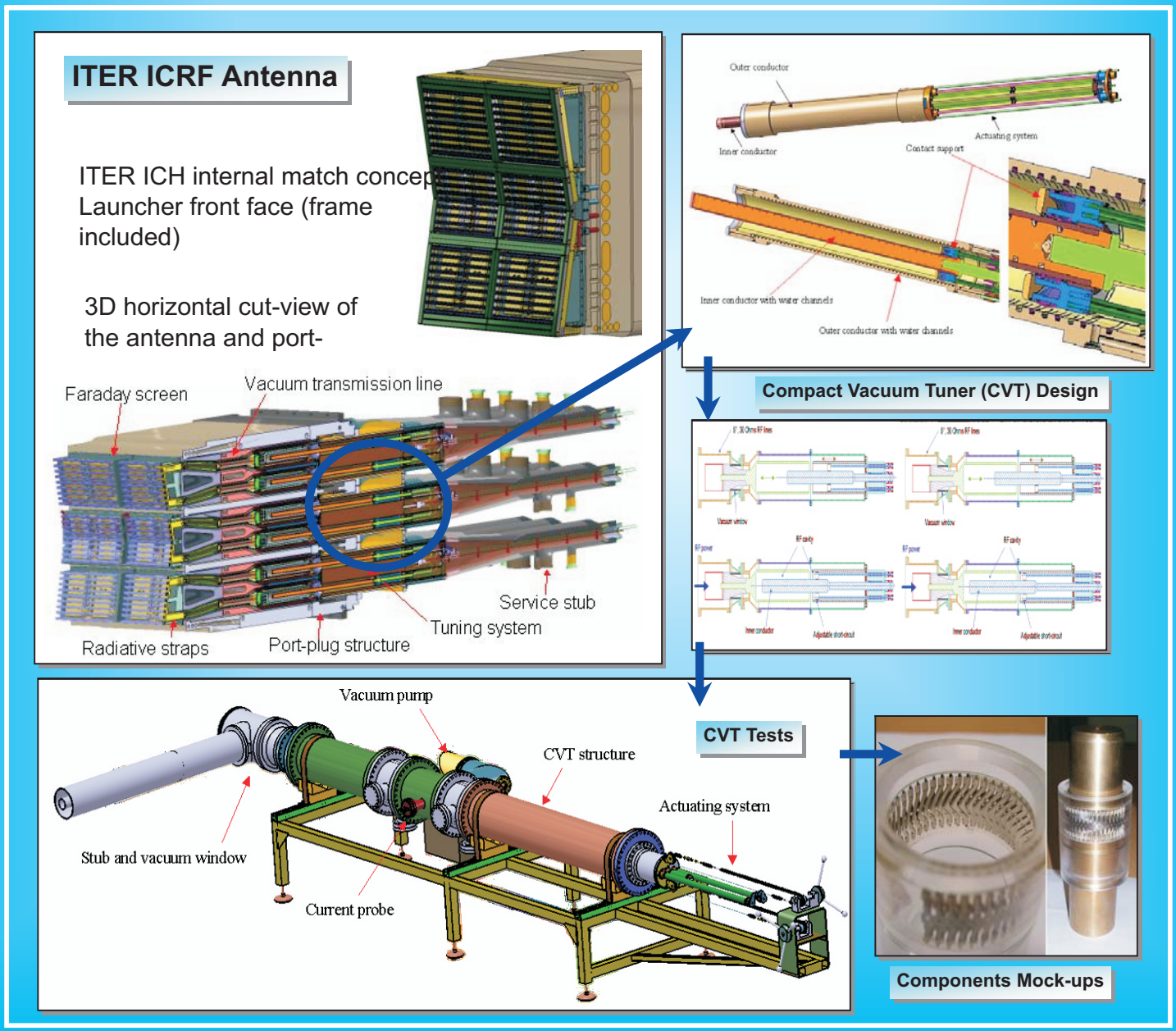
FUSION TECHNOLOGY

Annual Report of the Association EURATOM-CEA 2006

(Executive summary)

Compiled by : Th. SALMON and F. LE VAGUERES



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CEA/DSM/DRFC
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This report and the full report are also available on-line at:
http://www-fusion-magnetique.cea.fr

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Cover: *The Compact Vacuum Tuner (CVT) is a high power tuning device to be developed for use in the ITER Ion Cyclotron Heating launcher. As such, it is designed to be compatible with ITER vessel mechanical interface, EM loads, mechanical, thermal, and nuclear specifications. To validate the CVT, dedicated R&D strategy based on mock-ups has been chosen to assess the most critical aspects. Our Association is involved in the CVT design studies and mock-ups.*

Content

This document is the executive summary of the full annual report which summarizes activities performed by the EURATOM-CEA Association in 2006 within the frame of the European Technology Programme ("EFDA" activities, "Underlying Technology" programme and "Inertial Confinement Fusion (ICF)" keep-in-touch programme).
The full report is available in the enclosed CD-Rom and on line at
http://www-fusion-magnetique.cea.fr

In this document, activities are sorted out according to the main technical topics.
(see the full report for the distinction between the different programmes and a task by task annual reporting; see also APPENDIX: content of the full report).

Introduction

European research on controlled thermonuclear fusion is carried out in an integrated programme with the objective to develop a safe, clean and economically viable energy source. Part of this programme is under the responsibility of the European Fusion Development Agreement (EFDA) which provides a framework covering the activities in the field of technology (both Next Step and Reactor) and the collective use of the Joint European Torus (JET).

This document is the executive summary of the full annual report, summarizing activities performed by the EURATOM-CEA Association in 2006 within the frame of the European Technology Programme ("EFDA" activities, "Underlying Technology" programme and "ICF" programme).

Four specific CEA operational divisions, located on four sites, are involved in the Euratom-CEA fusion technology activities:

- the Nuclear Energy Division (DEN), for In-vessel component design (first wall, divertor, blanket, ...), neutronics, structural materials and safety activities,
- the Technology Research Division (DRT), for activities on materials (elaboration, breeding materials) and robotics,
- the Physical Sciences Direction (DSM), which includes the Controlled Fusion Research Department (DRFC) operating Tore Supra and responsible for plasma physics, cryoplant, magnets and plasma facing components activities,
- the Life Sciences Division (DSV), for activities related to the impact of tritium contamination on staff.

These activities are also completed by specific R&D collaborations with industry, French National Centre for Scientific Research, the Ecole Polytechnique, and Universities.

Progress in fusion technologies has now reached a new era with the beginning of ITER project. Euratom-CEA, together with other European Institutions is on the foreground of technological advances which are of prime importance for the success of the ITER construction. For the forthcoming years, the aim of technological success on every areas of ITER will be a guideline for our Association's programs in technology.
On the longer term, progress in technology which is, on our point of view, a key point for the ITER to DEMO step, will be pursued, in order to improve the vision of an electricity producing reactor and to increase the credibility of fusion energy as a genuine energy for the future. The authors and the editors should be commended for their dedicated contribution in making this report available.

M. CHATELIER

Physics Integration activities

Euratom-CEA activities carried out in the field "Physics Integration" have mainly been dedicated to both **heating and current drive** and **diagnostics** studies.

For heating and current drive, our Association has mainly devoted its efforts in 2006 to the ITER Ion Cyclotron Range of Frequencies (ICRF) Antenna studies, and to SINGAP (negative ion accelerator) project.
Concerning ICRF, two main studies have been pursued, on faraday shield modelling and RF sheath dissipation, and on development of high performance tuning component.
The faraday shield on the ITER ion cyclotron antenna, is the main plasma-facing component for the antenna and must withstand the same heat loads and disruption effects as the first wall. In 2006 our Association work led to the conclusion that there is a need of testing the fall-back dielectric tensor on HFSS, and after assessment of the obtained RF field maps to proceed with RF sheath evaluation, this task will consequently be pursued in 2007.
The Compact Vacuum Tuner (CVT) is a high power tuning device to be developed for use in the ITER Ion Cyclotron Heating launcher. As such, it is designed to be compatible with ITER vessel mechanical interface, EM loads, mechanical, thermal, and nuclear specifications. From a mechanical point of view, the tuner consists in a simple coaxial structure [figure1]. Our Association has begun the tests dedicated to validate the main uncertainties related to this high power tuning device (RF response of the system over the operating frequency band ; voltage stand-off capability of the structure ; operation of the RF contacts inserted in the system sliding under RF current).

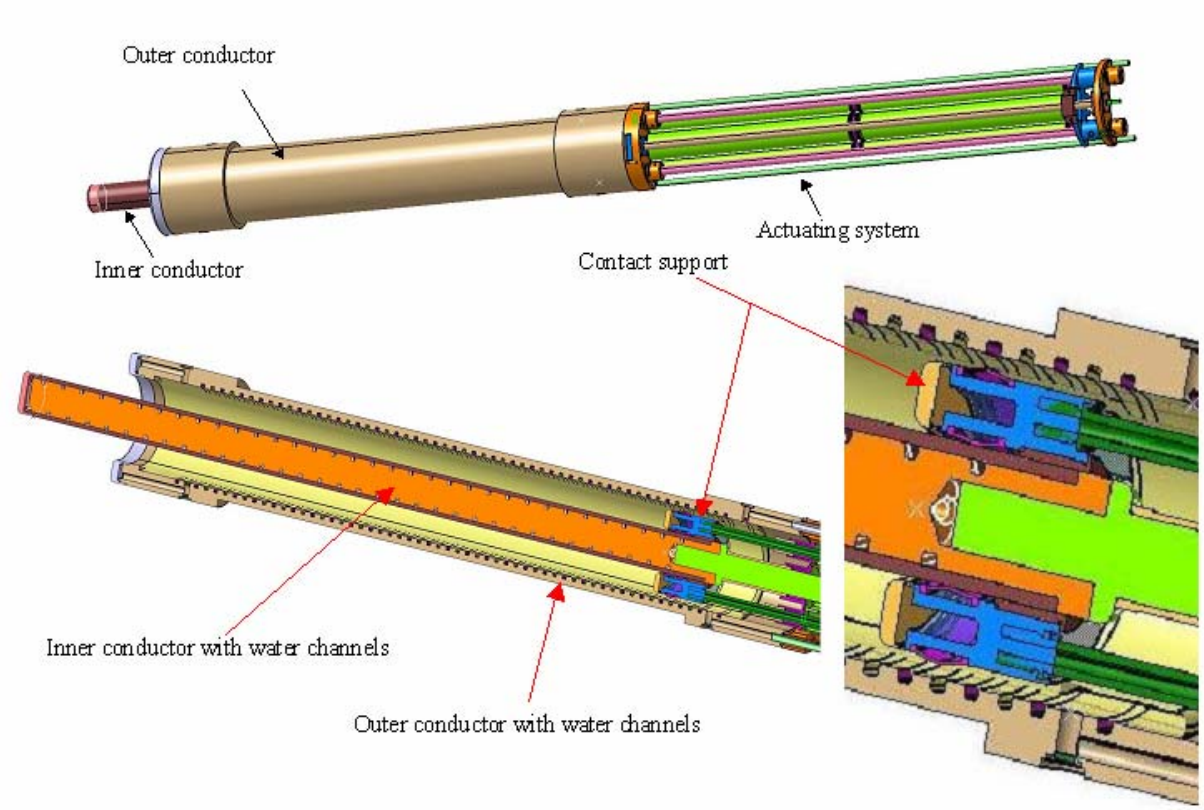


Figure 1: *View of the CVT structure integrated in the ITER IC launcher.*

Our Association pursued its research on SINGAP accelerator concept, which is an attractive alternative to the ITER reference design the so-called MAMuG (Multi-Aperture, Multi-Grid), and three specific tasks have been launched in 2006:

- measure of the beam halo (and method to mitigate it)
- the dark current at the SINGAP test bed
- the electrical discharge and stored energy in a HV breakdown

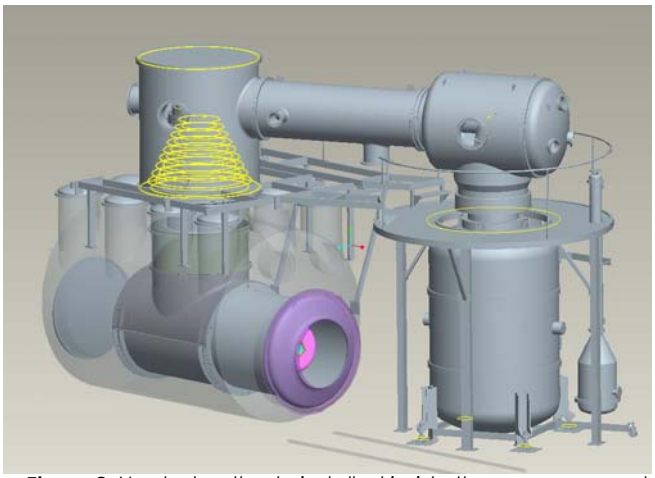


***Figure 2**: Heated cathode installed inside the vacuum vessel*

A new heated cathode structure has been designed and will be installed during 2007. This structure will give a more uniform electric field around the accelerator than at present. A new test set-up for studying the electrical discharge between the acceleration grids in the ITER SINGAP has been designed. This test set-up will demonstrate the voltage withstand before and after a breakdown.

Euratom-CEA has also been strongly involved in diagnostics development, obviously in support to ITER diagnostic systems design, but also in diagnostics projects on JET.
As ITER requires an extensive set of diagnostic systems to provide several key functions such as protection of the device, input to plasma control systems and evaluation of the plasma performance, our Association carried on in 2006 several activities dedicated to this field (among others : bolometer, optical design, magnetic sensors, thermocouples, Infra Red fibres for thermography applications, port integration…). As an example of achievements, this year, the final solution on optical design for ITER has been selected.
The full optical design for the 3 different views into port plug EQ#01 is presented in the sketch [figure 3].

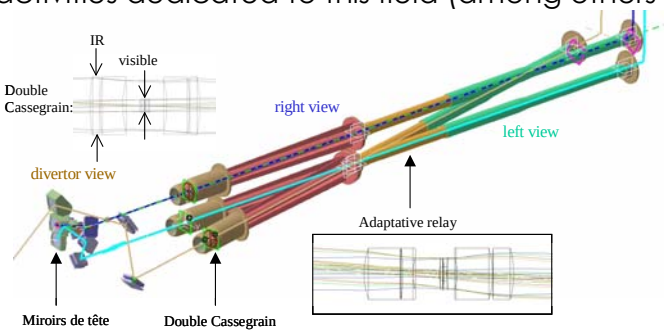


***Figure 3:** full optical design*

Another example of a diagnostic study: thermocouples on first wall components. For ITER, the measurement of surface and body temperatures in the high heat flux components of the divertor (inner and outer targets) and of the blankets are required. Thermocouples will be used to measure the surface temperature within the range 200°C to 1500°C with an accuracy of 20°C. Our Association's work led to the definition of the location of the thermocouples, which has been proposed for ITER divertor [figure 4].

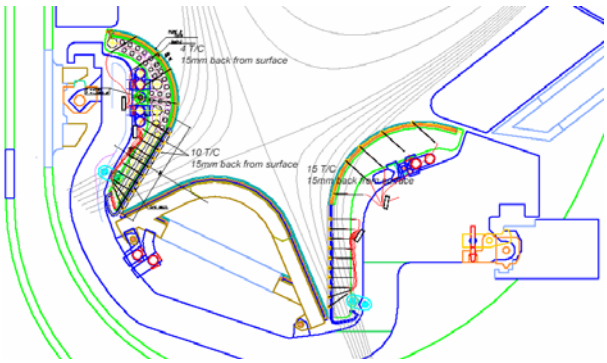


***Figure 4:** Location of the Thermocouples on the divertor.*

Studies on magnetic diagnostics have also been pursued, as EU will supply the magnetic diagnostic for ITER. The objective of the 2006 activity dealt with the analysis of the magnetic diagnostic general design focusing on ex-vessel tangential and

normal coils, external continuous Rogowski loops and fibre optic current sensor which were recently considered for plasma current measurement [figure 5].

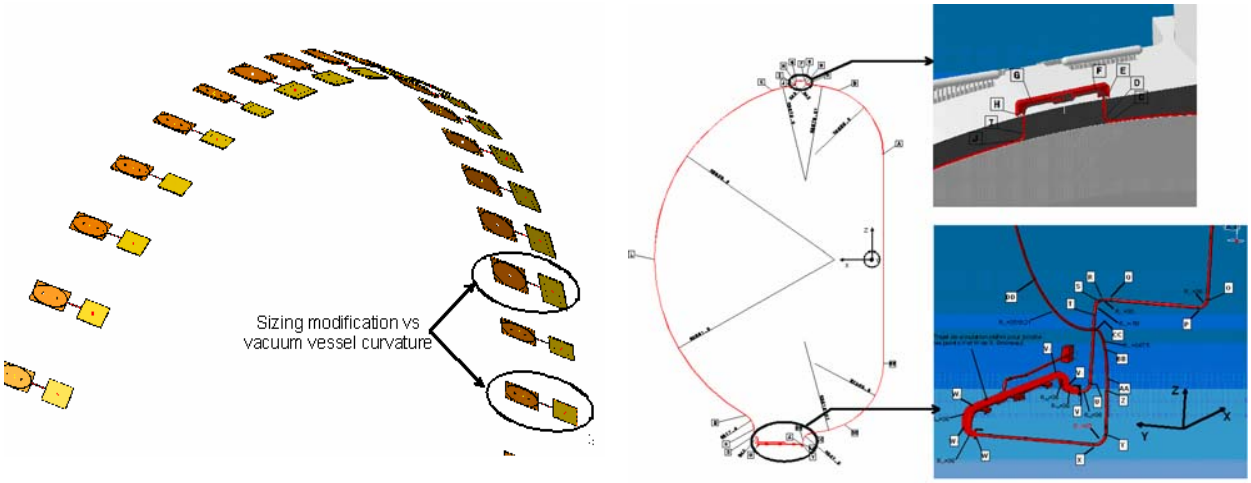
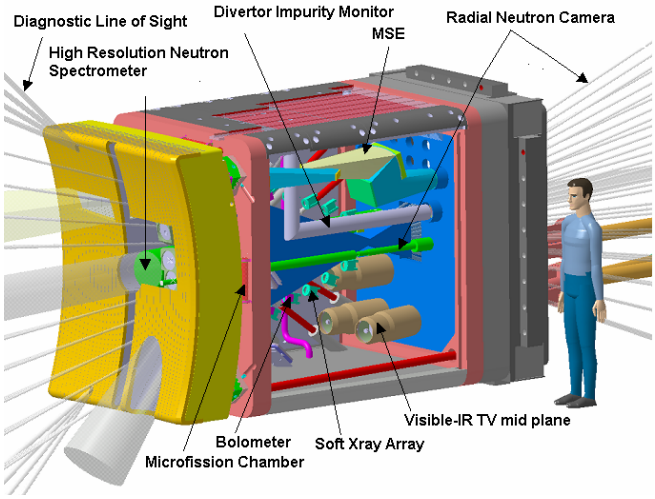


***Figure 5**: Illustration of coils integration on vacuum vessel & Rogowski path in TFC casing*

On ITER diagnostic port integration, in 2005 CEA contributed to general, structural and thermal analysis, those assessment focused mainly on the equatorial port plug EQ#01 [figure 6] chosen as representative. These detailed analyses have highlighted some design issues which were worked out through different solutions.



These engineering activities have been pursued during 2006, they concerned the validation of a new reference design of the port plug structure and specific design options, in parallel wider activities have also been assessed.

***Figure 6:** View of the equatorial port plug Eq#01 and associated diagnostics*

In addition to our support to ITER diagnostic, our Association also developed in the framework of the JET-EP project, a new diagnostic for thermography analysis. This system allows the observation of a large section of the internal components in the vessel such as divertor, main chamber, ICRH antenna, etc, aiming at the measurement of the surface temperature during normal operation and off-normal events such as ELMs and disruptions. This diagnostic is ITER relevant both for the technology used and for the physic outputs. This system will allow to evaluate the power deposition in the main chamber during transient events and could be used in the future, with implementation of a feed-back control, for real-time machine protection. The diagnostic has been installed on JET during the 2005 Shutdown. Due to the delay in the restart and the plasma operation, commissioning of the diagnostic has been delayed to 2006. The scope of the work in 2006 was the calibration and the commissioning of the infrared camera and the endoscope. In spite of minor problems, the diagnostic was operational for the beginning of the C15 campaign on 24th April 2006. During the year 2006, data has been recorded on more

than 2000 shots allowing observation, for the first time in JET, of plasma wall interaction on the main chamber during ELMs and disruptions.

The temperature increase is observable on the plasma facing components during a type I ELM in JET, where we can see that energy deposition occurs not only on the divertor but also on the top and outer limiters.

There are still some problems occurring in high power shots due to hash environment conditions. More investigations are needed to determine the source of the parasitic effects and to strength the diagnostic towards radiation.

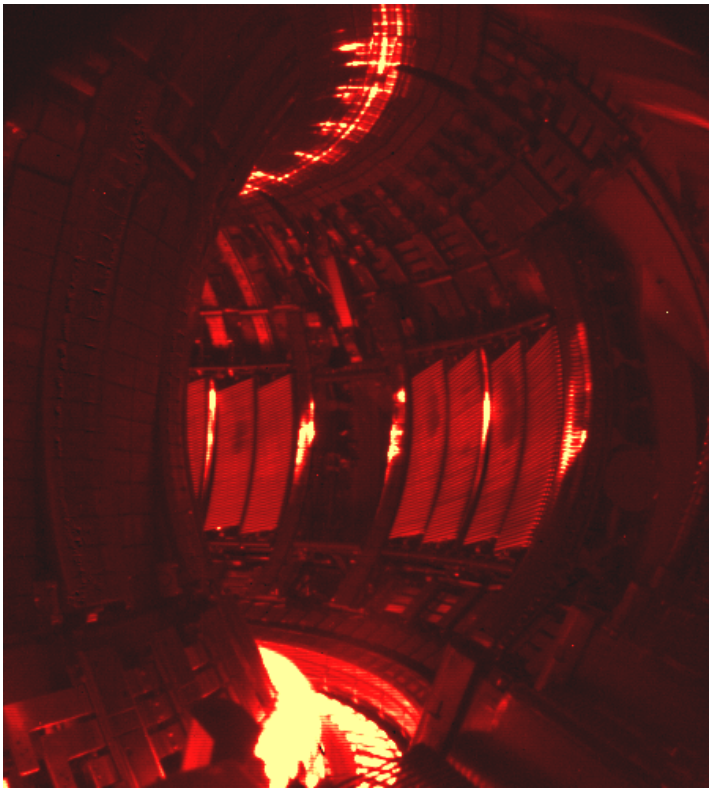


Figure 7: *Infrared image of the in-vessel components showing energy deposited on the divertor, on the top limiter and on the outer limiters during ELM in JET*

Related tasks in the full report:

CEFDA03-1111Bolo, CEFDA03-1111WAVS, CEFDA04-1180, CEFDA05-1329, CEFDA05-1336, CEFDA05-1343MS, CEFDA05-1343PI, CEFDA05-1343TH, CEFDA06-1376, JW6-FT-1.1-D01, JW6-FT-1.1-D02, JW6-FT-4.9, TW5-TPDC-IRRCER-D03, TW5-TPDS-DIADEV, TW5-TPHI-ICRFDEV, TW5-TPO-CODACGW, TW6-THHN-ASD3, TW6-TPDS-DIADEV-D03a

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Vessel/In Vessel activities

The Euratom-CEA Vessel/In Vessel activities have been pursued in 2006, especially concerning studies on **inspection methods of joints of Vacuum Vessel**, on manufacturing of the **ITER Primary First Wall (PFW) panel** (improvement of the knowledge on CuCrZr), and on **providing ITER Materials Properties Handbook** files for Type 316L(N) steel weld metals and joints.

Studies on inspection systems for the narrow gap austenitic welds in the ITER vacuum vessel have been pursued, the objective being to enhance the knowledge on the influence of both the TIG narrow gap weld structure and geometry on the performances of phased array method for one sided welds mock-up that may be used during ITER VV manufacture. Results obtained show that phased array is a valuable technique for detecting and sizing embedded and surface breaking notches in TIG narrow gap welds. In particular, the study carried out presents the analysis of the effect of the TIG joint weld structure related to the UT inspection using phased array techniques. Special effort is paid to analyse the role of both the root joint weld geometry and the attenuation through the weld when flaws are inspected.
Regarding the effects of the geometry, the results show that it is possible to detect apart notch of 3mm high and the signal coming from the bead of weld for a radius of 3mm and 5mm. Nevertheless, it depends on the configuration of the experimental inspection. In general, the inspection of notch 3 mm high is difficult to characterize if the inspection is performed in the direction through the weld. It is due to the geometrical echo generated by the weld root area [Figures 1, 2, 3].

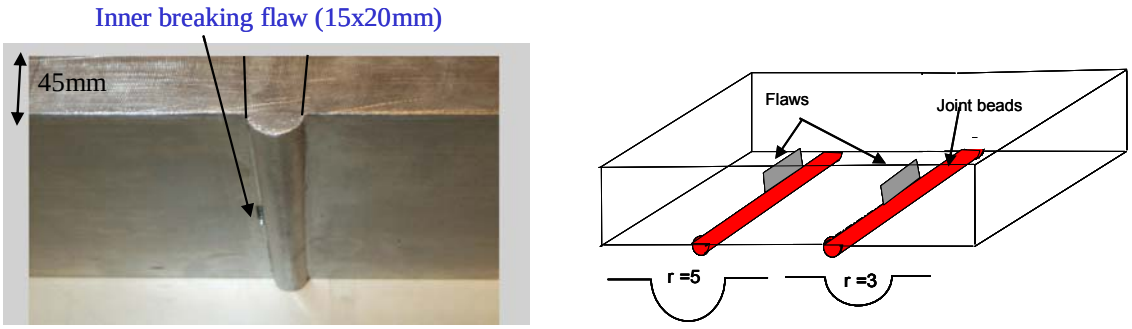

Figure 1: *scheme of the new mock-ups in which was manufactured a bead to simulate the longitudinal deposit of weld metal produced by a fusion-welding process*

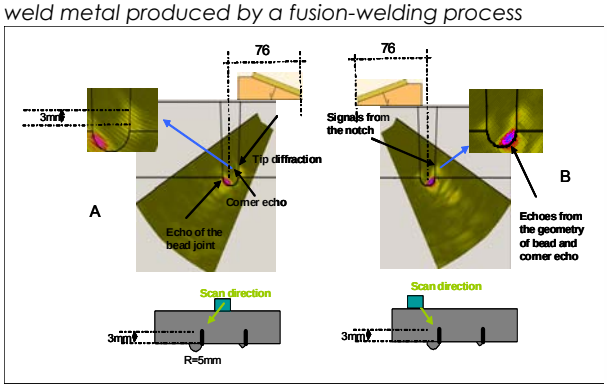

Figure 2: *3 mm inner breaking notch inspection – the radius of the bead is 5 mm.*

A) the notch is inspected before the bead
B) the notch is inspected after the bead

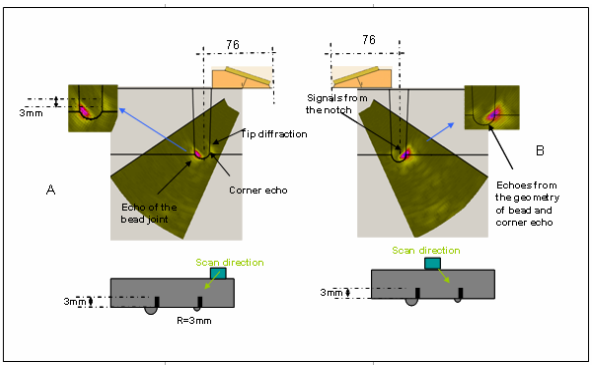

Figure 3: *3 mm inner breaking notch inspection – the radius of the bead is 3 mm.*

A) the notch is inspected before the bead
B) the notch is inspected after the bead

Another aspect of our Association studies on ITER Vacuum Vessel inspection systems has been devoted to a comparative assessment of ultrasonic inspection systems for

narrow gap austenitic welds. This work has been carried out in co-operation with PHOENIX (UK), SINTEZ/ECHO (Ru), the main objective of the global project being to prepare three methods for acceptance by manufacturing code and the French Certification Authorities to be used successfully during the ITER VV fabrication.
The main activities during 2006 were the following:

- manufacturing of four mock-ups [figure 4]
- beginning of the round robin trials of the manufactured mock-ups

The inspection process of the mock-ups will be pursued in 2007.

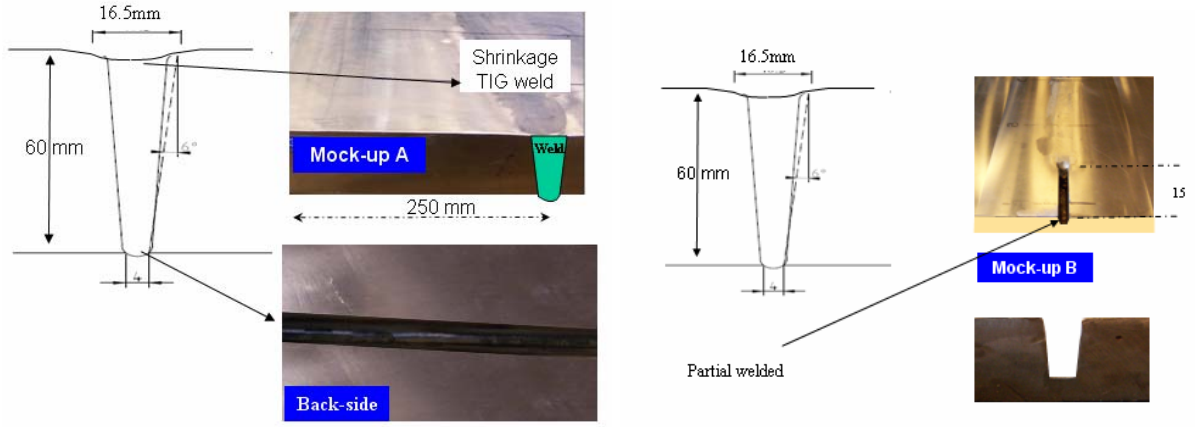

Figure 4: *2 mock-ups (A designed by CEA, B designed by Phoenix)*

On CuCrZr, our Association carried on studies on investigation of the effect of creep-fatigue interaction on the mechanical and lifetime of this material. This year's work showed that there are no explanation to date about why the grain of the heat treated material is so much bigger than expected [figure 5]. It seems difficult to date to find a CuCrZr that can go through a 1040°C heat treatment without high grain growth. But the behavior of this precipitate hardened material has to be determined under creep-fatigue. It has thus been decided to perform the experiments on a different heat treatment avoiding the grain growth, but which gives approximately the same strength of material. The study has been reoriented and experiments will be performed on a material heat treated along a new route.

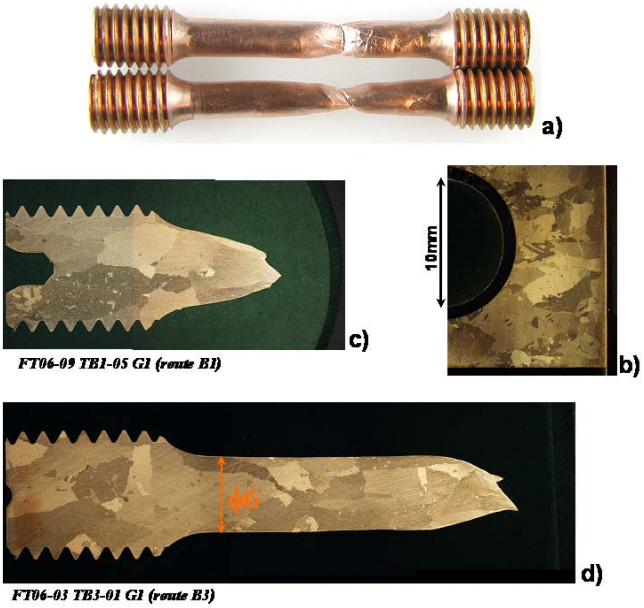

Figure 5: *Photos showing the large grain size and its effect on tensile samples*

Another objective on CuCrZr was to test the lab-scale silver-free brazing alloys developed in 2005, to perform Be/CuCrZr junctions and to characterize these junctions. In 2006, two induction brazing campaigns were performed at FRAMATOME in Le Creusot. Beryllium and CuCrZr were joined with two brazing alloys, STEMET 1108 ATM (78.8Cu-11.1Sn-7.7In-2Ni-0.4P in wt.%) and BRAMn15_3 (76.8Cu-9.9Sn-8.2In-2Ni-3.1Mn in wt.%), at different temperatures and for different soaking times. Coupons of these brazed samples [Figure 6] were characterized by optical and scanning electron microscopy. First conclusion is that, both for STEMET 1108 ATM and

BRAMn15_3, a brazing temperature of 770°C is enough to obtain interfaces without defects. Second conclusion is that joints obtained with STEMET 1108 ATM are better than with BRAMn15_3 whatever the temperature. Following work will be dedicated to mechanical tests on Be/CuCrZr junctions brazed at 770°C using BRAMn15_3. Other temperatures and reduction of plateau duration will also have to be studied.

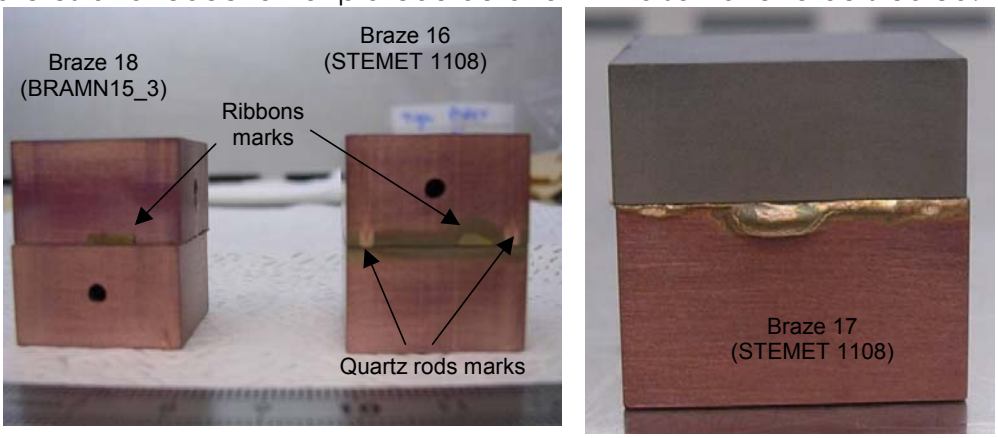

Figure 6: *Views of samples Braze 16, 17 and 18 after brazing*

On the properties of materials used in fusion components which need to be known in detail by designers, by licensing authorities and the materials specialists, our Association contribution to ITER Materials Properties Handbook (document that provides such information in an internationally accepted format) has been pursued, by providing MPH files for Type 316L(N) steel weld metals and joints. In 2005, the work was focused on low temperature (316L) and high temperature (19-12-2, OKR3U) weld metals. In 2006 work has been extended to 16-8-2 weld metal [Figure 7].

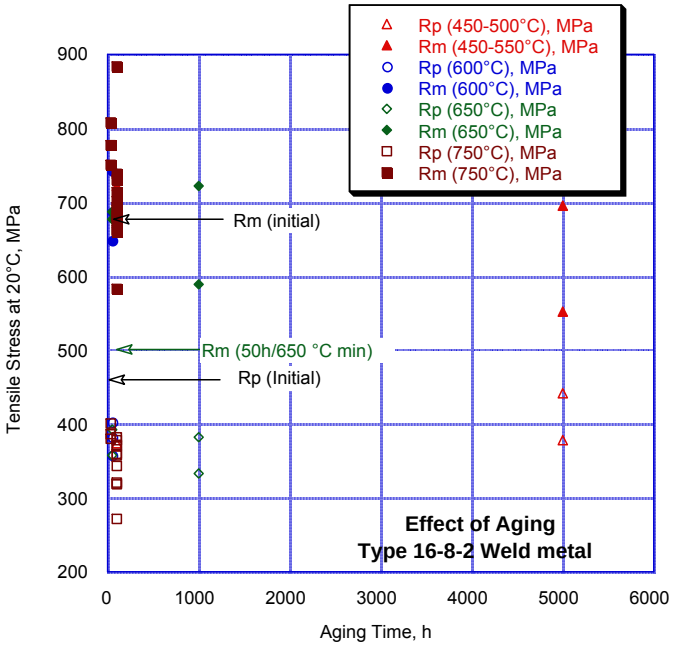

Figure 7: *. Effect of aging on tensile strengths of Type 16-8-2 weld metal. Notice that Rm values remain above lower limit specified for structural*

Related tasks in the full report:
CEFDA04-1202, CEFDA05-1226, TW5-TVM-Braze, TW5-TVM-COMADA, TW5-TVV-MPUT, TW6-TVM-Braze, TW6-TVV-STORVS, UT-VIV/VV-Hybrid-Modeli

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Plasma Facing Components activities

The Euratom-CEA Plasma Facing Components activities have been focused in 2006 on three main topics which are : **Be/CuCrZr** (joining techniques, mock-ups & tests), **CFC (carbon fibre reinforced carbon composite) studies** (joining techniques, erosion), and **test facilities** (test campaigns, studies and upgrades).

The ITER blanket-shield concept is a modular configuration mechanically attached to the vessel. It is constituted with shield modules made with stainless steel and First Wall (FW) panels mechanically attached on the shield blocks. A first wall panel is constituted with a stainless steel support, a copper alloy that acts as a heat sink material and beryllium tiles that act as an armour material against the plasma. 316LN stainless steel, CuCrZr copper alloy and beryllium tiles are assembled together by Hot Isostatic Pressing (HIP). In 2006 the manufacture of 15 First Wall mock-ups for irradiation experiments in order to check the effect of neutron irradiation on the strength and fatigue behaviour of the 316LN/CuCrZr and CuCrZr/Be joints under representative ITER FW panel operation conditions has been launched. Manufacturing of such mock-ups begins always with the fabrication of 316LN/CuCrZr supports. Beryllium tiles are then welded on these supports to finish the mock-ups [Figure 1].

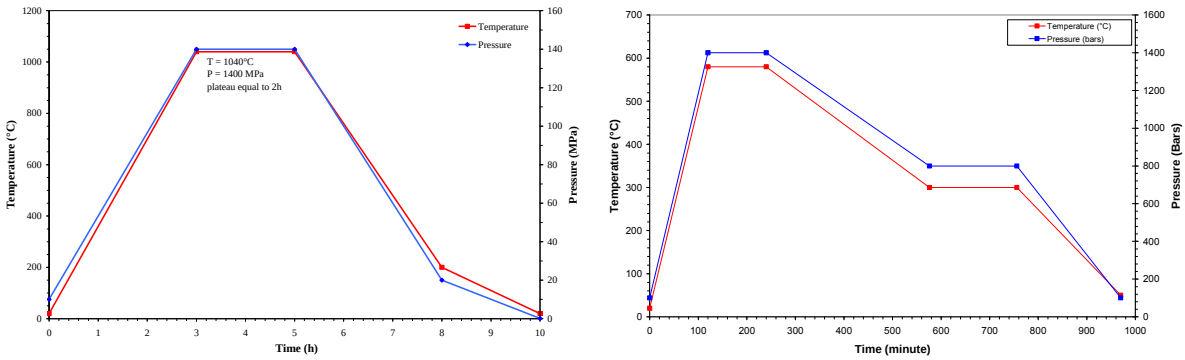

Figure 1: *(left) typical HIP cycle used to weld by diffusion CuCrZr on 316LN materials (right) theoretical HIP cycle to weld the beryllium tiles on the CuCrZr blocks*

Both copper alloy and beryllium tiles are controlled and cleaned before the assembly. The assembly is performed in a special room built by AREVA to manipulate the beryllium due to its properties. The fifteen mock-ups have been divided in two batches. The first batch is constituted of 9 mock-ups which were built in 2006 [Figure 2]. Each mock-up is made with two square beryllium tiles whose surface area is equal to 56mm². The second batch is still under progress.

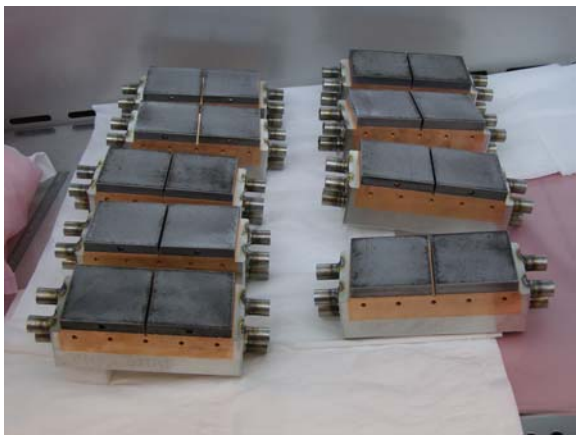

Figure 2: *mock-ups manufactured in 2006*

Concerning CFC, our Association activity also covered a large scale of themes. Joining techniques have been studied (development of a new process for joining CFC to the Cu compliant layer in view of the fabrication of divertor components). Thermal shock test samples CFC/Cu/CuCrZr have been manufactured. The final

joining step (CFC/Cu to be joined to CuCrZr) has been performed by Hot Isostatic Pressing [Figure 3]. The results of the thermal tests are promising. Finally, four mock-ups have been manufactured with the BraSiC® alloy C2 and will be characterised in 2007 by non destructive analysis on the test bench SATIR at CEA-Cadarache and then will be delivered to EFDA for high flux testing.

Figure 3: *CFC brazed onto the pure Cu interlayer, then joined to CuCrZr by HIP*

CFC have also been studied on an erosion point of view for the monoblocks of the ITER divertor vertical target. These monoblocks must sustain high heat fluxes of 10 MW/m² during 1000 s (normal operation) and 20 MW/m² during 10 s (off-normal event). This year work has demonstrated the capability to run erosion calculations with both ANSYS (transient calculation) and CAST3M (series of steady states) [Figure 4]. These calculations have pointed out some erosion instabilities for the studied cases (neighbour monoblock with reduced conductivity or with 90° defects) but that phenomenon can be considered not critical up to now.

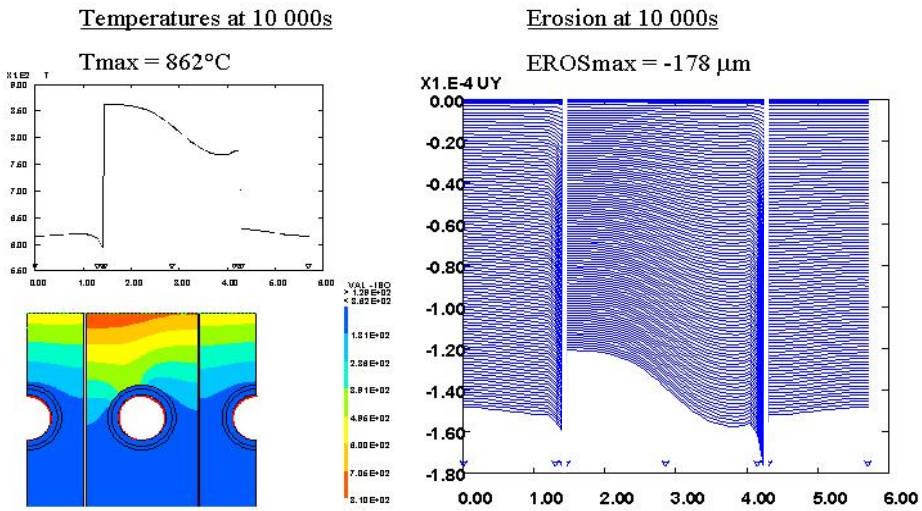

Figure 4: *Temperature map at 10 000s and evolution of the erosion up to 10000s*

Another CFC study has been dedicated to the potential failure of the heat sink to armour joints, which should compromise the performance of ITER: there will be tens of thousands of such joints in the divertor assembly, either carbon fibre composite (CFC) to copper alloy (CuCrZr) joints or tungsten (W) to CuCrZr joints. In preparation for writing the procurement specification for the ITER divertor PFC, this study has been undertaken by our Association with the objective of defining workable acceptance criteria for the PFC armour joints. Investigations on possible detection of the occurrence of the critical heat flux via acoustic emission are also developed [Figure 5].

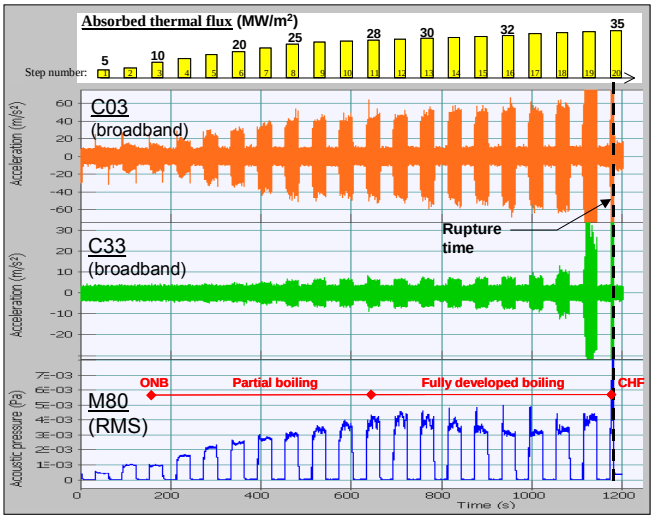

Figure 5: *Acoustic signal during increasing flux steps, up to rupture at 35 MW/m²*

Our Association is also strongly involved in test facilities for PFC, especially on active infrared thermography by internal thermal excitation, which is recognized as a technique available today for improving quality control of many materials and structures involved in heat transfer. An infrared thermography test bed named SATIR (Station Acquisition Traitement InfraRouge) has been developed by CEA in order to evaluate the manufacturing process quality of actively water-cooled high heat flux components (PFC's) before their installation in Tore Supra. SATIR diagnostic has been identified as the basic test to decide upon the final acceptance of the Divertor PFCs. In order to check the ITER components, the possibility to increase the defect detection sensitivity of SATIR has been investigated, and a SATIR upgrade for compatibility with full scale divertor elements with a higher-pressure drop, based on new cold-water injection pump, buffer tank and feeding pumps to keep constant flow rate has been launched.

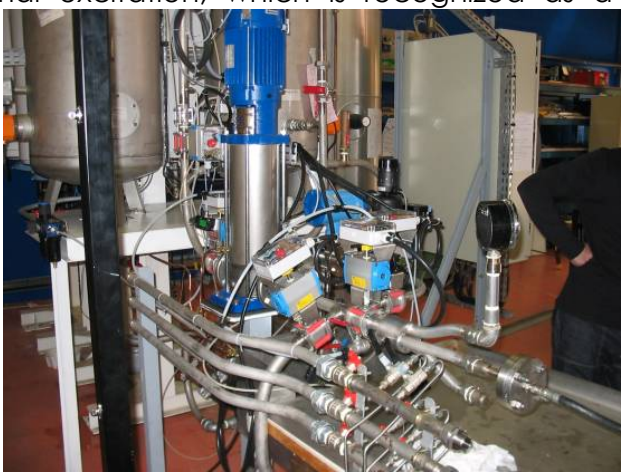

Figure 6: *(Above) View of SATIR test bed (Right) CFC Monoblocks testing*

Related tasks in the full report:
CEFDA02-583, CEFDA04-1218, CEFDA05-1243, CEFDA05-1248, CEFDA05-1257, CEFDA05-1261, CEFDA05-1309, CEFDA06-1372, CEFDA06-1373, CEFDA06-1411, TW5-TVD-CUCFC, UT-VIV/PFC-Damage, UT-VIV/PFC-HIP, UT-VIV/PFC-NanoSic, UT-VIV/PFC-Pyro

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Remote Handling activities

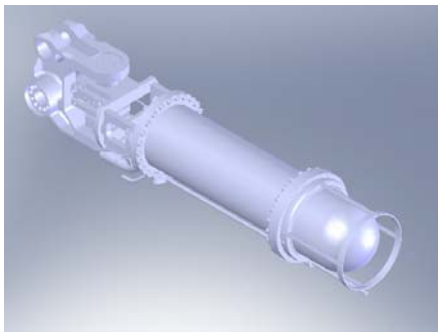
In ITER and future fusion reactors, due to neutron activation, the repair, inspection and/or maintenance of the next fusion device in-vessel components must be carried out by using robotic and Remote Handling (RH) means. Different topics are addressed in this field by the Euratom-CEA Association, such as the development of **specific robots, water hydraulic technologies** or the **radiation tolerance assessment** of required electronic components in a fusion environment.

The so-called Articulated Inspection Arm (AIA) project, led by our Association, takes place in the Remote Handling (RH) activities for the next step fusion reactor ITER. The aim of this R&D program is mainly to demonstrate the feasibility of close inspection of the Divertor Cassettes and the Vacuum Vessel first wall of a Tokamak with a long reach multi link and limited payload carrier.

Figure1: *storage cask*

The work performed includes the design, manufacture and testing of the articulated device demonstrator.
The AIA has to fulfill the following specifications:

- Elevation: +/- 45 ° range,
- Rotation: +/- 90 ° range,
- Robot total length: 7.4 meters,
- Admissible payload: 10 Kg,
- Temperature: 200 °C during baking / 120 °C under working,
- Pressure: $9.7 \cdot 10^{-6}$ Pa / Ultra high vacuum.



The manufacture of the complete AIA robot, including the deployer and the storage cask [Figure 1] is now achieved. The design of the video process is completed, the procurement and its manufacture are foreseen in 2007 [Figure 2].

Figure2: *3D design of the video process*

Hydraulic technology can provide powerful actuators in small volumes. This is an interesting technology followed by our Association in order to build heavy duty manipulators for maintenance operations in space constrained areas. Because of potential leaks, oil hydraulic cannot be used for maintenance operations in ITER. Pure water hydraulics proposes a good alternative to oil and today's developments are

focusing on that direction. To identify areas of improvement, a standard MAESTRO joint was supplied with water. Characterization of the joint was made both from a mechanical and from a control point of view [Figure 3].
The results of 2006 work showed that force and position performances of the joint equipped with a water hydraulics flow servovalve are globally similar or better than the one equipped with an oil servovalve. The cylinder manufactured by CYBERNETIX fits the initial requirements and the conception of an entire water hydraulics MAESTRO arm should be successful. The work will now concentrate on the evolutions of the design to adapt the joint design to water. Endurance tests are starting to identify all areas of improvement in the mechanical design.

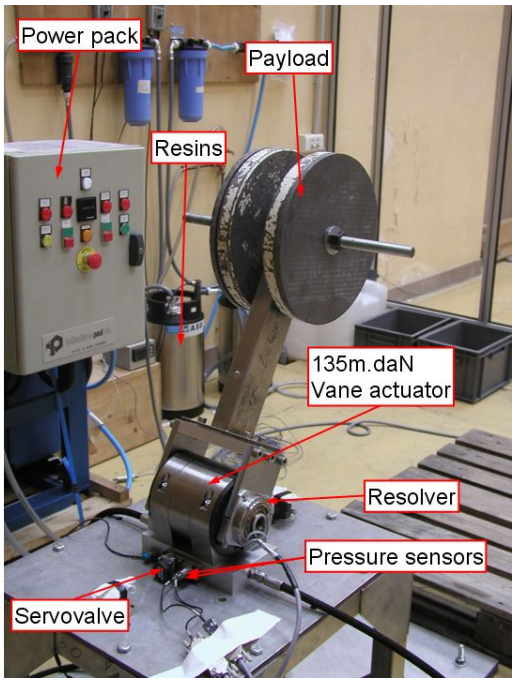


Figure3: *Test rig*

In 2006 our Association also pursued studies on the radiation tolerance assessment of electronic components, which is required before their use in instrumentation for ITER ; the periodic maintenance operations of a future fusion reactor will have to be performed in a severe nuclear environment, exposing operating tools to estimated total doses at the MGy level and temperatures ranging from 50 to 200°C. The main objectives of the study achieved in 2006 were to consolidate the earlier work by designing a new radiation hardened system (under the same environments) and manufacture it. A second aspect of this work was to evaluate the accuracy of the R/D 12-bits conversion subsystem before and during irradiation, for different angular positions. To perform on-line control and appreciate the degradation of the overall "smart" sensor, the most common method consists in relying on a simple comparison between the sensor and a previously calibrated sensor.

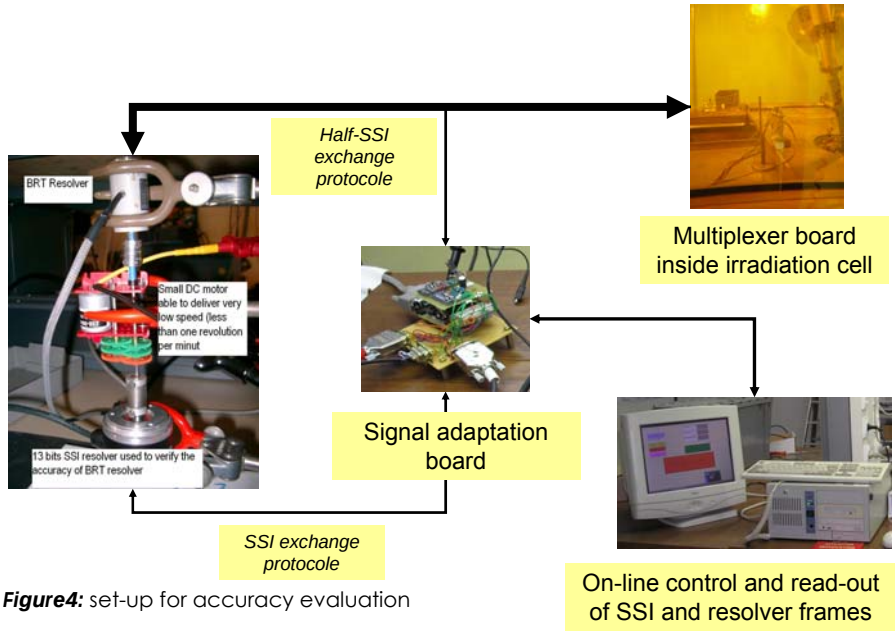


Figure4: set-up for accuracy evaluation

Due to the uniqueness of the module, it was proposed to compare it with a standard industrial angular sensor.
The reference resolver HEIDENHAIN ROC 413 13bits SSI output frame was used.
By mounting the tested resolver to the reference and then connecting this combination to a suitable rotation source, it has been possible to move both devices and compare the data.
A set-up has been developed and automated, using a simple DC motor, and it has been equipped with gear wheels to reduce rotation speed at a very low level. The final axe of the gear box allows the connection of the two resolvers and then a common rotation [Figure 4].

Related tasks in the full report:
CEFDA05-1359, JW6-FT-3.30RHpart, TW5-TVR-AIA, TW5-TVR-Radtol, TW5-TVR-WHMAN, UT-VIV/AM-AIA, UT-VIV/AM-ECIr, UT-VIV/AM-Hydro

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Magnets structures and Integration activities

In 2006, the Euratom-CEA Association has been involved in several activities in the field Magnets Structure and integration: **high performance Nb_3Sn strands, analysis for ITER superconducting coils, cryogenic studies for ITER, superconducting system of DEMO**.
Regarding all the studies performed, what will be described hereafter will not show all the achievements and topics followed by our Association this year, but some significant actions.

In the framework of the EFDA Technology Work program, additional full size conductor samples are being manufactured and tested in order to assess the gain in performance by using advanced Nb_3Sn strand. Two samples (TFAS1 and TFAS2) based on the Toroidal Field Model Coil (TFMC) conductor design were tested in 2005-2006, two other samples (TFPRO1 and TFPRO2) based on the ITER TF conductor design should be tested during 2007, and a fifth sample (NEFSS) using an ITER TF-type conductor fabricated by the Russian Federation should be also tested in 2007. Only four samples based on the TFMC conductor design were initially considered.

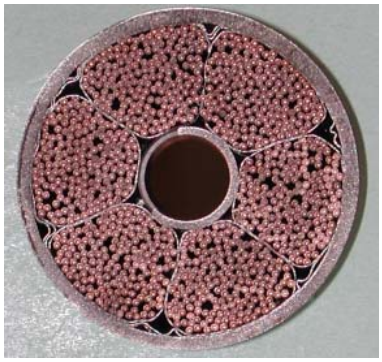

Figure1: *Cross-section of TFAS2 OCSI conductor*

Within these tests, our Association participated in the extensive test of TFAS1, conducted the preparation of the TFAS2 testing program and participated in the test of TFAS2. The tests were performed in the SULTAN facility at CRPP Villigen (Switzerland). The test results have been presented at the 24th SOFT Conference held in Warsaw (Poland) on Sep. 11-15, 2006.
With different behaviors, all these conductors show performance below the ITER TF specification, (the best conductor (OCSI) being about 0.3 K below the 5.7 K specification) and finally not much higher than the original TFMC conductor. These poor results have questioned the present ITER Nb_3Sn conductor design [Figure 2].

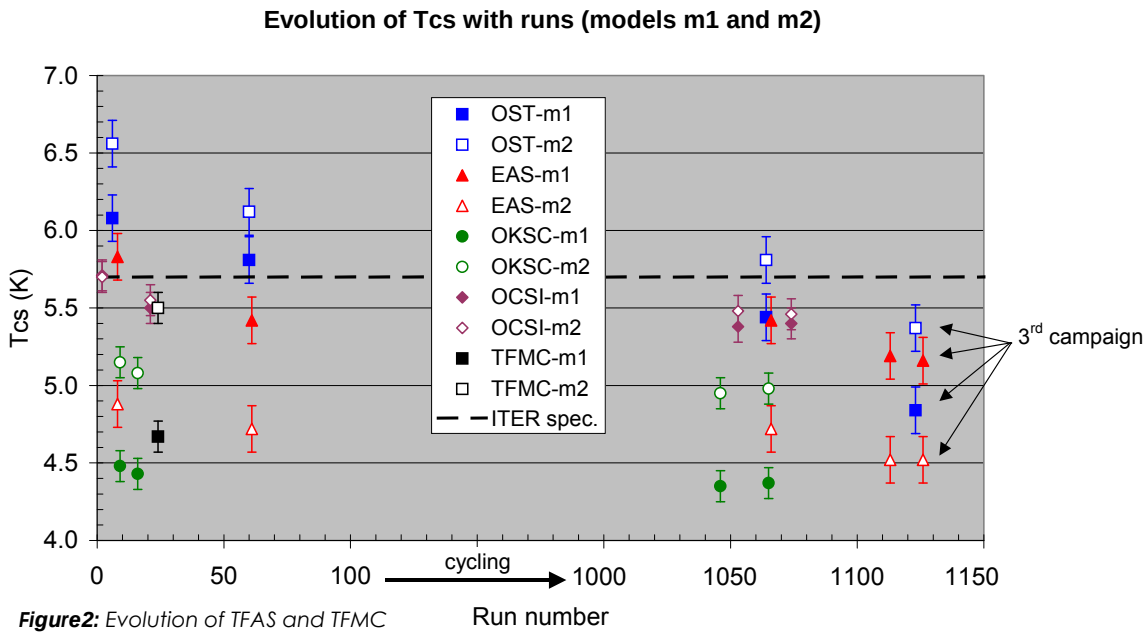

Figure2: *Evolution of TFAS and TFMC conductor Tcs with runs under ITER*

Concerning ITER coils, our Association pursued the ITER PF coils design, coils which consist of a stack of double pancakes of cable-in-conduit conductor with a square section jacket. A design was previously developed and the scope of this year's task was to design and build a mock-up, representative of the main features of the coil tail and to subject it to fatigue cycled test. The manufacturing of the coil tails parts and the adjacent steel plates was completed in 2005. The assembly and impregnation of the mock-up was performed by Alstom under CEA specifications. The fatigue cycled tests, of the mock-up, at LN_2 temperature, took place at ENEA Brasimone in February 2007 [Figures 3 and 4].

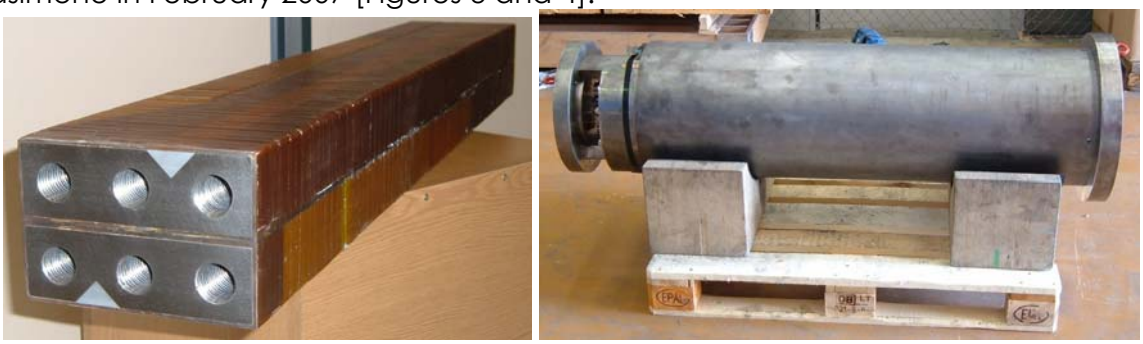

Figure 3
Left: PF tail mock-up after impregnation and end face machining
Right: Mock-up fully assembled with test structure – ready for transport

In preparation of the tests, optical targets and cameras were installed to measure displacements at two symmetric locations.

Temperature sensors were mounted at various levels on the test structure to monitor the cooling down phase. The mock-up and test structure were inserted in thermally insulated steel vessel. Thick glass/epoxy blocks were used for insulating the vessel bottom and the top plate of the mock-up test structure from the machine piston. To maintain the piston always in contact, considering the piston friction of 120 kN, a minimum load was set at 140 kN. Cycling was between 1400 kN and 140 kN with 10 seconds periods. Recording of data was set at 4 Hz. Cool-down took 7 hours to fill-up the vessel with LN_2. Few hundreds cycles were performed at half load to verify the linearity, immediately followed by the 60 000 cycles at full load, which lasted for 6 days. The recorded optical measurements on the mock-up were unchanged over the 60 000 cycles and showed linearity with the cycles at half load. The measured displacement of 2mm peak to peak is consistent with the estimates of 1mm displacement on the mock-up plus 1mm deflection in the test structure. The final analysis and assessment of data is underway but the indications are that the mock-up passed the test successfully.

Figure 4
Test machine and LN tank during cycled tests

Our Association also worked on the project dedicated to provide cryogenic information to the ITER project for further progress in the design of the machine. In 2006 the main achievement was performed on the design and layout of the ACB (Auxiliary Cold Boxes) for magnet structures. The conceptual design of the ACB structures has been successfully performed [Figure 5]. Due to the geometric constraints and the number of devices to install, the result is very compact. As for others ACBs, the compensation system for thermal contraction (300 – 4K) is adapted to avoid the use of bellows, which permits to increase the reliability of the system.

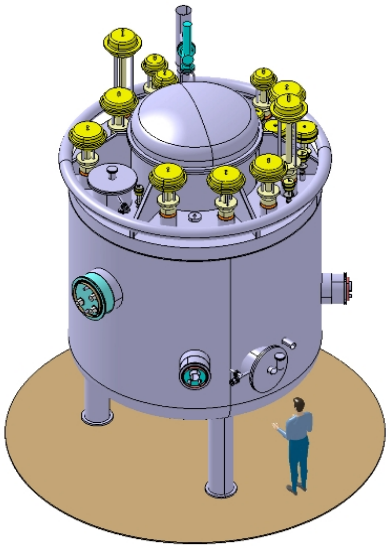

Figure 5
ACB "Structure"

Studies have been performed in 2006 by Euratom-CEA Association on high T° superconductors, which are just coming out of the R&D stage. Long term work will be needed until the availability of such conductors for fusion plants. Among the potential superconducting HTS materials Bi(2,2,1,2) seems the most advanced in term of industrial production as well as suitability for large superconducting magnets. Furthermore it is the only superconducting material allowing the drawing of round wire usable for cabling with conventional cabling machines.
This high T° superconducting material will be studied in order to identify its engineering performance and the parameters limiting its use. Emphasis will be given in electrical properties with the aim to provide data for the HTS magnets program.
The Nexans Company has been working since several years on a 800 kJ SMES (Superconducting Magnetic Energy Storage) operating at 20 K and with a local maximum field of 7 T. This SMES has been successfully tested in 2006. This conductor produced by Nexans in length up to 1000 m and contains 85 superconducting filaments in a silver matrix. It has been tested in our test station between 4.2K and 40K. In a complementary activity initiated by CEA-Saclay a round wire has been developed at Nexans and produced during 2006 [Figue 6].

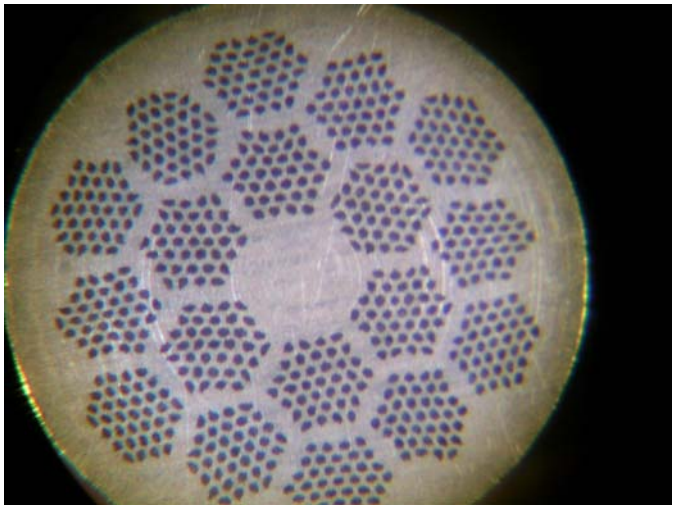

Figure 6 *: Cross section of a Bi(2,2,1,2) round wire developed by Nexans under CEA contract. Diameter 0.8 mm*

The CETACES test station is dedicated to critical current measurement under high field. Samples of superconducting wires can be tested up to 3000 A at 4.2 K in a magnetic field up to 16 T. A thermal regulation allowing measurements between 1.8 and 4.2 K has already been developed. Tested samples are usually Nb_3Sn wires used in VAMAS samples.

***Figure 7**: the upgraded CETACES test facility*

A new sample holder has been developed allowing critical current measurement in the 5 K to 40 K range. This special insert is based on a helium gas thermal regulation. Special attention has been paid to limit the thermal gradient in the sample area.

The upgraded facility [Figure 7] is equipped of a phase separator, used to eliminate the liquid helium and adjust roughly the temperature of the gaseous helium and the test station itself.

To limit the thermal losses in the sample area the 500 A current feed trough is made of superconducting ribbons of the type produced by Nexans.

The new facility has been assembled and tested during the second half of 2006 and worked satisfactorily.

Related tasks in the full report:

CEFDA03-1015, CEFDA04-1127, CEFDA04-1170, CEFDA04-1215, CEFDA04-1219, CEFDA05-1275, CEFDA05-1363, CEFDA05-1370, TW1-TMS-PFCITE, TW5-TMSF-HTSMAG, TW5-TMSF-HTSPER, TW6-TMSC-FSTEST

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Tritium Breeding Blanket activities

The Euratom-CEA Association is the leader for the development of one of the two European Test Blanket Modules (TBM) to be tested in ITER. This TBM is based on the Helium Cooled Lithium Lead (HCLL) breeding blanket concept. Our Association is consequently strongly involved in R&D, **from design to fabrication and safety studies, dedicated to HCLL TBM,** and to **DEMO TBM design studies**.

In 2006, our Association carried out the finalization of the design of a Prototypical Mock-Up (PMU), a relevant medium-scale (¼) prototypical mock-up to be tested out of beam in ENEA loops, in the European Breeding Blanket Testing Facility (EBBTF)/Brasimone for the PbLi loop and HEFUS3 for the He loop. This mock-up should allow prepare the manufacturing of the TBM, the monitoring of the TBM-systems and guarantee that they will not affect the ITER safety. On the PMU, activities have been dedicated to the conceptual study of the external heater, including technical solution and thermo-mechanical dimensioning simulating the plasma heating of the first wall of the TBM (0.5 MW/m²).
The mock-up dimension will be obtained by dividing the TBM height by 4 : 2 cells in the poloidal axis and 3 cells in the toroidal axis, the radial dimension stays the same. The experimental device is based on a cylindrical vacuum vessel in which the TBM mock-up with the heaters will be located.
This vacuum vessel is supported by a support frame [Figure1] which is designed to withstand 2000 kg.
Heaters and electric paths have been designed in detail and the conceptual design of what could be the general test section has been proposed.

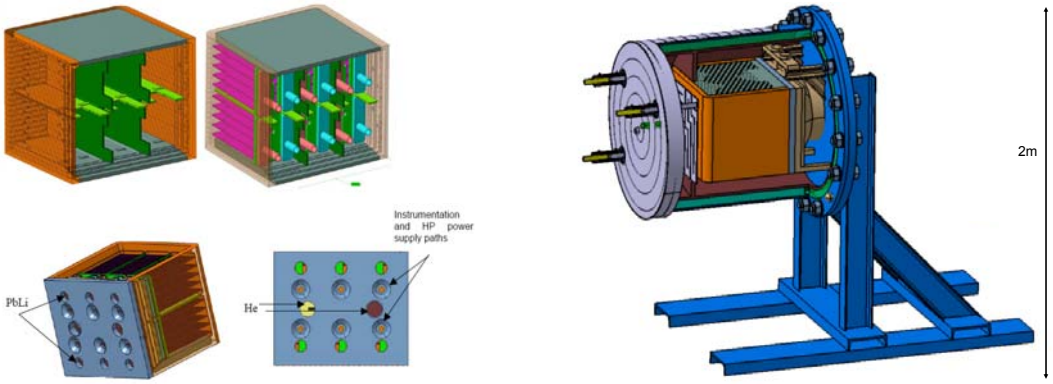

***Figure 1**:*
Left: design of the prototypical mock-up
Right: view of the prototypical mock-up in its vacuum vessel

The definition of the ITER building and especially of the Hot Cell building features is a key issue that has been already extensively studied by the ITER Team. The impact of the TBM testing programme in ITER on the RH and Hot Cell facility has to be evaluated (operation schedule, storage, repairing, etc.). For that the hot cell needs for TBM maintenance operations (replacement, inspection, repairing, storage, PIE) have to be identified and compared to the ITER capabilities. Discrepancies have to be discussed with the ITER team for definition of common solution.

The work performed by our Association is a preliminary version of a HCLL-TBM Integration Description Document [Figure 2]. It aims at defining guidelines and requirements in order to guarantee a high safety level for the HCLL-TBM integration into ITER. It includes operations describing scheduled mounting and dismounting and unscheduled neutralization of the HCLL-TBM. It has sought to identify in the first place the issues that are specifically related to the liquid PbLi use.

Many of the issues related to the HCLL-TBM integration are not determined yet although they have been identified in the document.

Progressively, these issues will be completed and other new issues will certainly be identified.

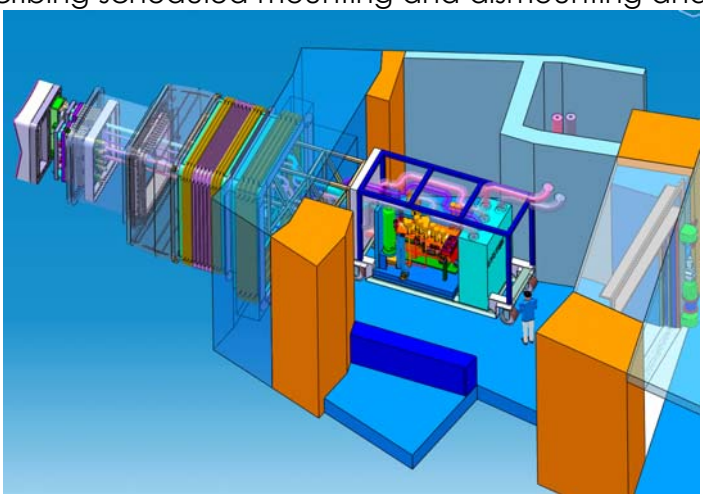

***Figure 2**: Implantation of the HCLL TBM syste in Port Cell #18 of ITER*

Concerning the TBM fabrication, our Association pursued the development of the fabrication process of the cooling plate of the Tritium Blanket Module and use of the (laser + HIP) process. In a first step, that development was engaged on straight channels. The results obtained led us to weld a mock-up which was tested in the DIADEMO loop. The work performed in 2006 consisted in the results of the first mechanical testing of a (laser + HIP) Eurofer junction and in the first welding results obtained on bent channels. Complementary developments to the laser process have been carried out with the arc welding TIG process: although that process brings in theory more welding deformations than the laser process, were assessed the possibilities with the TIG to repair (if necessary) the laser welds, to weld the channels, and to combine the laser to the TIG in order to optimize the welding procedure on bent channels [Figure 3].

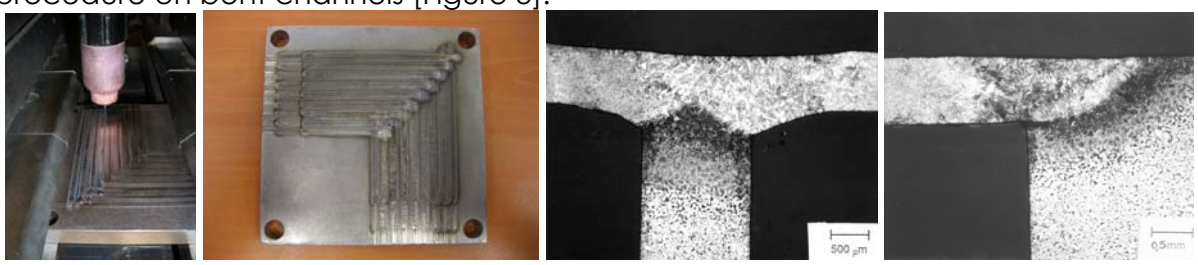

***Figure 3**:*
Left : view of the TIG S235 mock-up after welding
Right: metallographic cut of the TIG welds

With the technological tensile tests carried out at room temperature for that study, it has been checked that the weld seems not to be the weakest point in the welded structure. Nevertheless, it is thought that more tensile tests are necessary to have a good assessment of the tensile mechanical strength of the junction: new geometries to weld and to test are defined. In that study, the TIG welding process is parameterized in order to perform welded channels with strips in wall thickness 0.8 mm. On 120x120x12 mm^3 samples with 8 bent 4x4.5 mm^2 channels on each sample, a welding procedure has been developed with the TIG welding process and the laser YAG welding process.

On these samples, the displacements induced by welding are assessed. The TIG distortions are twice the laser distortions (with the S235 steel). The same

characterizations must be performed with the right EUROFER steel. Nevertheless, as it was seen for the "DIADEMO" mock-up, the welding distortions seem too high for the (fusion welding+HIP) process developed to build the cooling plate, whatever the welding process used (laser or TIG).

Another field studied by our Association is dedicated to tritium breeding blanket for DEMO and future fusion power plants, and among other studies, we present hereafter 2006's work on sensitivity of Pb-Li velocity profile on the estimation of the Tritium mass flow rate towards the He-coolant under DEMO and ITER TBM conditions for a Breeder Unit [Figure 4]. These activities have been done on the modeling of the permeation of T towards the He circuits for the DEMO 2003 mid-equatorial inboard and outboard HCLL blanket module.

It is shown, for the mid equatorial HCLL modules with the retained assumptions, that the permeation through the helium circuit, the mean outlet tritium concentration in Pb-15.7Li and the ratio between the permeation through the helium circuit and the production rate are insensitive to the magnetic field and the buoyancy effects. However, due to the flow, a concentration boundary layer exists and is to be regarded as an equivalent permeation reduction factor (PRF) of 30. Considering the difficulty encountered in the past years to obtain stable and reproducible high PRF with Al-based coating on Eurofer, an equivalent PRF of 30 is of great importance for the T inventory in the HCLL blanket.

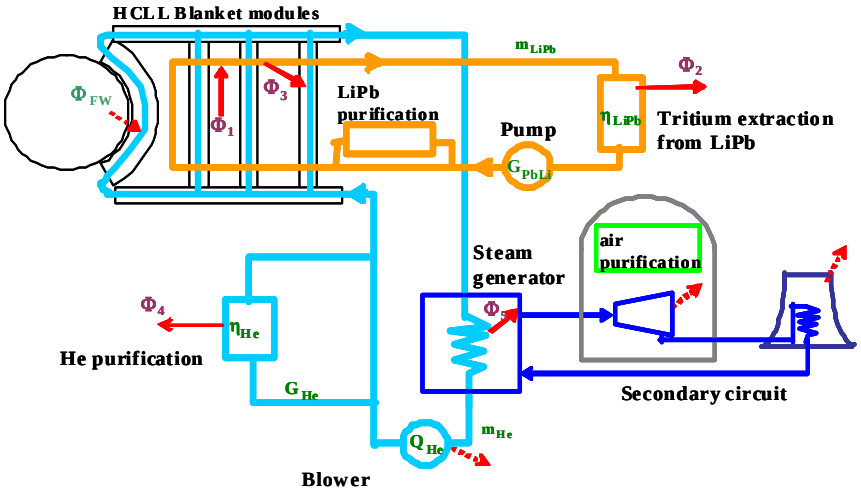

***Figure 4**:*
Main tritium flows in a fusion reactor

Related tasks in the full report: TW2-TTBC-002-D02, TW2-TTBC-002-D03, TW2-TTBC-003-D05, TW4-TTBC-001-D01, TW5-TTBC-001-D01, TW5-TTBC-001-D02, TW5-TTBC-001-D03, TW5-TTBC-001-D05, TW5-TTBC-001-D06, TW5-TTBC-001-D07, TW5-TTBC-001-D08, TW5-TTBC-001-D09, TW5-TTBC-001-D10, TW5-TTBC-001-D11, TW5-TTBC-002-D01, TW5-TTBC-002-D02, TW5-TTBC-002-D03, TW5-TTBC-005-D05, TW6-TTBC-001-D03, TW6-TTBC-002-D01, UT-TBM/BB-He

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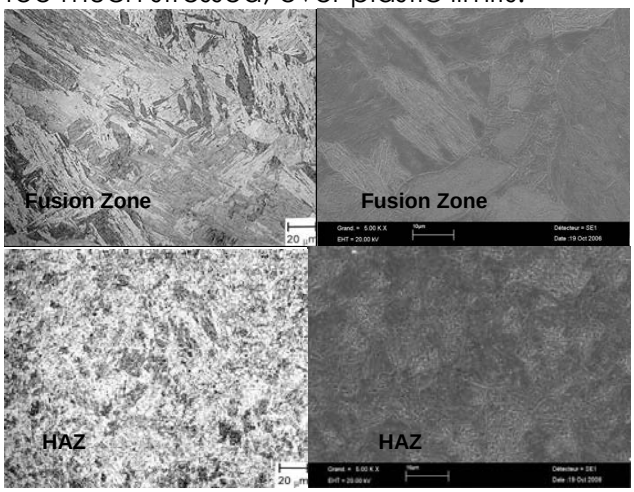
Structural Material activities

In 2006, our Association activities in this field have mainly been dedicated to **qualification of fabrication processes for TBM**, and studies on **Eurofer, modelling of irradiation effects** and **advanced materials**.

On qualification of fabrication processes, one of our Activities have for example been dedicated to Eurofer weldability, which must be established for data base assessment on such low activation martensitic steel. Electron Beam, Hybrid (Laser combined with MIG/MAG), laser and TIG welding processes have been carried out on Eurofer samples from 8 mm to 40 mm thickness.
This year's work enabled to conclude that Electron Beam process is not well adapted for this application [Figure 1]. Processing troubles have been demonstrated: too narrow Fusion Zone and Heat Affected Zone (HAZ) width, δ ferrite content, and too high hardness level gradient. Interesting results have been got using the MIG/laser Hybrid technique developed by CEA. High hardness level in Fusion Zone and carbide formation in HAZ let think to apply PWHT process or even pre- and post-heating process. Horizontal / Vertical Stiffening Grid mock-ups have been performed. Laser process is reference process, and TIG is second hand process. Distortion level induced by laser process is acceptable for manufacturing stage. HAZ and Fusion Zones are larger in TIG compared to laser ones.
For TBM design aspects, conclusions show clearly the need to modify the design of the Horizontal Stiffening Grid: the distance between the end face of Horizontal Stiffening Grid and the first cooling channel face must be at minimum for Laser process 5 mm and TIG process 7 mm. If not, first cooling channel will be distorted and too much stressed, over plastic limits.



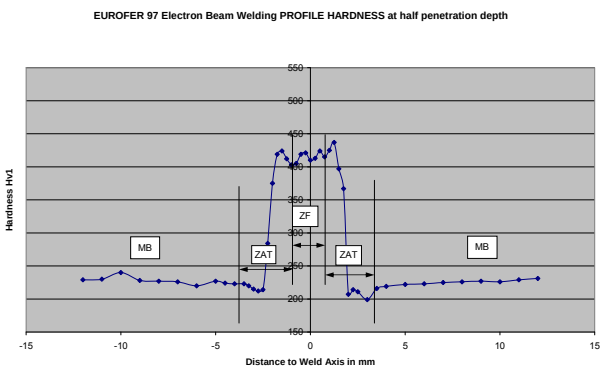


***Figure 1**:*
Left: micrograph cross section of EB weld
Above: HV1 profile hardness of EB weld

Another example of our Association's work is the developments aiming to process high quality welds according to Helium-Cooled Lithium-Lead (HCLL) DEMO Blanket Module design. Several parts of the test blanket module are planned to be joined by welding process. The main objective of this study was to produce simplified mock-up of the TBM's stiffening grid assembly in order to characterize the residual stress and the distortion induced by the welding process. The experimental trials have been compared to numerical simulations carried out with the finite element code SYSWELD®. The preliminary calculation of residual stress on a T shape mock-up has been performed.

Two modes have been considered: Mode I consists in, successive and opposite direction of the passes; Mode II consists in, simultaneous and same direction of the passes [Figure 2].

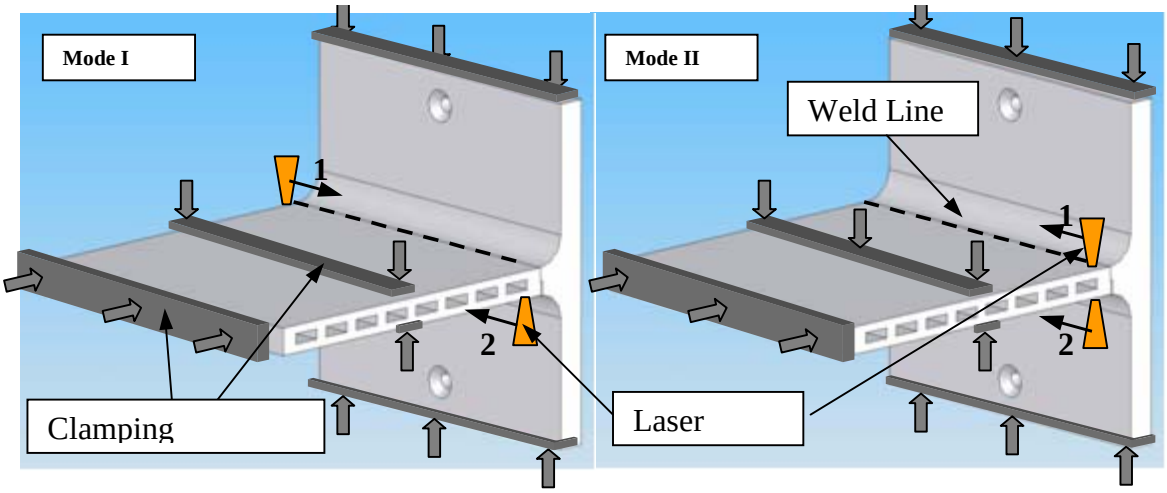


***Figure 2**: Assembly Mode I and II and clamping conditions*

Simulations are performed with the finite element code SYSWELD®, the thermo-metallurgical model for Eurofer is used [Figure 3].

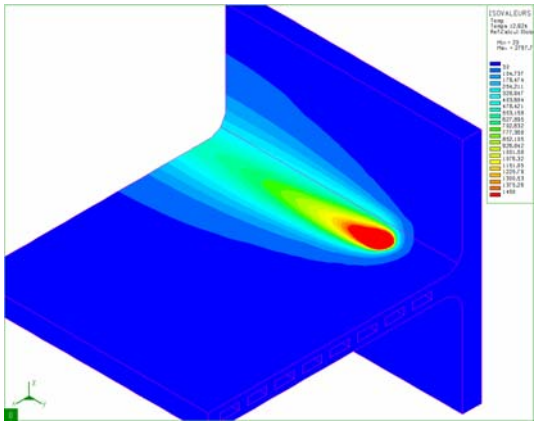


***Figure 3**: Temperature field during welding of the first pass*

Concerning modelling of materials, our Association performed studies on irradiations effects, using ab-initio calculations, for example on Ferritic steels, which play a central technological role as structural materials for fission and future fusion nuclear reactors. One of the principal sources of the mechanical property degradation of this steel is the intergranular embrittlement, which can be caused by segregation of solutes at grain boundaries (GBs). In particular, fusion reactors produce high energy neutrons (~ 14 MeV) that interact with the structural material, which cause nuclear transmutations and create significant amount of helium. The formation of helium bubbles at GBs is known to lead to high temperature embrittlement. A quantitative description of the interaction of He atoms with grain boundaries is therefore essential.

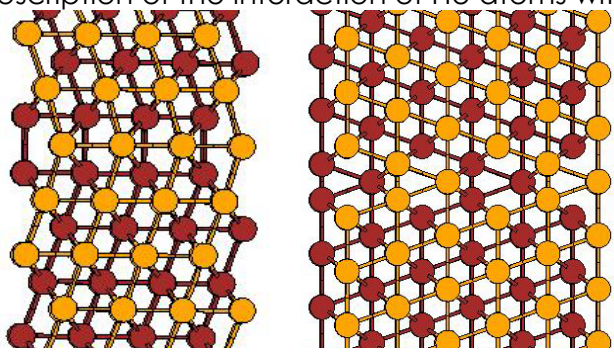


***Figure 4:** Schematic representations of the atomic structures of Σ3 {112} (110) (left), and Σ3 {111} (110) (right) symmetric tilt grain boundaries in α-iron. Spheres of different colors represent atoms in alternating planes perpendicular to the (110) tilt axis*

In 2006, ab-initio and empirical potential (E.P.) simulations have been performed to study the properties of representative symmetric tilt grain boundaries in α-iron [Figure 4]. The following conclusions can be drawn from the present ab-initio calculations on the energetic, structural and magnetic properties of symmetric tilt grain boundaries in α-Fe and on the effect of He on these GBs:

- Good agreements are found between ab-initio and two existing E.P. results on the formation energies of Σ3 GBs, but the empirical potentials underestimate the free surface energies and consequently the GB cohesive energies.
- An enhancement of local magnetic moments in Fe atoms at the interfaces is noted consistent with the increase of their interatomic distances.
- The interactions between He and these symmetric tilt GBs are attractive. Qualitative and quantitative discrepancies are found between ab-initio and empirical potential results on He-GB binding energies.
- He atoms prefer to be in GBs with large excess volume in agreement with existing E.P. studies.
- He atoms tend to occupy interstitial rather than substitutional sites close to the GB planes, which suggest that Helium may migrate easily via interstitial mechanisms along the interfaces.

Concerning advanced materials, our Association pursued its effort on modelling, as shown in the following study on SiC/SiC : the non-linear elastic mechanical behaviour of SiC_f/SiC is governed by damage phenomena that take place at the scale of the components of the composite (fibre, matrix, interface).

As a consequence, the most promising way to obtain a reliable behavioural model that can evolve with our knowledge of the material is to use multi-scale modelling [Figure 4].

In 2006, the implementation of General Transformation Field Analysis (GFTA) change scale method has started. Procedures were developed in order to calculate the concentration and influence tensors for the GTFA method in the case of the *Mori-Tanaka* estimate, suitable for the micro-to-meso scale change, and *Ross et al.* estimate, suitable for the meso-to-macro scale change. It was verified that the calculations of the homogenised stiffness and thermal expansion for properties varying with temperature and damage were correctly performed using the GTFA method. These procedures were developed for the case of a two sub-phase material. It should be

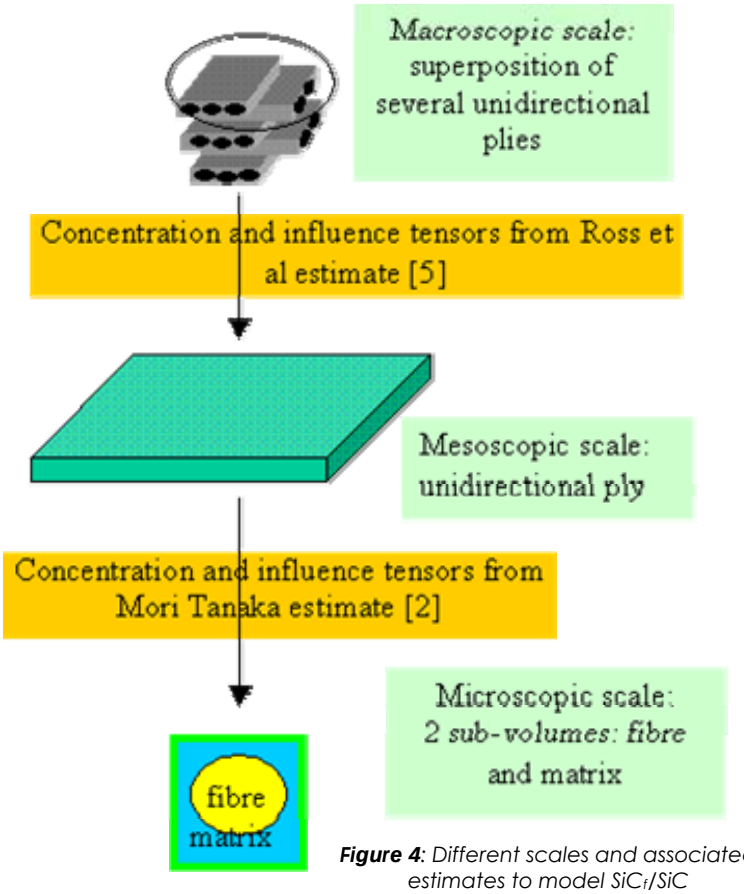


***Figure 4**: Different scales and associated estimates to model SiC_f/SiC*

extended to more sub-phases in the future. The concentration and influence tensors will be introduced as material properties in the future. This is one of the necessary implementations in the scope to perform finite elements GFTA calculations for SiC_f/SiC modelling.

Related tasks in the full report: TW3-TTMA-001-D04, TW3-TTMA-002-D04, TW5-TTMS-004-D02, TW5-TTMS-004-D04, TW5-TTMS-004-D06, TW5-TTMS-004-D07, TW5-TTMS-006-D01, TW5-TTMS-007-D04, TW6-TTMA-001-D02, TW6-TTMS-005-D01, TW6-TTMS-007-D02, TW6-TTMS-007-D08, TW6-TTMS-007-D11, UT-TBM/MAT-LAM/Opti, UT-TBM/MAT-Micro

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Safety and Environment activities

In 2006, our Association pursued its main activities in this field, in which **tritium management**, **dust removal, measurement** and **dust explosion** models and study of **activated corrosion products** have taken an important part.

ITER safety studies demonstrated that a few tens of plasma shots are required to reach the maximum acceptable inventory of tritium in the vacuum chamber, thus resulting in tritium trapping in Plasma Facing Components (PFC). Nevertheless, the studies on tritium trapping in PFC revealed that a large amount of tritium is retained in the deposited layers. Thus, vacuum chamber surface detritiation and tritium removal are regarded of essential importance for the future ITER.

For 2006, the main goal of the task program was to perform detritiation tests with the developed laser device in the Beryllium Handling Facility / JET (BeHF) environment and to assess the deposited layer detritiation performances with the available JET samples. Another main goal was to study the means of assessment on the feasibility to use the laser tool head with remote handling means.

The results have been very encouraging, as a compact optimized laser cleaning system (1.06 µm, 20 W, 20 kHz, 100 ns) has been developed and tested in BeHF (JET) environment. Laser detritiation tests on BeHF were successfully performed with the JET tiles 3 and 4. A good cleaning performance of the laser system was determined. Three consecutive scans of the laser beam were sufficient to remove the film from the JET tile 4 (~ 100-300 µm film thickness) [Figure 1]. High detritiation performance of the laser system was confirmed by the optical and NRA analyses of the cleaned zones. The detritiation tests were successfully performed with the JET tile 3 of a low (»10 µm) film thickness, but with a very hard deposited film with important Be concentration in it. Only 20 minutes were required for the complete cleaning of the tile. Most of the film removal may be collected as micro-particles (still with tritium) by the slit box filter. 0.3 m²/h removal rate for a 10 µm hard deposit was obtained.

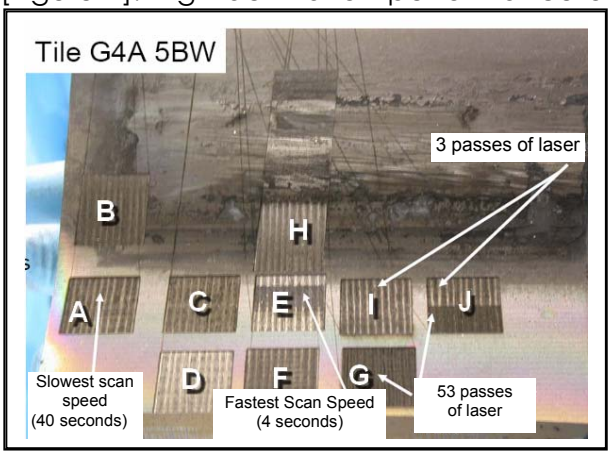
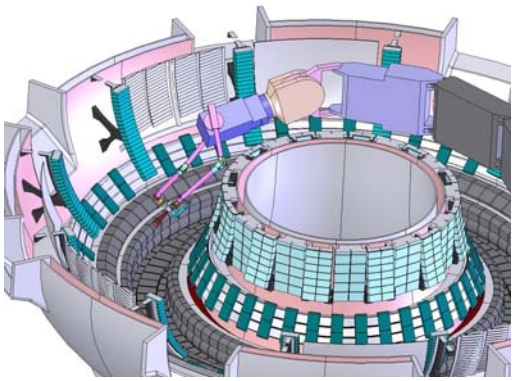


Figure 1: *the deposit on zone E with regime 4 was not completely removed after 6 laser passes*



On the RH part of this project, the design of a complete Remote Handling system has been investigated. A two arm tool seems to be a good solution but a lot of work has yet to be done to implement the electric motor solution in the whole system [Figure 2].

Figure 2: *Overview of the complete RH system (Integration of the laser tool on JET)*

Our Association has also been involved in studies dedicated to dust, for both measurement, removal and dust explosion risk models. The key point on that topic is that during ITER lifetime, dusts will be produced due to the interaction of all types of plasmas (including conditioning procedures) with the Plasma Facing Components (PFCs). These dusts will be activated, tritiated and potentially chemically toxic (presence of beryllium). ITER fixed a set of safety limits to manage the potential hazards which might be caused by these dusts. The aim of one of our Association tasks was to assess techniques that could be used during ITER lifetime to control dust inventories within the vacuum vessel and to be able to recover the dusts when the safety limits are reached. This year, dust production as well as assessment of diagnostics for dust monitoring (Optical systems, Sampling systems, Gravimetric systems) and removals techniques (Vibrating conveyor, Electrostatic conveyor, Dishwasher,…) [Figure 3] have been studied.

This work will continue in order to assess more in details techniques that could be complementary to the already studied ones.

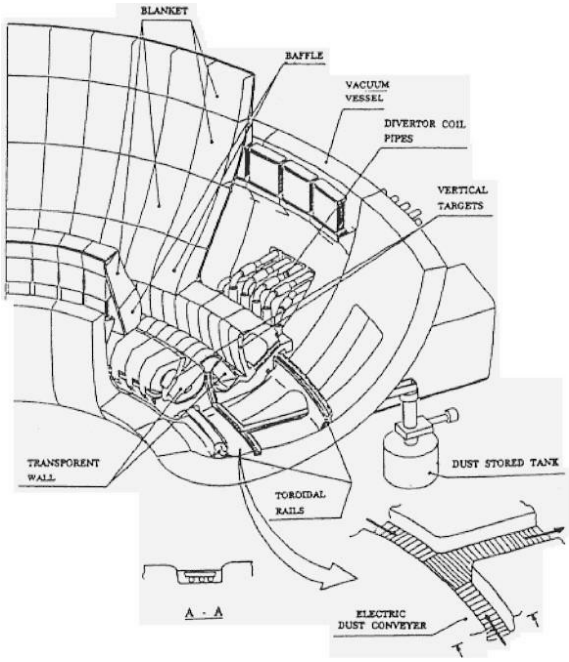


Figure 3: *Concept of continual dust removal system*

Related tasks in the full report:
CEFDA05-1368, JW5-FT-5.25, JW6-FT-2.28, JW6-FT-3.30, JW6-FT-4.8-D1, JW6-FT-4.8-D2, TW5-TSS-SEA3.5, TW5-TSS-SEA5.6, TW6-TSL-004, TW6-TSS-SEA5.1, UT-S&E-LASER/DEC, UT-S&E-LiPbwater, UT-S&E-Tritium-Impact

Related Laboratories: DEN/DPC, DEN/DER

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System Study activities

In the field of System Studies, our Association is involved in **design studies for DEMO and future fusion power plants**.

Within the framework of the European Power Plant Conceptual Study (PPCS), the relatively near-term reactor model AB based on the use of Helium-Cooled Lithium-Lead blanket (HCLL) has been developed and assessed. The HCLL blanket is based on the use of EUROFER as structural material, of Pb-Li (Li at 90% in ^{6}Li) as breeder, neutron multiplier and tritium carrier, and of helium as coolant with inlet/outlet temperature of 300/500°C and 8 MPa pressure.

The year 2006 was devoted to the conceptual design of the model AB HCLL blanket based on the Multi-Module Segment (MMS) maintenance scheme. The basic principle [figure 1] is to have relatively small modules, welded on a strong poloidal back structure, in order to form a blanket segment which can be removed from the top in a similar manner as a banana-shaped segment.

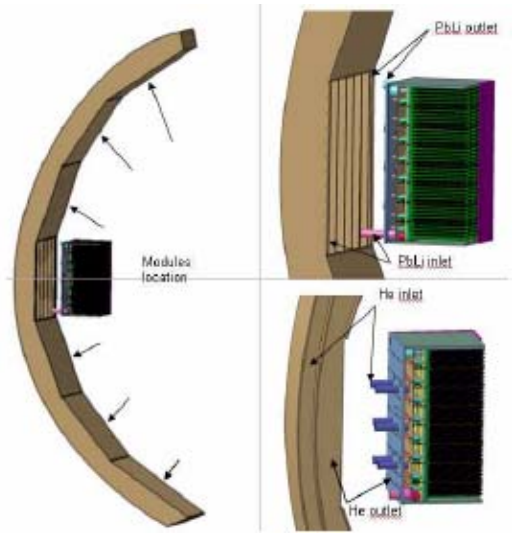

Figure 1: *Illustration of "segment module" shape*

This preliminary design study highlights several key points, on the MMS concept proposed, to be checked in more details:

- Attachment of the segments on the hot ring shield,
- Modules design on the extremity of the segment,
- Maintenance of segments,
- Thermo-mechanical assessment of the segment for its optimization, taking into account EM loads.

On the path to the development of the first commercial fusion reactor, DEMO will be the next step after ITER, with the aim of testing the main technological options at a somewhat reduced electrical power with respect to the commercial reactor (1 GW vs 1.5 GW). The DEMO reactor should work in steady-state, therefore, one of the main physics challenges will be the establishment and the control of a non-inductively driven current density profile. Our Association pursued the analysis of this problem by means of the integrated modeling code called CRONOS [Figure 2].

The results obtained in these first series of CRONOS simulations are encouraging since they show that a regime close to the hybrid one can be possibly attained in DEMO with a reasonable amount of CD power. Nevertheless, the main aim of obtaining a fully non-inductive regime (at a value of Q ~ 20) has not been attained. Since the total current drive power

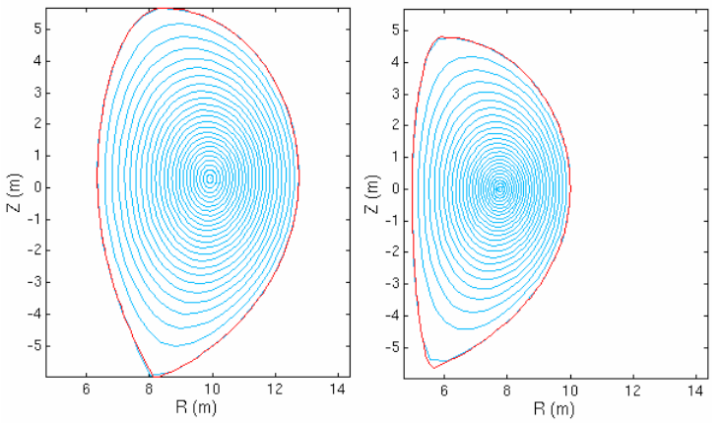

Figure 2: *Computed equilibria for the inductive (left) and the advanced (right) DEMO, respectively*

cannot be further increased without lowering the Q, the basic ingredient that has to be exploited is the development of a transport barrier. If a pronounced reversal of the q profile can be obtained on a large part of the plasma cross section by a suitable combination of off-axis current drive sources, the bootstrap current will increase at the barrier location, enhancing the effect in a positive non-linear loop. The next problem will of course be the control of such a process. Finally, the use of full NBCD and LHCD computations by advanced modules (Monte Carlo and ray-tracing/Fokker-Planck respectively) will be necessary in order to validate the scenario.

In the framework of DEMO conceptual studies, the TF magnet system has to be integrated at an early stage in the overall design development. As a matter of fact, located at the very heart of the tokamak, the TF system interact with many aspects of the machine. It is therefore important to rapidly identify and simply parameterize the mains aspects of this interaction in order to optimally adapt the machine design.
In that aim, the ESCORT code [Figure 3] appears as a very useful tool able to deliver preliminary designs of the TF system, during the DEMO conceptual studies.
It is now foreseen to deliver the main characteristics of three DEMO TF versions at three different temperatures of operation: 5 K, 20 K and 50 K.
In relation with the CEA team of Cadarache, a criterion is being established to ensure that there is sufficient place in DEMO preliminary design to extract the blanket modules through the upper port between two TF outer legs.

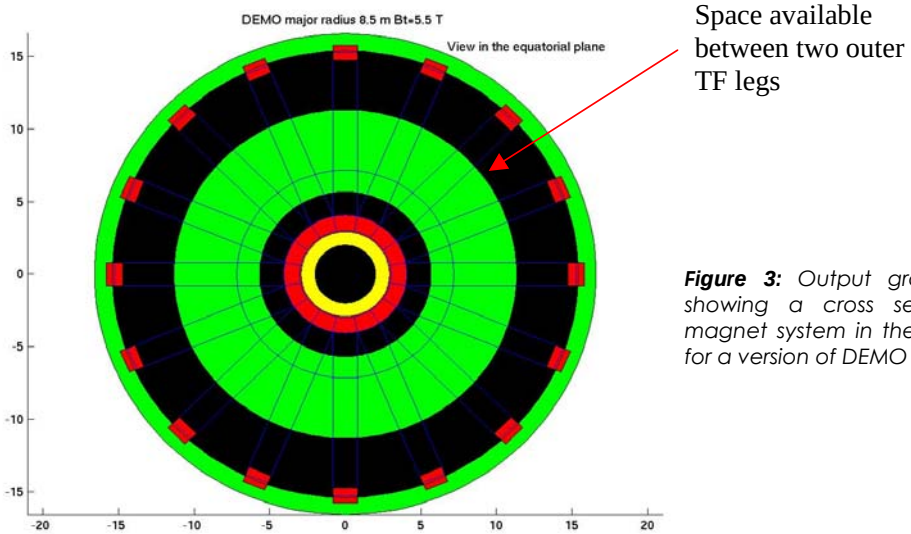

Figure 3: *Output graphic of ESCORT showing a cross section of the TF magnet system in the equatorial plane for a version of DEMO*

Related tasks in the full report:
CEFDA05-1281, CEFDA05-1285, TW5-TRP-002-D03a, TW6-TRP-002-D02, TW6-TRP-006

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ITER Site Preparation

The most important fact in 2006 was the signature of the agreement on the establishment of the ITER international fusion energy organization for the joint implementation of the ITER project, on the 21st of November in Paris.
The *Agence ITER France* (AIF) was created in October 2006, as a CEA service with an independent budget. AIF is the interlocutor of International and European legal entities, in charge in particular of site preparation and fund collection. Moreover, it manages EISS-5 tasks, under EFDA responsibility.
EISS activities have their own steering process with regular meetings and exchange of information with EFDA and the European Commission. The EISS-5 contract covers the period from 21st December 2005 to 21st June 2007. An intermediate deliverable (ref. GA51-DEL-2006-0001) refers to 30 specific deliverables covering the main topics of the EISS-5 contract. The reader who would wish extensive information is requested to ask for these documents.
The following issues are given for reminder and are covering only the main aspects of the contracts. The EISS project (and its corresponding tasks) is, as for previous years, structured to progress on all items on the critical path, with an emphasis on the licensing schedule.

SAFETY & LICENSING:
The required documentation is being prepared for submittal to the French Safety and Licensing Authorities. The writing of these documents (Preliminary Safety Report, files for the INB public enquiry…) is progressing, in strong partnership with the ITER Organization, and is supported by studies performed at European level in parallel.
The Preliminary Safety Report is being reviewed, following a wider ITER project review. The R&D launched within the framework of EISS-5 also allows to complete and detail the writing of several chapters.
Within the framework of EISS-5, several studies were performed to support this Safety and Licensing area, for example the aspects concerning an ITER waste management strategy.
All Safety and Licensing tasks are also covered by the ITER Task Agreement n° 81-15 between EFDA and the ITER Organization.

IN-FENCE STUDIES:
On the 13th of July 2006, the ITER Preparatory Committee agreed to establish a working group comprising the Host State, the Host Party and the ITER Organization to validate decisions on site preparations (so-called Site Preparation Coordination Group).
Simulations of drainage of the ITER site are being carried out, taking into account the results of local measurements.
Following preliminary seismic studies, a programme for qualification of the seismic isolation pads is being performed.
Different possibilities were studied to decouple PF coil manufacturing from the cryoplant, releasing a possible critical point in the ITER schedule. The results of these studies were used by EFDA to suggest to the ITER Organization an alternative to the generic design.

OUT-FENCE STUDIES:
For transportation interfaces, a packaging study for PF Coil n°1 was performed in order to fit within the ITER Site Requirements that are already taken into account by the Regional Direction for Equipment to adapt the itinerary from the harbour to the site.
Several studies allowed to suggest design criteria and applicable standard documents for electrical networks on the ITER site. Moreover, an inventory of reference and regulations to comply with for electrical equipment to be delivered to France is being compiled.
Specifications of external networking needs for the ITER site were also prepared for both internet and telecommunication aspects.

SOCIO-ECONOMIC ASPECTS:
The last step of the public debate consisted in the official publication on the 10th of October of the modalities of continuation of the ITER project.
Socio-economic activities are carried out in public information and communication, as well as in developing strategies to grant support from industries, universities and research establishments.
A "Welcome Office" is already operational to welcome the new ITER staff and ease their relocation, including housing, administrative support and French language courses.

REPORTS:

EISS-2 final report delivered in March 2004	(n° GA1-DEL-2002-0018 rev2)
EISS-3 final report delivered in June 2004	(n° GA31-DEL-2003-0004)
EISS-4 intermediate report delivered in March 2005	(n° GA41-DEL-2004-0006)
EISS-4 final report delivered in March 2006	(n° GA41-DEL-2004-0008)

Related tasks in the full report:
CEFDA05-1294

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Inertial Confinement Fusion

Our Association pursued in 2006 its activities dedicated to keep-in-touch programs dedicated to Inertial Confinement Fusion.

The studies have mainly been dedicated to the six following topics :

Kinetic Effects in Stimulated Brillouin and Raman Scattering Instabilities, where this year has enabled to conclude that while the LDI effects can be described via fluid-type equations and hence be integrated in a large-scale modelling, kinetic effects have to be modelled, if ever, phenomenologically, and perhaps only in a limited parameter regime. Our current and future efforts aim in finding such a modelling for kinetic effects for SRS and SBS, by involving simulations in more than one spatial dimension.

Opacities and resistivity modelling, where the main 2006 objective has been the development of theoretical and numerical tools to study radiative properties and transport coefficients (opacities, resitivities…) as well as the equation of state (EOS) for IFE plasma modelling. This study has been divided into three parts. The first one is related to new methods and code development for opacities and resistivities.
The second is the work on a new fully quantum mechanical approach to atoms and ions in plasmas.
The third is focused on experiments and comparisons to results from numerical codes [Figure 1].

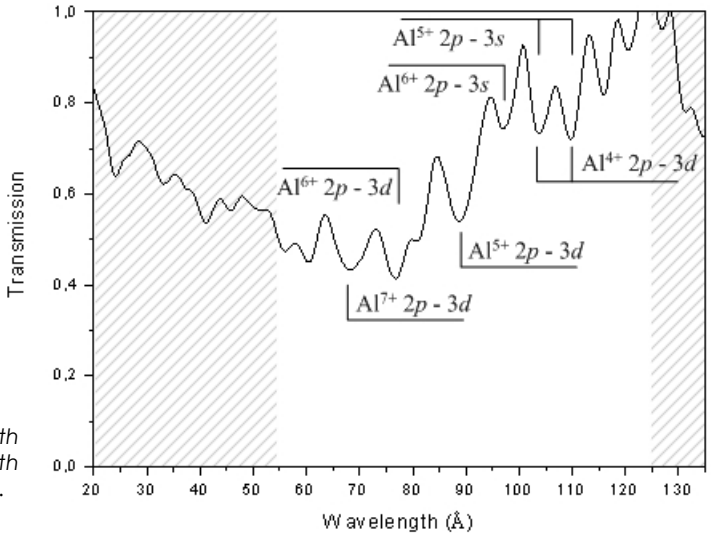

Figure 1: *Comparison of the experimental with the averaged transmission calculated with HULLAC code in case of the Aluminium plasmas.*

Intense laser and particle beams dynamics for ICF Applications, where the research activity is focused on Heavy Ion Inertial Fusion (HIIF) and to the physics of the fast ignitor scenario (FIS). The 2006 results concern the development of a new model to describe atomic collision for heavy ions in relation with HIIF and new theoretical and numerical investigations of the electron transport for fast ignition application.

Application of laser-accelerated high-energy protons for isochoric heating of matter: The experimental campaign has been conducted on the LULI 100TW laser facility in March and November 2006 and involved participants from European countries (Italy, United Kingdom and Serbia), USA and Japan.
In order to improve the measurement of the proton-induced energy deposition, two main diagnostics have been used: (i) a space sampling technique coupled to a streak camera (HISAC) providing a 2D time-resolved image of the heated plasma self-emission at two different wavelengths (in the visible) and (ii) time- and space-resolved frequency-domain interferometry (FDI) [Figure 2]. By using 2 colour

channels, an absolute measurement of the temperature can be obtained, through the ratio of the recorded emissivities, without having to rely on any "absolute" calculation, while using a sample technique allows measuring emissivities with good spatial (~ 40 µm) and temporal resolutions (~30 ps).

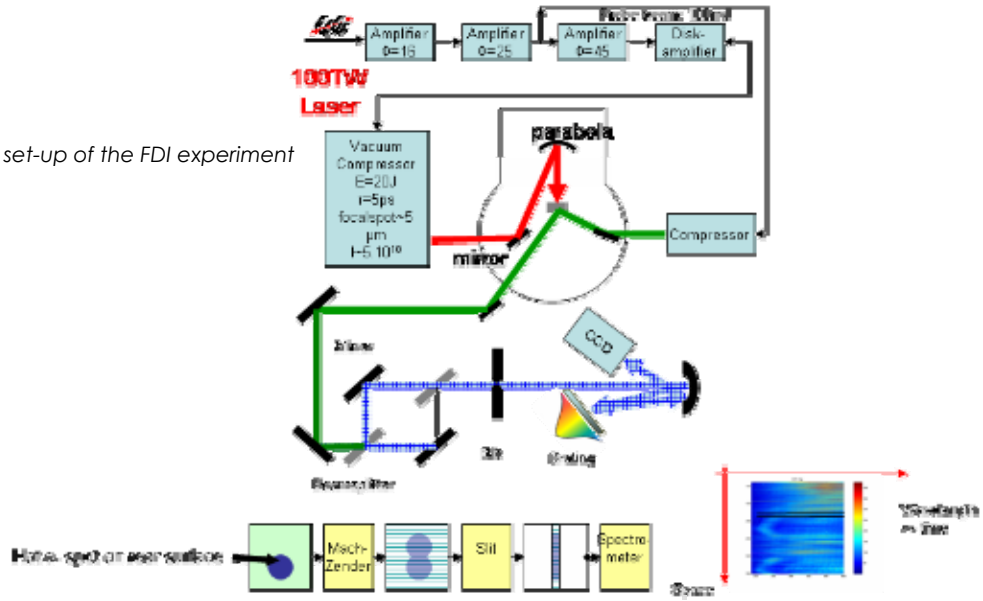

Figure 2: *set-up of the FDI experiment*

It is still under analysis but preliminary results look promising. This study will then continue: foam targets (instead of solid ones) will be used to increase the temperature of the heated medium as well as recently demonstrated ps x-ray backlighting technique to diagnose it.

Interaction of smoothed laser beams with hot plasmas, the objective of this work being to improve our understanding of the laser plasma interaction for the parameters previewed for the future fusion-scale experiments and to develop reduced models of key physical processes, which could be suited for implementation in large-scale numerical codes with predictive capabilities. In particular, three problems have been advanced in 2006: (i) the development of the model for the forward stimulated Brillouin scattering (SBS) driven by the smoothed laser beam and validation of the recently developed laser smoothing methods in the experiment, (ii) interpretation of the experiment on the energy transport and validation of the correspondent module in the hydrodynamic code; (iii) studies of the nonlinear and kinetic processes produced on the nonlinear stage of the SBS in the strongly driven regime.

Diagnostic of hot dense and transient plasma: temporal characterization of dense plasmas using 2D imaging XUV interferometer, where our aim is to diagnose near transient solid-density plasmas using High Order Harmonics (HHG) generated by frequency conversion in a pulsed gas jet as XUV probe beam. We already diagnosed near solid-density plasma by HHG transmission measurements, with a femtosecond range temporal resolution.
Our goal, now, is to get 2D information of plasma temporal evolution by another diagnostic: spatial interferometry at 32nm. First, we focus on the main characteristics of the diagnostic and secondly, we report on the temporal evolution of a plasma created by intense laser irradiation of Aluminium solid target. In the last part, we compare numerical simulations to experimental results.
We presented temporal evolution of an Al plasma characterised by a 2D imaging XUV interferometer [Figure 3] and showed that the mutual coherence of HHG

generated in gas can be used to diagnose dense and transient plasmas. A maximum electronic density of $7 \cdot 10^{20}$ cm^{-3}, 700 ps after the interaction of the pump with the target, has been inferred from single shot interferograms and directly compared to one dimensional-hydrodynamic code simulations exhibiting a quite reasonable agreement. The advantage of this new diagnostic is the possibility to extend considerably the size of the object under study. These encouraging results open large perspectives for bright XUV compact ultra-short sources (as HHG from solid target) usable to diagnose dense transient plasmas, a perspective particularly crucial in the context of the fast ignition for energy production route.

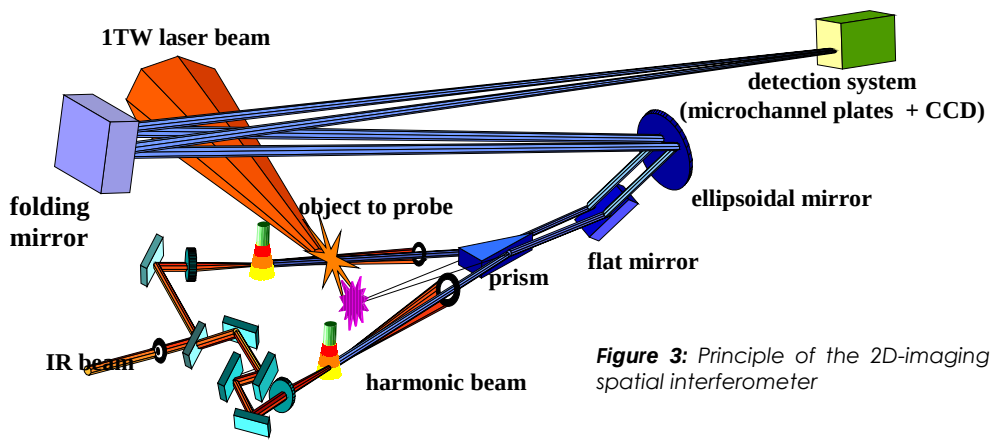

Figure 3: *Principle of the 2D-imaging spatial interferometer*

Related tasks in the full report:
ICF-Instabilities-02, ICF-Opacity, ICF-Particle-Beams, ICF-Protons, ICF-Smoothed-Laser, ICF-XUV-Diag

Related Laboratories: DSM/DRECAM, Ecole Polytechnique (CPhT,LULI), Univ. Paris XI (LPGP), Univ. Bordeaux I (CELIA)

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APPENDIX: Content of the full report

Plasma Facing Components

Remote Handling

Magnet Structure

Tritium Breeding and Materials

Breeding Blanket

Helium Cooled Lithium Lead (HCLL) blanket

Materials development

Reduced Activation Ferritic Martensitic (RAFM) steels

Advanced materials

Neutron source

Safety and Environment

System Studies

Power Plant Conceptual Studies (PPCS)

ITER Site Preparation

European ITER site Studies (EISS)

JET Technology

Fusion technology

Enhancement Programme

Heating Systems Technology Project

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