EUROPEAN Contribution to the Design and R&D Activities in View of the Start of the ITER Construction Phase

Presented by M. Gasparotto on behalf of the EFDA CSU Team

EFDA Close Support Unit Garching, Boltzmannstr. 2, Garching by Munich, Germany

e-mail contact: maurizio.gasparotto@tech.efda.org

Abstract. The European effort in supporting the ITER design and R&D programme was maintained at a considerable level in the last four years in order to be ready to start the construction phase as soon as the ITER site was decided and the ITER Team established. This paper describes the status of the design and R&D activities conducted in Europe highlighting the major achievements. Particular emphasis is given to areas where R&D is still required: Remote Handling, ITER Test Blanket Modules, Diagnostics and Heating and Current Drive Systems.

1. Introduction

The European Fusion Technology programme, under the coordination of EFDA (the European Fusion Development Agreement established in 1999 as a framework contract between EURATOM and the European Associates) over the past few years has been mainly devoted to:

1. the completion of design and R&D studies in preparation for the procurement of ITER systems and components in close collaboration with the ITER team and according to the ITER design and schedule;
2. the provision of support to European Associations and Industries in key areas of fusion R&D to ensure a competitive and timely approach to the planned procurement.

The EU contribution to ITER design and R&D activities has been maintained at a significant level with the objectives of:

1. continuing, and in some cases expanding, the effort in areas where design and R&D were still required: in particular in Remote Handling, ITER Test Blanket Modules, Diagnostics, Heating and Current Drive Systems;
2. continuing and completing manufacturing R&D to determine the most technically and cost effective manufacturing methods for ITER components to be built in Europe;
3. preparing new test facilities needed during ITER construction;
4. supporting the European site preparation process and the preparation of safety and licensing documentation for ITER in Cadarache;
5. maintaining support to EU industries in R&D activities of relevance to fusion.

At the end of 2005 the procurement cost sharing of the ITER components and sub-systems was agreed by the seven ITER parties. Europe, as major contributor, will provide 10 Toroidal Field magnet windings, the Poloidal Field coils from N. 2 to N. 6, 80% of the Vacuum Vessel main body, 30% of the Blanket First Wall, the Inner Divertor Target and the Divertor Cassette integration. Important contributions are also planned in Remote Handling, Heating and Current Drive Systems, Diagnostics, Vacuum Pumping and Fuelling, Power Supply Systems, Tritium Plant, Cryoplant, Safety, Site preparation (including licensing) and Buildings. In the following sections the status of the ITER design and R&D activities developed in Europe is reported, highlighting the main achievements. Particular emphasis is given to areas where R&D is still required, such as Remote Handling, ITER Test Blanket Modules, Diagnostics and Heating and Current Drive Systems.

2. The ITER Core Load Assembly

2.1. Vacuum Vessel

The ITER Vacuum Vessel (VV) consists of nine sectors 11.4 m high, 200 tonnes weight fabricated from 60 mm thick Stainless Steel (SS) plates arranged in a double wall structure.
The reference VV fabrication route, which includes heavy jigs to limit the amount of shrinkage linked to the high concentration of welded zones, has been verified by the construction of a full-sized, 20 Ton VV Poloidal Sector Mock-Up (VVPSM). Figure 1 shows the ribs of the upper inboard segment being welded to the inner wall. The final part of the construction of the VVPSM involves the welding together of the two partial segments, simulating the rest of the Vessel Sector by a large structural steel construction weighing about 50 tons, which can be partially dismantled to simulate the effect of the removal of the jigs.

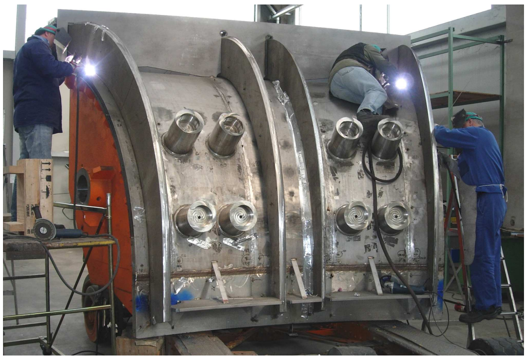

FIG. 1. Mock of VV Poloidal Sector

The manufacturing programme is accompanied by a complementary analysis on the modelling and prediction of distortions during manufacture, including instrumented mock-ups representing each type of weld to validate the local deformation predicted by computer models, which will be used to design the method for controlling the critical segment–to-segment welding by the application of additional compensation welding to the jigs.

The results of this work programme will ensure that the firms competing for the main VV Sector contract will be provided with sufficient information to enable them to have confidence in achieving the tight tolerances of better than ± 10 mm required from the construction, far smaller than normally achieved for such a size of vessel.

2.2. In Vessel Components

The First Wall (FW) panels of the blanket modules consist of a bi-metallic structure made from a 20-mm thick Cu alloy heat sink layer bonded to a 40-mm thick 316L(N)-IG SS backing plate. A 10-mm thick Be layer is used as the protective plasma facing material and is bonded to the Cu alloy layer in the form of tiles.
An extensive development work programme performed in Europe has allowed to produce very good Be/CuCrZr alloy joints by HIPping [1]. Performances achieved with representative

FW mock-ups (30000 cycles at 0.6 MW/m²; no Be detachment up to 3 MW/m²) exceed the ITER design requirements. A neutron irradiation programme is still in progress to complete the full characterisation with irradiated mock-ups.
Brazing has also been considered as an alternative technique, potentially cheaper, for joining Be tiles to the CuCrZr heat sink layer. Furnace brazing was developed during the ITER Engineering Design Activities (EDA) for joining Be to CuAl25 alloy. It was found that this braze alloy had poor wetting properties at the brazing temperature of interest for CuCrZr alloy and the quality of the product was changeable. Fast induction brazing techniques (see Fig. 2) are chosen to minimize the holding time at high temperature and consequently retain enough mechanical properties of the CuCrZr alloy; a silver free braze alloy is under development. Preliminary results are promising and after completion of this development phase, a scale-one FW panel prototype will be brazed and then thermal fatigue tested.

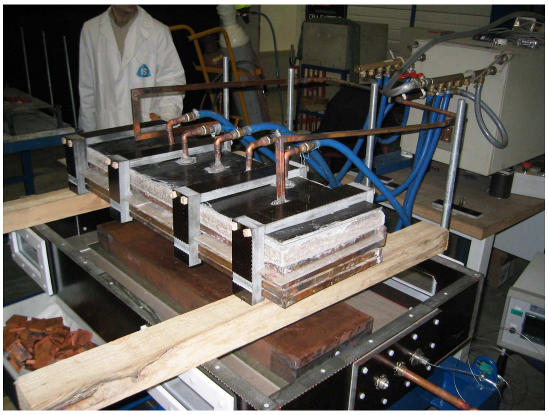

FIG. 2. Induction Brazing of FW Panel

The Shield Block (SB) of the blanket modules consists of a 316L(N) SS massive structure of typically 1.5 m length, 1 m height and 0.4 m thickness. Conventional fabrication techniques were first considered for the manufacture of the SB made from SS forgings [2]. This fabrication route resulted in a very large number of welds, increasing the risk of water leakages inside the vacuum vessel during ITER operation and in a lot of expensive machining and welding operations. An alternative design and fabrication route have also been developed to increase the reliability of the components e.g. by removing all the seal welds. This alternative design is based on the experience gained during the ITER EDA by fabricating a SB prototype by powder HIPping [3]. A complete cooling tube gallery was fabricated and HIPped with 316L(N) SS powder at 1100°C and 130 MPa for 4 hours.

Because of the EU responsibility in the procurement of the divertor Inner Vertical Target and of the divertor Cassette Body, several activities are being carried out with the EU industry and with the EU national laboratories. High heat flux technologies have been developed and a number of small, medium and full-scale prototypes have been successfully manufactured by Plansee (Austria), Ansaldo Ricerche (Italy) and Areva-Framatome (France) and tested at heat fluxes well above the ITER requirements [4]. Carbon Fiber Composite prototypes have been successfully tested up to 1000 cycles at 23MW/m² (ITER design target: 300 cycles at 20 MW/m²) and tungsten prototypes up to 1000 cycles at 10 MW/m² (ITER design requirement for the divertor baffle: 3 MW/m²). Recently, another medium-scale vertical target prototype (see Fig. 3) was manufactured by ENEA Frascati. It is aimed at demonstrating whether the “hot radial pressing” technology can be a viable alternative manufacturing route for the vertical target. High heat flux testes are underway in the FE200 electron beam facility (located in Le Creusot, France, and operated by Areva-Framatome and CEA Cadarache) and preliminary results (3000 cycles at 10 MW/m²) are encouraging.

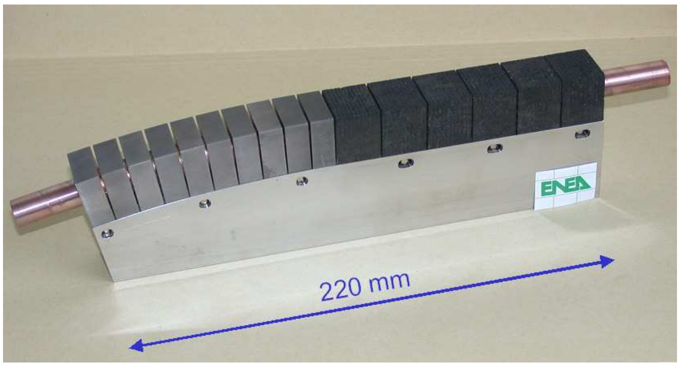

FIG. 3. Medium Scale Divertor Vertical Target Prototype

A first study on acceptance criteria for the ITER divertor plasma-facing components (PFCs) was completed in collaboration with CEA Cadarache. A review of the existing experimental database on PFC performances and defectology was carried out. On this basis, the main strategy towards the definition of workable acceptance criteria was defined in agreement with the ITER International Team. More than one hundreds mock-ups with calibrated artificial defects were designed and are being manufactured by Plansee and Ansaldo Ricerche. Their high heat flux testing will form the experimental basis against which the present assumptions on the acceptance criteria will be finally validated and refined.

2.3. Magnet System

The superconducting magnet system equals almost one third of the total estimated direct capital cost for the experimental reactor ITER. It consists of two main systems: the Toroidal Field coils (TF) and the Poloidal Field coils (PF) (including the Central Solenoid coil (CS)). The CS and the TF coils will be manufactured using Nb_3Sn conductor due to the high magnetic field and required current. The PF coils and the Correction coils will be manufactured using NbTi conductors.

The basic conductor configuration is circular cable-in-conduit type cooled by a forced flow of supercritical helium. The cable is formed by multi-stage cabling of superconducting strands with the final stage consisting of six bundles twisted around a central cooling channel.

During the ITER-EDA phase, two models have been manufactured and tested to demonstrate the feasibility of the ITER coil system: the Toroidal Field Model Coil (TFMC) and the Central Solenoid Model Coil (CSMC).

Both, the TFMC and CSMC achieved their design values without quenching, but the temperature margin was found to be about 1 K less than expected. Recent (2006) tests of ITER-like Nb_3Sn conductors (not the exact ITER configuration) in Sultan indicate worst than expected ‘advanced’ Nb_3Sn strand performance in the conductors. The degradation appears to be caused by transverse magnetic loads on the strands. The higher performance strands appear to be more sensitive to this effect. R&D is underway to understand the problems, likely to be related to the design of the conductor, and to address this issue. A Dipole facility is in construction phase at CRPP Lausanne to supplement the existing SULTAN facility for the quality control of the ITER superconducting cables.

The full-size PF conductor in the NbTi has not yet been tested in long length and pulsed field. Therefore Europe has manufactured a PF Conductor Insert (PFCI) coil with cable provided by the Russian Federation. The PFCI is a single-layer wound solenoid consisting of nine turns. It is planned to test such insert at the CSMC facility in Naka (Japan).

2.4. Remote Handling

Within the Remote Handling (RH) area, according to the latest ITER procurement sharing arrangements, Europe will provide the divertor handling equipment, 50% of the transfer cask system, the Neutral Beam RH equipment and the tokamak in-vessel viewing system.

The exchange of the divertor is a vital maintenance activity for the long-term operational success of ITER and is expected to be necessary after every 3-4 years of plasma operations.
Based on the above requirement and the likelihood that Europe would play a major role in the eventual supply the divertor RH system, Europe has maintained an aggressive campaign of design and prototyping in this area since the mid-1990's. Initially this manifested itself through the construction of the Divertor Test Platform (DTP) in Brasimone, Italy, as the centre place of the ITER L7 large R&D project (1995-2002). However, major geometrical changes in the divertor region in the transition between ITER'98 and ITER 2001 lead to a decision to effectively repeat the L7 exercise by creating a new RH test facility (the DTP2) in Tampere, Finland (see Fig. 4). Construction of this facility began in Spring 2006.
The overall aim of the DTP2 project is to ensure that the cassette movers supplied to ITER during its construction are be based on well matured "second generation" designs which have benefited from the experience and lessons learnt from the building and operation of a first generation of prototypes.

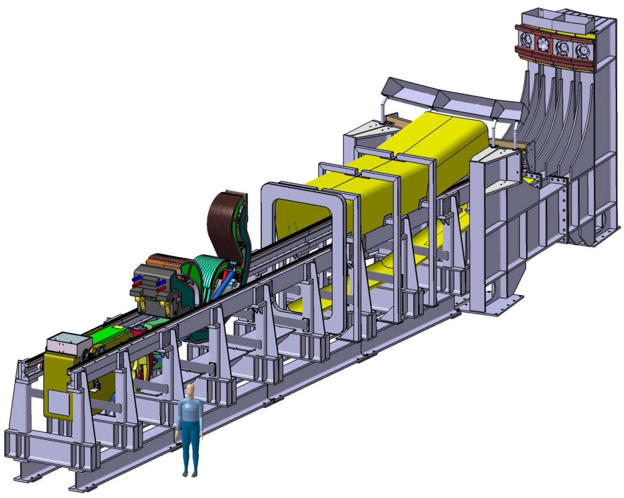

FIG. 4. The DTP2 RH Mock-Up Facility

In ITER, a system of remotely controlled casks will be used to transfer highly activated and contaminated in-vessel components between tokamak and hot-cell. These casks will travel by way of remotely controlled transporters floating on a system of low friction air bearings. Since the late 1990's Europe has provided steady support to the ITER Team in this area in terms of engineering design development which has kept pace with the design evolution of the cask interfaces i.e. in-vessel components, in-cask vehicles, tokamak and buildings. Significant work is still needed in this area to finalise the engineering designs in preparation for the ITER procurement exercise, especially in relation to the cask-vacuum vessel interface and port plug handling.

2.5. Diagnostics

ITER will have a comprehensive plasma measurement system consisting of about 45 individual diagnostic systems. The integration of these diagnostic systems into the ITER nuclear environment will present significant new challenges and demand a level of engineering integration beyond those required in present experiments. An extensive programme of design and R&D focussing on those diagnostic systems for which the EU will likely have procurement responsibilities has been implemented to develop designs for ITER diagnostic systems, to develop components capable of performing reliably in the ITER environment and to address the generic issues associated with the engineering and integration of diagnostic systems for ITER [6].

This programme encompasses a wide range of measurement techniques, such as LIDAR Thomson scattering, neutron emission, density reflectometry, active charge exchange spectroscopy (CXRS), bolometry, magnetic measurements, neutral pressure measurements, and visible and infrared observations of plasma-wall interactions. Progress has been made in the design analysis for each of these systems. For example, in the magnetics diagnostic, which has many sensors and components requiring early delivery to allow integration into the vacuum vessel and divertor, designs are being developed for a variety of sensor types, and R&D on prototypes is underway to establish construction and manufacturing methods. Common design topics for optical diagnostics, such as the LIDAR and CXRS, include the simplification of labyrinths against neutron streaming and the development of concepts which allow the replacement of sensitive optical components (relevant for many optical diagnostics). Most optical diagnostic systems require front-end bulk-metal mirrors, which are exposed to large fluxes of energetic particles and experience deposition of sputtered material. R&D is underway to develop suitable low-sputtering mirror materials, such as rhodium coatings and monocrystalline molybdenum, to explore techniques for erosion and deposition mitigation, and to study in-situ cleaning techniques for mirrors. A specific R&D programme is addressing the development of radiation hard ceramic materials and components for in-vessel systems, including bolometers, pressure gauges, mineral-insulated cables and windows [7,8].
Major diagnostic sub-systems will be integrated into port-plug structures which provide mechanical support, nuclear shielding and cooling for the diagnostic components. In addition, they must provide an environment in which diagnostic components can operate reliably. The EU has made significant contributions to the engineering design of port plug structures, to analysis of fabrication techniques and to studies of diagnostic integration into the port plugs [9]. The development of testing facilities for integrated diagnostic (and heating and current drive) port plugs in which the full functionality of the systems can be tested, resilience to forces and vibrations can be demonstrated and where leak testing can be performed will be a significant challenge for the EU's diagnostic development programme in the future.

2.6. Heating and Current Drive (H&CD)

2.6.1. Electron Cyclotron Resonance Heating

The ITER Electron Cyclotron (EC) H&CD system is designed to inject high power millimetre waves through equatorial and upper ports of ITER so as to deposit power locally in the plasma via electron cyclotron resonance absorption. The primary system consists of 24 gyrotrons, each operating CW at 1 to 2 MW so as to deliver 24 MW to the plasma, either via a single launcher (with 24 waveguides) in an ITER equatorial port, or up to 4 upper launchers (each containing 8 waveguides). The capability for localized heating and current drive and the

ability to vary the deposition location over a significant fraction of the ITER plasma makes the system well adapted to a range of functions including plasma heating, current profile control and magnetohydrodynamic (MHD) mode control.
The EU will have procurement responsibility for the supply of the upper launchers, a fraction of the gyrotrons and the main gyrotron power supplies and has been pursuing an extensive programme of design and R&D in preparation for the supply of these sub-systems. The upper launcher is designed to stabilize MHD modes, primarily the so-called neoclassical tearing modes (NTMs) at the q=3/2 and q=2 surfaces, under a range of ITER plasma conditions, and therefore must be capable of steering the mm-wave beams to ensure deposition can range over the majority of the plasma minor radius beyond r/a = 0.6. In addition, the launcher optics must ensure sufficient beam focussing to achieve adequate driven current densities to suppress the MHD modes. An extensive design and R&D programme is studying two variants of the launcher: *front steering* (FS), with the steering mirrors located in the port plug behind the blanket shielding module (the ITER reference solution) and *remote steering* (RS), with the steering unit external to the port plug in the ITER secondary vacuum [10].
The superior capability of the FS approach for maintaining a highly focussed beam over a wide steering range leads to a high stabilization efficiency at both the q=3/2 and q=2 surfaces over the range of scenarios investigated. Resources within the EU programme, therefore, are principally directed towards the FS concept (see Fig. 5), while the RS concept is still pursued as possible back-up option. The engineering of the steering mechanism, in terms of both design and material choice, has received considerable attention. The mechanism is based on a pneumatically driven, frictionless and self-centring mechanism, which is backlash free. This design should avoid the main problem encountered in present day machines, which is the seizing of moving parts due to friction. The design of the port plug (structure and shielding elements), as well as that of the CVD-diamond window, is being adapted to the FS concept, taking advantage of solutions previously developed for the RS launcher that were generic enough for easy adaptation. The flexibility afforded by this concept has resulted now in a 4-launcher system that has the ability to stabilise NTMs over the required radial range, as well as to drive current in the region of q=1, sharing part of the duties of the equatorial launcher for sawtooth control. The projected stabilisation performance is high enough that only ~ 60% of the available power should be required for total mode suppression in most of ITER scenarios.

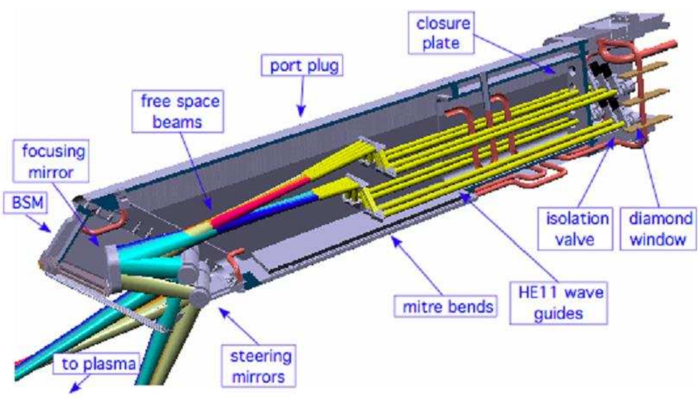

FIG. 5. The front steering ECRH upper launcher for MHD mode control

The European effort for the EC power sources is centred on the development of a 170 GHz, 2 MW, CW coaxial cavity gyrotron of Collector Potential Depressed (CPD) type. The essential arguments for implementing an R&D Programme aimed at developing a gyrotron with higher unit power compared to the ITER baseline (1 MW) are: to reduce the cost of the installation and to allow a compact upper port launcher design (or, conversely, to increase the power delivered from one port). Coaxial cavity gyrotrons have the potential to allow high unit power to be achieved and 2.2 MW in single mode were demonstrated in short pulse operation. On a so-called pre-prototype gyrotron, the nominal $TE_{34,19}$ mode at 170 GHz has been excited stably in single-mode operation over a wide parameter range. The construction of the first industrial 2 MW prototype (see Fig. 6) and of the associated superconductive magnet producing 6.86 T on the cavity axis is well underway, while experimental support continues to be provided by tests on the pre-prototype [11]. A new full performances EC test facility has been established at CRPP, Lausanne in support of this programme. Testing on the first gyrotron prototype is expected to start within 2006.

FIG. 6. 2MW-170 GHz prototype produced by THALES Electron Devices

2.6.2. Ion Cyclotron Radiofrequency Heating (ICRF)

The ITER Ion Cyclotron H&CD system is required to couple 20 MW of power from a single antenna (~ 9.3 MW/m² power density at the antenna) in the frequency range of 40~55 MHz [12]. The reference design is based on the "resonant double loop" concept with "conjugate-T matching" for edge localized mode (ELM)-resilience. The EU, which will be responsible for the supply of the ITER ICRF antenna, has implemented an extensive programme of design and R&D studies for such component and its associated matching elements including the ITER-like antenna which will be installed in JET in early 2007. Within the EU programme two matching systems using variable tuning elements are being studied: *internal* (tuning components internal to the antenna port plug) and *external* (tuning components located adjacent to the ITER ICRF port) matching systems. It is anticipated that a final concept selection between these designs will be made in 2007.

The main issue that the ITER antenna has to address is the achievement of reliable coupling at high power density while maintaining good matching in the presence of rapid changes in the loading resistance, such as those produced by ELMs. The advantage of the "conjugate T" matching is that substantial changes in the loading of the antenna do not result in a large amount of power being reflected toward the radio frequency (RF) sources. The principal activities pursued within the EU programme have focussed on the development of the internal and external matching designs to demonstrate that both satisfy the functional requirements for

ITER, and that they can be made compatible with the ITER design constraints, including neutron shielding efficiency, for installation in a mid-plane port.
The matching system is an integral part of the antenna design, and, for both concepts, the study of matching control algorithms and the evaluation of ELM resilience in the presence of inter-strap coupling and of load asymmetries is a key aspect of the design analysis. Results show that matching of the ITER array in these conditions is rather complicated, and that it requires a sophisticated control system.

2.6.3. NB Injection

The procurement of the ITER NB system presents several challenges and a robust R&D programme is necessary in order to achieve the required parameters within the tight constraints due to the ITER construction plan and the close integration of the NB injectors with the ITER machine and tokamak building.
In EU the experimental investigations take mainly place in MANTIS and SINGAP test facilities at CEA, Cadarache where the arc driven source and the concept of an accelerator based on a single gap, alternative to the ITER baseline based on a Multiple Aperture Multiple Gap (MAMuG) design, are pursued and at IPP, Garching, were a RF driven ion source, also alternative to the ITER baseline, is being developed. For SINGAP, important results have been achieved: voltage holding (up to 940 kV) across the 350 mm acceleration gap, efficient trapping of electrons in the pre-accelerator and successful demonstration of beam steering by displacing the post acceleration grid. Progress in the study of essential issues like space-charge interaction between beamlets and beam groups and long pulse effects is impaired by the very limited current capability of the SINGAP tests facility (100 mA) making the testing of the SINGAP concept at higher currents an absolute need.
The ion source development aims at reaching the target performances specified for ITER, namely: current density of 200 A/m^2 in D^- and extracted electron to ion ratio < 1 at a source pressure of 0.3 Pa for a pulse length up to 3600 s. The experimental results from the KAMABOKO III source in MANTIS have allowed to improve the understanding of the operation of the arc driven source which is the ITER baseline. In particular, the importance of controlling the plasma grid and source walls temperature to achieve high negative ion densities was established. The very high Cs consumption was found to be due to Cs trapping on the walls and plasma grid in a mixture with the evaporated W from the filaments. In summary, the achieved current density was, in long pulses, still lower than the one required by ITER and the Cs consumptions remained approximately three orders of magnitude higher than the one anticipated for ITER. Presently, EFDA is concentrating resources on the development of an alternative RF driven ion source. Promising results have been achieved at IPP, Garching in the MANITU facility. Current densities of 330 A/m^2 in H and 250 A/m^2 in D were achieved on a small scale source with good reproducibility and operation in D in ITER relevant parameters space was achieved over many pulses. Finally, experiments on an ITER half-size source are starting. The RF driven ion source can now be considered a valid alternative to the ITER baseline.
The experimental activities on SINGAP and the RF ion source have been supported by a substantial design effort aimed at bringing the two options at the same level of detail of the ITER reference choices (MAMuG accelerator and arc driven ion source mainly developed in Japan). Finally, the limited performances of the existing test facilities do not allow a reliable extrapolation to the ITER scale. The establishment of a full scale test facility is therefore necessary in the ITER NB development plan and has become a centre piece of the European NB development strategy.

3. Safety and Environment

Safety activities have been focused on experimental R&D and safety assessment in support of ITER licensing and computer code validation.

Small-scale experiments have been successfully performed to provide the database for computer model development in relation to the simulation of hydrogen and dust explosion due to air ingress into a vacuum chamber. Energies necessary to trigger explosions of a metal dust cloud injected together with air into a vessel at initial low pressure have been measured. These energies and the pressure transients resulting from the explosions have been quantified for graphite, tungsten, graphite/tungsten mixtures and hydrogen/dust mixtures. It could be shown that the tungsten additive contributed noticeably to the pressure loads, which would be generated by hydrogen alone.

To support model development for ice formation on cryogenic surfaces, which is necessary to analyze the consequences of a loss of coolant (helium and/or water) inside the ITER cryostat, a small-scale device (EVITA) has been set up. A typical example for recent experiments is the injection of superheated water into a small vacuum vessel containing a plate at cryogenic temperatures, with and without presence of non-condensable gas. The results of these tests are being used in an international benchmark exercise, which is being performed under the umbrella of an IEA (International Energy Agency) activity, to validate EU and US computer codes developed and used within the framework of the ITER safety analysis.

To help understanding the behaviour of dust mobilization during accident, a small-scale device called STARDUST has been set up. This device consists of a tank, simulating the ITER vacuum vessel, where different kinds of characterized dusts can be mobilised by means of Helium or air ingress. The dust removed is collected through a filter. Experimental results from this device are also being used for computer code validation inside an international benchmark exercise under the umbrella of an IEA activity.

With the objective to provide analytical tools for the analysis of behaviour and consequences of electrical arcs outside the cryostat, a first arcing model has been developed based on measurements of arc behaviour in a small scale device. The scalability of this model to higher powers will have to be checked by measurements in a slightly larger arcing device under vacuum conditions at present being under testing.

The main safety analyses performed have been concentrated on completion of the database needed to write the ITER Preliminary Safety Analysis Report. These activities included failure mode and effect analysis for the most significant systems and the deterministic assessments of accident sequences, including consequences to the installation and to the workers and the doses to the population in cases of release to the environment of tritium, radioactive dusts and corrosion products.

In parallel to the evolving detailed design of systems and their maintenance strategies, evaluation of occupational doses have been refined, to demonstrate that these doses are ALARA (As Low As Reasonably Achievable).

In the area of radioactive waste management, with the objective to support optimization of the strategy, studies on water and concrete detritiation were carried out.

The results of the studies in this area show that not only the ITER project release guidelines are met, but also the ITER limits for the Cadarache site, with good margin. These limits have been presented in a safety rapport called “Dossier d’Options de Sûreté” (DOS to the French Safety authorities in 2002) by CEA, mandated to act on behalf of a future ITER organisation until the signature of the treaty. Europe has been supporting the writing of the next safety documents to be prepared: the “Rapport Preliminaire de Sûreté, Preliminary Safety Analysis Report”, which must take into account recommendations made by the Safety Authorities following DOS examination, DAC (file for the concession of the authorisation to start construction), and DARPE (file for the authorisation for water intake and effluent emission to start operation) [13].

4. Test Blanket Modules

The demonstration of breeding blankets capable of tritium self sufficiency, efficient heat removal and sufficient shielding is still a challenge on the path to power fusion reactors. In the short term the installation of Test Blanket Modules (TBMs) of the two European concepts (Helium Cooled Lithium Lead – HCLL and Helium Cooled Pebble Bed - HCPB) in ITER will be the first opportunity to test the breeding blanket technology in a fusion environment.
In the HCLL concept the liquid Pb-Li is used as tritium breeding material (Li) and as neutron multiplier (Pb); it is circulated at low speed to extract the tritium outside the blanket and to avoid large pressure drops due to Magnetohydrodynamic effects. In the HCPB concept a He purge gas stream is passing through the Li-ceramic (Li_4SiO_4, Li_2TiO_3) and Be neutron multiplier pebbles and transports the released tritium to the external tritium extraction system.
The engineering design of the TBMs is nearly completed and significant effort is been devoted to develop manufacturing technologies for the production of EUROFER plates with internal rectangular channels for the He circulation at high pressure (8 MPa).

Emphasis has been also given in the EU program to the development of predictive tools that will be further qualified during testing in ITER. For instance:

- For the liquid breeder TBM concept (HCLL), finite elements modelling (CAST3M) has been developed to evaluate the tritium permeation into the He coolant for various PbLi flows and velocity profiles, in particular to take into account the effect of magnetic field.
- For the solid breeder concept (HCPB), pebble bed thermo-mechanical modelling tools have been developed and qualified with experiments in simplified mock-ups. In particular, specific models were implemented in the ABAQUS code to reproduce the non linear elasticity and plastic regions of the mechanical strains and to simulate the pressure-dependent thermal conductance between pebbles and at the pebble-wall interface. This model allowed to reproduce with less than 10ºC discrepancy the experimental results obtained in an ad-hoc experiment called HELICA.

The EU program foresees in the next years the test of medium and large size TBM mock-ups. Existing He facilities in EU allow today testing components up to 1:3 TBM size, but the fabrication of a full TBM size He facilities (HELOKA at FZK/Karlsruhe and HeFUS-3 upgrade at ENEA / Brasimone) has already been engaged and shall be available around 2009.
Although the expected fluence on TBMs will be very limited in ITER (<1-2 dpa), the TBMs will be manufactured using materials and fabrication technologies that are expected to be used also in the future demonstration reactor DEMO.
The structural material in the development and qualification phase in the EU is the Reduced Activation Ferritic Martensitic steel called EUROFER, which consists of 9% Cr W V Ta with W, V, Ta contents in the range of 1.0-1.2%, 0.15-0.25% and 0.10-0.16% respectively. This structural material will be used in the TBMs and will have also suitable properties for its application in DEMO: high radiation resistance up to about 100 displacement per atom, low residual activation, reasonable fracture toughness, high creep strength and good compatibility with the cooling media and Pb-Li. Extensive irradiation tests are in progress to qualify mechanical and tritium release performances of EUROFER, beryllium and ceramic breeders up to high fluence. This allows to identify early material limitations that could have an impact

on the final choice of materials for the TBMs and hence for the future DEMO reactor components.

5. Site Preparation and ITER Buildings

Following the selection of Cadarache as the official construction site for ITER, a number of engineering activities are in progress, for the finalization of the civil and plant engineering design and the expeditious start of tendering for time critical construction activities in 2007-2008. The framework in which the largest share of the technical preparation of the EU site is performed goes under the acronym of EISS, which stays for European ITER Site Studies.
The main activities that refer to the preparation of the ITER construction within the boundary of the fence surrounding the 180 hectares of the Cadarache site (in-fence site preparation) are: the definition of the ITER plant layout including adaptation to the Cadarache site conditions, optimisation of construction logistics and excavation volumes/cost, specifications for site clearing, platform levelling, water supply, sewage, temporary electrical and telecommunications connections, roads and underground networks, drainage and sewage.
In parallel engineering activities are launched utilising qualified engineering companies for the support of ITER for the overall revision of the general layout, nuclear buildings design (with particular regard to seismic design and validation of seismic supports for the foundations) and integration of subsystem with the final objective to prepare the technical specifications for contracting an Architect Engineer that shall support ITER and the European Domestic Agency (DA) in all the activities from detailed design, tender evaluation, and contract follow-up. The process for contracting the site clearing of the site has started with the objective to launch the actual deforestation within January 2007.
In addition, a number of complementary off-fence activities are in progress under French responsibility and financial support. A significant part of the work deals with the road modifications (reinforcement of bridges, adaptations of slopes, relocation of electricity and telephone lines, etc.) along the itinerary selected for the transportation of the largest components from Fos-sur-mer to Cadarache. Works will be conducted between 2007 and mid-2009, date foreseen for the first test convoy. Additional off-fence activities cover the design of the water supply system, the definition of the interface requirements for the delivery of electrical power (including both the 15 kV supply required during construction and the 400kV supply necessary to operate the site with a prescribed value of reactive and active power) and the details of the necessary telecommunication systems.

6. EFDA QA System for ITER Related Activities

French regulatory requirements for QA, specified in the law contained in the ‘Arrete 31-08-1984’ impose that all activities performed by EFDA (and in the future by the European DA in Barcelona) and by its subcontractors, for the design, R&D in support, construction and test of safety relevant components shall respect the quality assurance principles and guideline contained in the IAEA series 50-CQ. It is likely that similarly stringent requirements shall be used also for critical systems and components whose failure, even if not directly impacting the safety might have a large effect on the machine availability.
During 2005, EFDA with the support of experts from fusion associated laboratories has prepared a first draft quality manual, mainly focused to ITER related activities. The approach used, starts from the main process of EFDA technological branch that is contracting (to associated laboratories and industrial companies) design and R&D activities for ITER, and develops a set of rules for the preparation of ‘quality plans’ by the subcontractors. This approach allows to enforce all the technical important aspects of QA without unnecessary

emphasis on formal certification of sub-suppliers. The EFDA QA manual with some adaptation can form the basis of the European DA QA manual.

7. Conclusions

Europe, as the major contributor to ITER, and under the technical coordination of EFDA, has continued to strongly support the project in various areas of design and R&D, in particular focussing in the preparation for procurement of those components where the EU has a strong stake. Work was continued, and in some case extended in areas where design and R&D are still required: in particular in Remote Handling, ITER Test Blanket Modules, Diagnostics, Heating and Current Drive Systems, and to some extent Magnets. Moreover, continuation and completion of manufacturing R&D was carried out in many cases to determine the most technically and cost effective manufacturing methods for ITER components to be built in Europe. New support or R&D facilities are being prepared when needed for ITER construction. The site preparations at Cadarache have been enhanced to follow the project schedule requirements. The effort towards the timely licensing of the facility has also been increased. The forthcoming creation of the EU domestic agency in Barcelona will soon enable a new framework that will be more geared to provide components specified by the ITER Organisation.

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