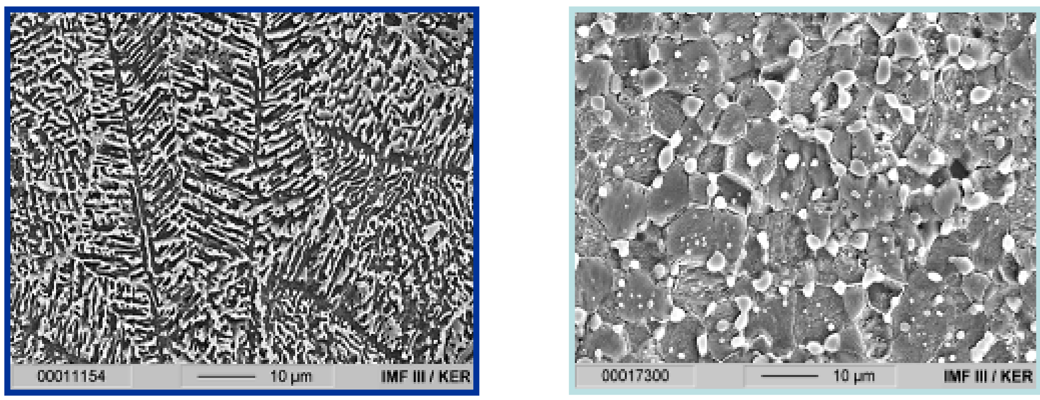
- tritium release experiments at room temperature - 850 °C (first results).

The EU schedule foresees the assembling of the equipment and licensing in the period 2007-2010, and fabrication of beryllide rods and characterization in the period 2009-2014).

**T4 - R&D on Advanced Tritium Breeders for DEMO Blanket**

The main activities were focused on development of advanced breeder pebbles of lithium orthosilicate ($Li_4SiO_4$) including its production and reprocessing technology. Progress was made in the following areas:

- production tests of $Li_4SiO_4$breeder pebbles: a microstructure change of breeder pebbles was observed: annealing at 970$^o$C for a week in the production process led to a change in the morphology of the microstructure from the two-phase system $Li_4SiO_4$+ $Li_6SiO_7$ to the other two phase system $Li_4SiO_4$+ $Li_2SiO_3$;




**initial** | **970°C / 1 week**

**$Li_4SiO_4$ + $Li_6Si_2O_7$** | **$Li_4SiO_4$ + $Li_2SiO_3$**

Figure 67 - Microstructure of Li4Si4 + 2.5wt% SiO2 and effects of annealing

- investigation on reprocessing of breeder materials: a comparison was made on reprocessing characteristics between Li4SiO4 by the melt-spraying method and Li2TiO3 by wet chemistry. The results indicated a merit of reduced waste for the melt-spraying method;

- activation calculation of Li4SiO4: activation is mainly driven by Co-60 and Pt-193 radioactive nuclides, which originate from raw material impurity and crucible impurity, respectively.

The EU schedule in this field foresees the preparation of equipment (2007-2009), the production and characterization of advanced pebbles (2010-2012), and reprocessing studies (2011-2013).

**T1 - R&D on SiCf/SiC Composites**

A NITE-SiC/SiC composite, a candidate DEMO reference material, was assessed in 2007 and early 2008, providing information on fundamental material properties. A CVI (chemical vapour infiltration) SiC/SiC composite was also prepared in 2008 as alternative candidate. The mechanical testing system and the specimen holders for electrical resistivity and He/H permeability measurements during irradiation were .designed

**T2 - R&D on Tritium Technology**

The conceptual design work for the multi-purpose radioisotope (RI) handling facility (to be installed in the DEMO R&D Building at Rokkasho) was carried out in 2007 and early 2008. On this basis, detailed design studies were carried out, including general layout, plant flow diagram and the general arrangement drawings of main components (e.g. the tritium removal system and the waste water treatment system). The radiation exposure of workers in the facility and the amount of tritium released to the environment were evaluated, providing input to the preliminary application for the licensing of the multi-purpose RI handling facility.

**T3- R&D Materials Engineering for DEMO Blanket**

The fabrication and chemical/metallographic analyses of plates of reduced activation ferritic/martensitic (RAFM) steel (designated as F82H-BA07) were performed in 2007 and early 2008. This material was used for weld technological trials and creep tests on welded joints.

**T4- R&D on Advanced Neutron Multiplier for DEMO Blanket**

In 2007 and early 2008, a conceptual design and database preparation for the licensing of the Beryllium Handling Facility (to be installed in the DEMO R&D Building at the BA Rokkasho site) were carried out. The work proceeded in 2008 with the detailed design of the test equipment and the occupational safety equipment for the beryllium handling facility.

**T5- R&D Advanced Breeder for DEMO blanket**

In 2007 and early 2008, a conceptual design work was carried out on the equipment for production of advanced breeder pebbles. This work was followed in 2008 by preparatory design of the equipment as well as preliminary investigations on the dissolution and purification steps of the pebble fabrication process.

**DEMO R&D Activity for preparation of Rokkasho facility and equipment**

The conceptual design for preparing the facility and equipment for DEMO R&D was carried out until March 2008, as “the initial urgent task for DEMO R&D for the IFERC”. The design works aims at preparing i) the multi-purpose RI handling facility including the tritium handling room (task T2), and the material test and nano-analysis rooms (tasks T1-T5), ii) the beryllium handling facility (task T4), iii) a non-radioactive experimental room for R&D of tritium breeder materials ( task T5) and for other R&D fields, and iv) related equipment and apparatuses. The conceptual design works above were followed by detailed design works in 2008, which successfully led to preparation of a preliminary application for licensing of the multi-purpose RI handling facility. These results obtained in 2008 will allow the actual licensing works for the RI handling facility in 2009, which respects the schedule for the completion of the DEMO R&D Building by the end of March 2010.

**Coordination and review meetings**

The progress in DEMO R&D was presented in three workshops: Rokkasho, Japan (3-5 July 2007); Tokyo, Japan (January 2008); and Frascati, Italy (July 2008).

**DEMO Design Activities**

The DEMO design, in 2007 and 2008, proceed as EU and JA domestic activities. The IFERC programme is making an effort to establish links and coordination between the EU and JA Implementing Agencies, using the tool of DEMO workshops. Three workshops took place in 2007/2008 where a broad spectrum of physics and engineering aspects were discussed together with possible development strategies.

EU and JA views on the roadmap to DEMO presently differ on substantial points and in view of a coordinated programme, in the second phase of the BA, it has been agreed to concentrated the 2009 work on a set of critical design issues and organisational aspects.

**CSC activities**

Five technical meetings and a number of videoconferences have taken place in 2007 – 2008 to define a schedule for the implementation of the Computer Simulation Centre, provide information for finalizing the specifications for the building based on the analysis of likely requirements for a Petaflops scale computer in early 2012, specify the interfaces between the building and the super-computer and refine the procurement sharing. At the end of 2008 there is a general overall agreement on the sharing, the schedule, and only the cooling system and the electric power interfaces remain to be agreed in detail. Regarding the network access to the future computer center, assessment of existing lines has been made including preliminary performance tests.

**Activities of the Special Working Group 1 (SWG1)**

SWG1 was established by the Steering Committee in July 2008 to represent the needs of the future users of the super-computer. The group is composed of four members per Party.

The charges given to SWG1 include

- defining the areas of applications (mid. of 2008),
- drawing up the actual minimum needs (end of 2010),
- conducting market research (mid. of 2010),
- selecting benchmark codes (end of 2008),
- assessing the performance of the benchmark codes on candidate high performance computer systems (mid of 2011),
- reviewing the technical specifications for the super-computer, formulated by the procuring IA (end of 2010),
- defining user requirements for maintenance, operation and technical support (end of 2010), and
- monitor the acceptance tests and the early operating phase of the super-computer system (mid. of 2012).

The SWG1 has completed the tasks scheduled for 2008 and, after analysing 13 codes relevant for a wide range of simulations in the field of fusion, delivered a selection of seven high-level benchmark codes covering all the areas of application (3 from JA, 3 from EU and one widely used US code). The SWG1 has also defined a timetable for decisions and actions for the finalisation of CSC specifications and made good progress on the topics of intellectual property rights and codes confidentiality.

Annex

# I

# List of Acronyms

| | |
|---|---|
| A/E | Architect Engineer |
| AC | Alternating Current |
| AGPS | Accelerator Power Supplies |
| ALARA | As Low As Reasonable Achievable |
| ANB | Authorized Notification Body |
| ANS | Analytical System |
| AVDEs | Asymmetric Vertical Displacement Event |
| ATS | Air Transfer System |
| BA | Broader Approach |
| BSM | Blanket Shield Module |
| BTP | Build-to-Print |
| CEA | Commissariat a l'Energie Atomique |
| C&I | Control and instrumentation |
| CFC | Carbon Fibre Composites |
| CMM | Cassette Multifunctional Mover |
| CMQP | Common Management and Quality Programme |
| CRYO2 | Analysis of the conceptual design and layout of the ITER cryoplant system |
| CRYO3 | Analysis of ITER cryogenic system functionality |
| CS | Central Solenoid |
| CVB | Cold Valve Boxes |
| CVD | Chemical Vapour Deposition |
| CXRS | Core Plasma Charge-Exchange Recombination Spectroscopy System |

| | |
|---|---|
| **D** | Deuterium |
| **DA** | Domestic Agency |
| **DACS** | Data Acquisition and Control System |
| **DBTT** | Ductile to Brittle Transition Temperature |
| **DC** | Direct Current |
| **DCLL** | Dual Coolant Lithium Lead |
| **D-D** | Deuterium-Deuterium |
| **DEMO** | Demonstration Fusion Reactors |
| **DGEBF** | *Di-Glycidyl Ether of Bisphenol F* impregnation resin |
| **DMS** | Document Management System |
| **DNB** | Diagnostic neutral beam |
| **D-T** | Deuterium-Tritium |
| **DTP** | Divertor Test Platform |
| **EAF** | European Activation File |
| **EB** | Electron Beam |
| **EBBTF** | European Breeding Blanket Test Facilities |
| **EC** | Electron Cyclotron |
| **EC UL** | Electron Cyclotron Upper Launchers |
| **ECH** | Electron Cyclotron Heating |
| **EFDA** | European Fusion Development Agreement |
| **EFF** | European Fusion File |
| **ELM** | Edge Localized Mode |
| **EPC** | Engineering Procurement Contract |
| **ESOF** | Euro Science Open Forum |
| **EU** | European Union |
| **EUROFER** | A 9% Cr reduced activation ferritic-martensitic steel |
| **EUROFER ODS** | Oxide Dispersion – Strengthened version of EUROFER steel |
| **EVEDA** | Engineering Validation and Engineering Design Activities |
| **F4E** | Fusion for Energy |
| **FS** | Functional Specification |
| **FW** | First Wall |

| | |
|---|---|
| **FWP** | First Wall Panel |
| **FZK** | Forschungszentrum Karlsruhe |
| **HAZOP** | Hazard Operability studies |
| **HCLL** | Helium-Cooled Lithium-Lead |
| **HCPB** | Helium Cooled Pebble Bed |
| **H&CD** | Heating & Current Drive |
| **HFTM** | High Flux Test Module |
| **HIP** | Hot Iso-static Pressing |
| **HISS** | Hydrogen Separation System |
| **HNB** | Heating Neutral Beam |
| **HV** | High Voltage |
| **HVAC** | Heating Ventilation & Air Conditioning |
| **HVD** | High Voltage Deck |
| **HW** | Hardware |
| **IAEA** | International Atomic Energy Agency |
| **IC** | Ion Cyclotron |
| **ICH** | Ion Cyclotron Heating |
| **IDR** | Integrated Design Report |
| **IFERC** | International Fusion Energy Research Centre |
| **IFMIF** | International Fusion Materials Irradiation Facility |
| **INB** | Installation Nucleaire de Base |
| **IO** | ITER Organisation |
| **IR** | Infra Red |
| **ISEPS** | Ion Source and Extraction Power Supplies |
| **ISS** | Isotope separation system |
| **ITA** | ITER Task Agreement |
| **IVT** | Inner Vertical Target |
| **IVVS** | In-Vessel Viewing System |
| **JAEA** | JA Implementing Agency |
| **JA HT** | JA Home Team |
| **LFS-CTS** | Low Field Side – Collective Thomson Scattering |

| | |
|---|---|
| **Li** | Lithium |
| **LN2** | Liquid Nitrogen |
| **LPCE** | Liquid Phase Catalytic Exchange |
| **MAR** | Materials Assessment Report |
| **MFG** | Motor Flywheel Generators |
| **MHB** | Material Handbook |
| **MHD** | Magneto-HydroDynamics |
| **MIG** | Metal Inert Gas |
| **MV** | Medium Voltage |
| **NB** | Neutral Beam |
| **NBI** | Neutral Beam Injector |
| **NBPS** | Neutral Beam Power System |
| **NBTF** | Neutral Beam Test Facility |
| **ODS** | Oxide Dispersion Strengthened |
| **P&ID** | Process and Instrumentation Diagram |
| **PA** | Procurement Arrangement |
| **PF** | Poloidal Field |
| **PFC** | Plasma Facing Components |
| **PFD** | Process Flow Diagram |
| **PIE** | Post Irradiation Examination |
| **PMU** | Prototypical Mock-Up |
| **PP** | Procurement Package |
| **PPC** | Pre-Production Cryopump |
| **PrSR** | Preliminary Safety Report |
| **PTC** | Prototype Torus Cryopump |
| **$Q_{1/2/3/4}$** | Quarter |
| **QA** | Quality Assurance |
| **QPC** | Quench Protection Circuit |
| **R&D** | Reasearch & Development |
| **RAFM** | Reduced Activation Ferritic Martensitic |
| **RF** | Radio Frequency |

| | |
|---|---|
| **RH** | Remote Handling |
| **RMP** | Resonant Magnetic Perturbation |
| **RNC** | Radial Neutron Camera |
| **RWM** | Resistive Wall Mode Control |
| **SDC** | ITER SDC (Structural Design Criteria/Code) |
| **SHPC** | Safety and Health Protection Coordination |
| **Sic-Dual** | SiC/SiC composite material for electrical and thermal Insulation (for use in Dual Coolant Breeder Blankets) |
| **SOFT** | Symposium on Fusion Technology |
| **SS** | Steady State |
| **STP** | Satellite Tokamak Programme |
| **SW** | Software |
| **T** | Tritium |
| **TBM** | Test Blanket Modules |
| **TES** | Test Extraction System |
| **TF** | Toroidal Field |
| **TH** | Thermal Hydraulical |
| **TIG** | Tungsten Inert Gas |
| **UT** | Ultrasonic |
| **VC** | Voluntary Contributor |
| **VS** | Vertical Stability |
| **VV** | Vacuum Vessel |
| **WAVS** | Wide Angle Viewing System |
| **WBS** | Work Breakdown Structure |
| **WDS** | Water Detritiation System |
| **WP** | Work programme |

Annex

# II

# Key Outputs

## List of Articles, Reports and Conference Papers Published in 2007 and 2008

1. SATIR inspection of the ITER vertical target qualification mock-ups manufactured by EU industry.
A Durocher et al,
Proceeding of the 25th Symposium on Fusion Technology, Rostock, September 15-19, 2008 - to appear in Fusion Engineering & Design.

2. Manufacturing issues for the production of beryllium clad prototype components for ITER.
P Sherlock, P Lorenzetto, N Walton , S Keil,
Proceeding of the 25th Symposium on Fusion Technology, Rostock, September 15-19, 2008 - to appear in Fusion Engineering & Design.

3. Mechanical testing of a PFW panel attachment system for ITER,
V Oliva et al,
Proceeding of the 25th Symposium on Fusion Technology, Rostock, September15-19, 2008 - to appear in Fusion Engineering & Design.

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F Escourbiac et al,
Definition Proceeding of the IAEA conference, October 2008.

5. EU considerations on design and qualification of PFCs for near term machines (ITER),
P Lorenzetto,
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6. Manufacturing and testing of Vertical Target Qualification Prototypes for ITER,
T Huber et al,
Proceedings of the Plansee Seminar, Plansee.

7. The European contribution to the procurement of the ITER Remote Handling systems,
C Damiani et al,
Paper presented at the 25th SOFT Conference in Rostock.

8. Procurement of the ITER Magnet Components Supplied by F4E,
C Sborchia, A Bonito Oliva, H Rajainmäki and A Vostner,

Paper presented at the 25th SOFT Conference in Rostock, submitted for publication to Fusion Engineering and Design.

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I Cristescu et al,
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13. Investigation of a Test Loop for the Cooling System of the ITER TF Coil under pulsed heat load,
B Rousset, A Girard, S Maze, JM Poncet, P Roussel, V Kalinin, D Murdoch, M Sanmarti,
CEC/ICMC'22 Chattanooga, USA, July 2007.

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J Jaquinot et al,
SOFT 25th Symposium on Fusion Technology, September 15-19.

15. Progress in the design of the Neutral Beam Injector for ITER,
V Antoni et al,
IAEA FEC, 12-18 October 2008.

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17. The ITER full size plasma source device design,
P Sonato,
SOFT 25th Symposium on Fusion Technology, September 15-19.

18. Review of the European Programme for the development of the gyrotron for ITER,
F Albajar et al,
15th Joint Workshop on Electron Cyclotron Emission and Electron Cyclotron Resonance Heating, Yosemite, 2008.

19. Status of experiments on the EU 2MW coaxial cavity ITER gyrotron pre-prototype at FZK,
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P Moreau et al,
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Rev Sci Instrum 79 10F104 (2008)

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M Beurskens et al,
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