(executive summary)
Annual report of the Euratom/CEA Association: fusion technology 2003, executive summary

FUSION TECHNOLOGY

Annual Report of the Association CEA/ EURATOM

2003
(executive summary)

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

Cover: Exploded view of the Helium Cooled Lithium Lead (HCLL) breeding blanket concept.
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This document is the executive summary of the full annual report which summarizes activities performed by the EURATOM-CEA Association in 2003 in the frame of the European Technology Programme (both “EFDA” activities and “Underlying Technology” programme). The full report is available in the enclosed CD-Rom and on line at “http://www-fusion-magnetique.cea.fr”. It does not include keep-in touch activities in the frame of the Inertial Confinement Energy program and activities strictly performed for the European ITER Site Studies (EISS) performed in the frame of EFDA in support of ITER negotiations (see http://www.iter.gouv.fr or http://www.itercad.org for further information).

In this document, activities are sorted out according to the main EFDA topics without taking into account the difference between “EFDA” activities and “Underlying Technology” program (see the full report for this distinction 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 program with the objective to propose 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 common exploitation 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 2003 in the frame of the European Technology Programme (both “EFDA” activities and “Underlying Technology” programme).

Three specific operational divisions of the CEA, 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 and magnet and plasma facing components activities.

These activities are also completed by specific R&D collaborations with industry and external laboratories.

In conclusion, we note that this report clearly shows the wide range of competencies available within CEA and other European institutions. Progress is impressive in all subjects related to ITER construction. The authors and the editors should be congratulated for the quality of the contribution and for their dedication to make them available to the entire community. Such rate of progress provides a demonstration of the readiness of Europe to proceed speedily with the construction of ITER. Cadarache was unanimously selected as the European candidate for ITER. One of the main reason for this decision was indeed because the ITER organization would, in Cadarache, benefit immediately from the unrivalled amount of resources invested by Europe on fusion technology during the past decades.

Jean Jacquinot
Mirrors used for diagnostic purposes must survive in an extremely hostile environment and maintain a good optical performance. The response of metallic mirror samples of ITER candidate materials for first bulk mirrors to long pulse exposure-erosion and redeposition has been investigated in the tokamak Tore Supra (TS). Three different materials have been selected: copper based, stainless steel and monocrystalline molybdenum. These investigations included engineering calculations, pre- and post-exposure characterisations such as surface accuracy, reflectance measurements and microscopic (visible and SEM) analysis as well as erosion and redeposition modelling. Six mirror samples (two of each material) delivered by IPP NSC KIPT-Kharkov/Ukraine and two copper alloy-CuCrZr mirror structures from the Euratom-CEA Association have been installed in Tore Supra for long-term plasma exposure. Up to now, roughly 5 hours of cumulated plasma exposure (mainly in deuterium) in addition to glow discharge and boronisation procedures have been performed. Post examination and analysis will be performed in 2004.

In tokamaks, magnetic measurements are one of the most important diagnostics to control the plasma position. Such instrumentation is close to the plasma and needs to withstand severe constraints. The method foreseen to perform magnetic measurements in such an environment is the association of passive coils with electronic integrators. However the most important issue of electronic integrators is their drift, which introduces an increasing absolute error while integration time evolves, reducing accuracy on plasma localization. In 2003, work has been carried out in order to qualify the analog integrators used on Tore Supra for ITER criteria. It has been shown that from the drift point of view, the Tore Supra integrators are appropriated for several thousand of second and therefore satisfy the ITER requirements. The effects of Radiation Induced Electro Motive Force (RIEMF) current on the integrator saturation need further investigations.

A new diagnostic for thermographic analysis has been developed for JET. This system should allow to see a large section of the internal components in the vessel such as divertor, main chamber, ICRH antenna etc., with the aim to measure the surface temperature during normal operation and off-normal events such as ELMs and disruptions (Figure 2). This diagnostic is ITER relevant both for its technology and for its physics output, which is required to validate the choice of materials in the divertor and first wall of ITER. The technical specifications have been finalized in order to launch the manufacturing. Unfortunately, the Call for Tender launched in March 2003 failed due to the cost. Therefore, a redefinition of the project has been made with reduced objectives (withdrawing of the visible view and of the capability to operate during D-T). The new design has started in July and a second call for tender
has been launched in November 2003, allowing the possibility to implement the diagnostic on JET during the restart in May 2005.

The Euratom-CEA Association is involved in the development of the first ITER Neutral Beam (NB) injector (in particular the source and the accelerator), in the design of the ITER NB Test Facility and in the preparation of the technical specifications for the procurement of long delivery items.

The development of negative ion sources is carried out with the KAMABOKO ion source on the MANTIS test bed in collaboration with JAERI, Japan, who supplied the ion source. During previous campaigns, continuous beam pulses of duration up to 1000 s have been demonstrated both in hydrogen and in deuterium; however, with both isotopes the current density was found, at the expected arc power and filling pressure, to be low in comparison to the anticipated ≥ 200 A/m². During long pulse operation, an unexpectedly high rate of Cs consumption has been observed. Work over the past year has been aimed at identifying the causes of the high Cs consumption and the low value of accelerated D- current density. An attempt to control the flux of Cs into the source plasma was made by introducing a Cs trap (Figure 3) into the source but with no incidence on the negative ion yield. A "new" Cs trap has been designed with two functions: to trap Cs, as before, and to shield the Cs from the evaporated tungsten. It is proposed to

Figure 2: View of the image obtained with the wide-angle infrared endoscope and the IR camera.

Figure 3: The Cs trap
carry out this experiment in 2004. To investigate further the accelerated current loss, a new “neutralizer” is proposed. This will be segmented, with each segment instrumented to allow the spatial distribution of power in the neutralizer to be determined, and, hopefully, to gain insight into the source of the extra current. The European concept for a 1 MeV, 40 A negative ion based accelerator for the neutral beam system on ITER, the SINGle GAP, SINGle APerture (SINGAP), is an attractive alternative to the ITER reference design, the so-called Multi-Aperture, Multi-Grid (MAMuG) accelerator. An experimental campaign has been dedicated to increasing the current density of the accelerated D- beam. The best current density obtained was 120 A/m² of D- at the target with an energy of 498 keV for 2 s (Figure 4). The experimental results have been modelled and dedicated experiments have been performed to test various aspects of the simulation codes in detail, especially the effect of space charge. Very good agreement between the experimental data and calculations has been found for cases with low space-charge conditions. The design of the 40 A SINGAP accelerator for ITER is in progress. A new ion source and “ITER-like” prototype accelerator have been under construction during 2003. It will have an actively cooled version of the ion source currently used on the 1 MV test bed and it will be able to demonstrate the beam optics of the ITER SINGAP accelerator.

An **ITER-relevant ICRH** launcher will be installed on one of the main horizontal ports of JET during the 2004 shutdown. The design of the ICRH system was carried out in the frame of the EU ICRH project. The Euratom-CEA Association has contributed to:

- The design of an alternative antenna front, housing and wave transmission line (VTL) configuration using large capacitors (Figure 5). This concept was finally not retained due to the unproved large capacitor reliability (in spite of the ongoing R&D work) and to the maintenance scheme in case of a capacitor failure, implying a longer shutdown than in the design using small capacitors.
The design, fabrication and high power testing of RF contact test lines. The fabrication of the test lines took place at the end of 2002. The power tests have been performed in April and May and reported in September 2003.

The design of the antenna limiter (including call for tender documentation and evaluation).

Related tasks in the full report: CEFDA01-624, CEFDA01-645, CEFDA01-646, CEFDA02-670, CEFDA02-676, CEFDA02-689, CEFDA03-1031, CEFDA03-1044, CEFDA03-1047, TW1-TPHIRANT, CEFDA02-1003, TW2-TPDS-DIADEV-D01, TW2-TPDS-DIADEV-D02.

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Vessel/In Vessel activities

Work performed by the Euratom-CEA Association in this field in mainly focused on the development of manufacturing and assembling techniques (HIP, welding). The development of an industrial cutting and welding laser tool for fabrication and maintenance of hydraulic connector parts of the ITER blanket module has been achieved in 2003. The main objective was the welding/cutting of a 100 mm diameter, 3.75 mm thickness tube from the inside, using an Nd:Yag laser with optical fiber beam transportation. The access to weld this part is through another tube, 30 mm diameter. The weldability with a laser welding tool has been demonstrated (Figure 6). Severe constraints on the cutting are the presence of a 3.75 mm thickness wall, positioned 5 mm behind the tube to be cut, and the low diameter (30 mm) of the access hole. Cutting needs a nozzle close to the surface to be cut; a retractable nozzle is therefore used (Figure 7). Parameters allowing cutting without degradation of this second wall have been obtained.

Figure 6: View of the tool: welding operation
Figure 7: The two positions of the welding cutting nozzle
During the welding process, residual stresses are produced that can cause distortion of a component, crack initiation or problems in the lifetime of the component through enhanced fatigue or corrosion. As these effects are undesirable, control and minimisation of the welding stresses would be a great benefit. A study on mitigation of welding distortions and residual stresses in welding has been performed in collaboration with the Process Team of British Aerospace in Bristol (Great Britain). Different techniques have been investigated, such as Thermal Tensioning by Heating (TTH), Thermal Tensioning by Cooling (TTC), and Mechanical Tensioning (MC). The most applicable is Thermal Tensioning by Cooling using sprays of cryogenics liquids (Figure 8).

Work on the manufacture of the ITER Primary First Wall (PFW) panel by HIP forming is ongoing. This method consists in manufacturing a PFW panel made of a serpentine tube expanded into a proper matrix. The objective is to obtain the minimum achievable pitch between the Stainless Steel (SS) tubes and the SS tubes and the front face. During 2003, dummy mock-ups were produced to validate the manufacturing feasibility. For all the mock-ups manufactured, during the HIP cycle one of the coils constituting the serpentine leaked (Figure 9), because of cracking preventing diffusion bonding welding between the parts. None of the techniques investigated during this work is appropriate for the manufacture of the SS panel. To complete this task a new design has been proposed by the EFDA Close Support Unit, and the feasibility of this design will be checked by manufacturing a mock-up.

Related tasks in the full report: CEFDA03-1067, TW1-TVV-HIP, TW1-TVV-ONE, TW2-TVB-HYDCON, TW2-TVV-DISMIT, TW2-TVV-ROBOT, TW3-TVM-JOINT, TW3-TVV-DISFREE, TW3-TVV-ROBASS, TW3-TVV-UTDYNA

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Plasma Facing Components activities

The flat tile technology for plasma facing components has several advantages: existing industrial experience, possible repair process, dedicated non-destructive examination techniques. However, if one tile is missing on such a high heat flux component subjected to a convective flux with a glancing incidence, the thermal loading of the neighbour tile would be doubled, leading to a possible cascade failure. An evaluation of the possibility of failure in cascade of flat tiles has been performed in 2003. Different mock-ups were manufactured and high heat flux tested at the electron beam facility FE200. The main features of the 4 tested mock-ups were: heat sink made of CrCr2Zr armored with NS31 CFC flat tiles or with tungsten teeth, bonding of refractory materials to heat sink via Active Metal Casting (AMC) concept and electron beam welding (Figure 10).

As a typical glancing incidence of 3° is difficult to adjust experimentally with an electron gun, preliminary calculations were made and have shown that lack of one tile could be well simulated experimentally with a quasi perpendicular peaked heat flux locally deposited on the edge of a tile. A first experimental step along this line was performed. Local sublimation was observed in the case of NS31 CFC; the W melted under the peak heat flux. However, the 4 components resisted well to cycling heat fluxes representative of a normal loading of 10 MW/m² without any tile detachment.

A second experimental step consisted in the simulation of cascade failure triggering with glancing incidence. For this, some tiles of the hypervapotron were machined. The 3° of incidence could be adjusted, the 4 components were loaded with fluxes as high as 220 MW/m² on the leading edges adjacent to the machined tiles. A fast erosion of the NS31 CFC leading edge was observed (electronic shaping); the pure W melted immediately on the edges and presented cracks at the surface (Figure 11). The cracks on the surface may be attributed to the brittleness of pure tungsten. The use of an alloy of tungsten/lanthanum or tungsten/rhenium - more ductile - could be recommended. However, no triggering of cascade failure event was observed.

These new experimental results show the robustness of the flat tile technology, optimized with a hypervapotron cooling concept.

Thermal fatigue is one of the most important damaging mechanism for the plasma facing components (PFCs) of the ITER machine due to the high number of operating cycles (several thousands) and to the expected surface heat loads. Therefore, an assessment of the behaviour of PFCs under cycling heat loads is essential to
demonstrate the viability of the selected design solutions. A full scale prototype mock-up of the **ITER divertor vertical target was high heat flux tested** in the FE200 facility. The reference solution (CFC monoblocks made of SEP NB31 tiles and CrCrZr tubes) confirmed a good behaviour under fatigue testing. The ITER goal of 20 MW/m² was exceeded with the HIP technology. Tungsten monoblocks with lamellae sustained well 100 cycles at 5 MW/m² and 600 cycles at 10 MW/m², which is the double of the ITER design for the upper part of the vertical target. Afterwards, a water leak occurred on the bended part, showing a radial crack propagation coming from between two OFHC compliant layers to the internal CuCrZr tube. Analysis and calculations are being made to tentatively explain this phenomenon. High heat flux tests will continue in 2004.

**Destructive examinations of mock-ups and Primary First Wall (PFW) panels** have been performed, after high heat flux thermal fatigue tests in the FE200 facility (for parts without Be) or in Jülich and Brasimone (for parts with Be). Up to now, only mock-ups without Be have been examined. These mock-ups are made from a combination of copper alloy as heat sink material (CuCrZr or Glidcop®) and Stainless Steel as structural material. The interfaces observed are Cu alloy/Cu alloy, Cu alloy/SS tubes, and SS tubes/SS substrate. The main conclusions of this study are that defects or cracks found after examinations (Figure 12) are not formed during the thermal tests but they are directly related to a manufacture problem. In particular, greater precautions must be taken during the welding operations required for the manufacture of mock-ups.

The fabrication of a lot of fusion reactor components is based on the **Hot Isostatic Pressing (HIP) technique** and there is a need to improve laboratory practices in order to improve the industrial relevancy of these processes. Four subjects are studied: machining and cleaning of surfaces for diffusion welding, modelling for powder compaction, and, for both processes, outgassing and tool and anti-diffusion materials. Numerical simulation of the HIP process is requested to avoid problems such as tearing of the canister during the heating stage due to difficulties to close clearance, unwanted deformations, rupture of the brittle material due to high tensile residual stresses or deformation localisation... HIP modelling...
and experimental characterisation of a Eurofer powder is ongoing. Since no data is available for this powder, an example using 316L powder has been studied. This task has permitted to enlighten deficiencies in terms of modelling and experimental data. The conclusions can be applied directly to future Eurofer characterization (Figure 13).

Related tasks in the full report: CEFDA01-585, CEFDA02-583, CEFDA03-1029, CEFDA03-1051, CEFDA03-1077, DV4.3, TW0-T438-01, UT-VIV/PFC-HIP, UT-VIV/PFC-NanoSc, UT-VIV/PFC-W/Coat

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Remote Handling activities

Due to neutron activation, the repair, inspection 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 Association Euratom-CEA, such as the development of specific tools for ITER or the radiation tolerance assessment of required electronic components in a fusion environment.

In 2003, the studies on the feasibility to perform the remote maintenance (cutting, welding, inspection) of the ITER divertor cooling pipe with bore tools have been achieved. A general standard for the cooling pipes inside the cryostat was assumed to be stainless steel bent pipes (100 mm internal diameter, 6 mm thick, bend radius > 400 mm). Cutting/welding is required up to 10 m away from the tool insertion. These studies included design activities, manufacturing and testing of a demonstrator of the basic steps of 100 mm bend pipes maintenance. Clamping modules, an alignment module, a tack welding tool and a swaging tool were manufactured and tested separately (Figure 14). Integrated tests on a swaging operation were successfully performed during 2001-2002.

Figure 14: Carrier and bore tools for ITER divertor cooling pipe. A right, the test line.
The possibility to perform the cutting and welding operations with a single laser tool was studied and tested in 2003. A specific laser tool was designed, manufactured and integrated in the bore tool system. The total length of the carrier in the travelling configuration is about 1900 mm (Figure 15). A viewing system is used to find, define and set a welding path, which follows the previous cut. The interfaces and functions were initially clearly defined, and assembly of the tool in the existing carrier were made without any problem. Functional verifications of all the equipment were carried on successfully. However, validation tests in a real situation were impossible and showed that the limits of the current prototype were reached.

The main technical reasons that were identified are the complexity of the tool (as an example, the number of wires increased from 6-9 in the previous configuration to 24) and the weight of the carrier which is now too important due to the modifications. An essential key of success in Remote Handling technology is to reduce to the minimum the difficulty of the task to be performed, probably resulting in an increased number of tasks and tools. In this particular case, it has been concluded that distinct tools to cut and weld the pipe would reduce the complexity of the requirements and could probably be an interesting alternative. On the other hand, recent ITER designs are converging towards a new standard for the cooling-pipe, focused on a 89mm or 64mm diameter, seemingly incompatible with a unique bore tool system. Due to this significant design modification, the reference maintenance scheme requires changes. The reflections are now converging on cooperative work between a carrier sent into the pipe to perform some basic operations and an orbital tool set in position by a slave manipulator used to perform maintenance tasks in the divertor region. Accessibility of an orbital tool model to the divertor cassette cooling pipes was checked: preliminary checks show a possible scenario, but a lot of operational assumptions require further investigation to assess the feasibility (Figure 16).
The Euratom-CEA Association performs a R&D program to demonstrate the feasibility of close inspection of the ITER Divertor cassettes and Vacuum Vessel first wall. It is assumed that a long reach and limited payload carrier penetrates the first wall using the 6 penetrations evenly distributed around the machine and foreseen for the In-Vessel Viewing System (IVVS). The feasibility of this carrier has been assessed in 2002 (Figure 17). The main requirements of this multipurpose tool are driven by the reactor operating (nuclear, vacuum and/or temperature conditions) and maintenance rationales. The main requirements are:

- a temperature of the vessel during the intervention of 120 °C,
- a vacuum of 10⁻⁶ Pa,
- a desired payload of 10 kg for the process tool
- the possibility to operate under magnetic field (option)

A feasibility study of an In-Vessel Penetrator (IVP) working in vacuum, at high temperature and large magnetic field, has been performed in 2003. The results show that the major part of the components needed to work under magnetic field are not available today. Pending issues to design the IVVS deployer will require significant R&D programs to demonstrate the feasibility of Remote Handling under magnetic field. In order to validate the operation without magnetic field but with operational conditions in term of vacuum and temperature, the IVP original design has been upgraded with all selected suitable technologies. This new prototype is now called AIA for Articulated Inspection Arm (Figure 18). The selected main design options are:

- Use of metallic alloys for the structure materials such as titanium.
- Some other non organic materials could also be used like Vespel.
- Use of HCMOS military electronics components with a dedicated robot network. Electronics will be embedded in tight boxes with tight feedthroughs or connectors.
- ...

The next step of this study is the validation of the technologies on a single module prototype. Manufacture of this prototype module will be completed at the beginning of 2004 and a test campaign under ITER relevant conditions will be proposed in Tore Supra.

Hydraulic manipulators are candidates for fusion reactor maintenance. Their main advantages are their large, with respect to volume and mass, payload, their reliability and their robustness. However, due to their force control limitations, difficulties could occur for precise manipulations. CEA, in collaboration with...
CYBERNETIX and IFREMER, has developed the advanced hydraulic robot MAESTRO. The Euratom-CEA Association involvement concentrates on the development of a dynamic simulator of the robot arms and on the use of water instead of oil as a fluid medium. Servo-valves are on the critical path for the change of fluid medium. Flow control servo-valves using water are already available on the market but their reliability in long term tests still needs to be proven. In cooperation with the manufacturer In-LHC, the development of a prototype of pressure control servo-valve using water that fits the requirements of the Maestro slave arm is ongoing.

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 electronic components envisaged up to now for the ITER instrumentation are based on existing electronic technologies, using digital data transmission and time domain multiplexing. The radiation tolerance assessment of all components is therefore required before their use in instrumentation for ITER.

In 2003, complex electronic components and functions have been evaluated under different irradiations conditions. Most of these components remained functional after the end of the irradiation. A heating recovery period at 150°C has practically allowed retrieving of the nominal characteristics. Concerning the multiplier function of the AD534 components, it has been confirmed that the observed radiation-induced drifts do not disappear in a combined radiation and temperature environment. The influence of higher recovery temperatures (up to 220°C) has also been confirmed. A further investigation of this component using input adjustment to control output drifts has to be performed as soon as possible to validate the feasibility of its use.

Concerning logic components, irradiation of mock-ups (Figure 19) confirmed the possibility to use CMOS technologies, even if the increase of supply voltage has led to some failures when it is not large enough to allow correct CMOS conduction. In order to take into account the failures occurring when different CMOS technologies are integrated to design a complex function, the use of open-drain components is required to avoid voltage disagreement. In other terms, the partial redesign of the function will be encouraged to limit technology interference.

**Figure 19:** mock-ups of logic functions when bottled before irradiation at SCK/CEN facility (18 days at a dose rate close to 20kGy/h and an average temperature around 60°C).
Magnets structures and Integration activities

In this field, the Association task was to provide input information for establishing the final dimensional details of the cryoplant buildings and to develop further details for the cryoplant design. The design of the cryolines and Cold Valve Boxes (CVBs) for the distribution of the cryogenic fluids to 10 torus cryopumps has been performed. This work also included the development of a wide range of working characteristics of the LHe plant in order to study its reliable operation over the full range of plasma scenarios, including cool-down after a fast energy discharge of the TF coils.

The development of NbTi conductors and connections for the ITER Poloidal Field (PF) coils has to address several questions, such as characterization of strand performances, control of strand-to-strand conductance, design of connections, as well as conductor behavior in operation. In 2003, two subsize connections and conductors relevant to PF operation have been manufactured and tested in the JOSEFA test facility (Figure 21). Some additional analysis were performed on the Poloidal Field Full Size Joint Sample (PF-FSJS) and were found consistent with the results obtained in 2002. A final upgrading of the JOSEFA test facility was undertaken to reduce the thermal gradient along the sample and to implement the temperature regulation for the next joint development studies, which should be completed within the year 2004.

An extensive characterization of three types of NbTi strands candidate for the ITER PF Coils conductors has been performed. One strand was provided by VNIINM Bochvar Institute (Russia, Figure 22) and comes from the production used for the Poloidal Field Coil Insert (PFCI)
The two other strands come from the production used for the PF-FSJS conductor. One was Ni-plated and provided by Europa Metalli and the other had a CuNi internal barrier and was provided by Alstom. The strand used in PFCI showed consistent behaviour with a good agreement with VNIINM tests. The Europa Metalli and Alstom strands showed the central role of self-field in the PF-FSJS conductor behavior. All strands meet the $J_C$ ITER Design criteria ($4.2T, 5T$) and none the $\Delta T_{\text{MARGIN}}$ criterion (PF1 and PF6 coils). This is mainly due to the non-linear $J_C$ variations with temperature.

The thermohydraulic properties of cable-in-conduit conductors with a central channel, and in particular the heat transfer coefficient between the annular area and the central channel, are studied in a specific experimental activity. This coefficient drives the recooling time of forced flow coils and the quench behavior. It can hardly be theoretically evaluated; only experiments, possible at room temperature, can bring information about it. A dedicated experimental facility, which operates in pressurized water at 60°C was installed. The first test results have been compared to a dedicated code and have led to a first evaluation of the heat transfer coefficient. However, an unexplained temperature evolution along the conductor was observed. It was then decided to upgrade the facility as well as its instrumentation in order to increase the measurement accuracy and analysis possibilities. Complementary tests are planned in 2004 to try understanding the temperature evolution along the conductor.

The Euratom-CEA Association has responsibilities in the testing of the Toroidal Field Model Coil (TFMC). The analysis of the measurements performed during the phase II tests of the ITER TFMC, carried out in 2002, have been completed. These tests demonstrate first that this coil design relies on sound concepts, suitable for the ITER TF coils. However, they have also pointed out a decrease of the achievable current density when increasing the Laplace force applied to the conductor, which calls for a revision of the design of the ITER Nb3Sn conductors. The previous criteria were in some cases too optimistic and in some other cases too pessimistic in a range which can overall correspond to a decrease of the temperature margin by 0.3 K to 0.4 K. This decrease can be easily compensated, keeping the same size of conductor and a stainless steel jacket, by taking benefit of the progress in critical properties of superconducting materials during these last ten years. On the other hand, manufacture of the PFCI-FSJS was delayed, postponing the tests to 2004.

Related tasks in the full report: CEFDA02-663, CEFDA02-684, CEFDA03-1015, M50, TW1-TM-CODES, TW1-TMC-SCABLE, TW1-TMS-PFCITE, TW2-TMST-TOSKA, TW3-TMSC-ELRES

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Tritium Breeding Blanket activities

In 2002, EU has endorsed the decision to concentrate the work on blanket activities on a single coolant, helium. Until then, two different coolants were envisaged for the EU Breeding Blankets: i) pressurized He for the HCPB concept (Helium-Cooled pebble-Bed) and ii) pressurized water for the Water Cooled Lithium Lead (WCLL) concept. This last concept has been reoriented toward the Helium Cooled Lithium Lead (HCLL) breeding blanket concept for DEMO and its corresponding Test Blanket Module (TBM) to be tested in ITER. The leader for this concept is the Euratom-CEA Association. It relies on the use of Reduced Activation Ferritic Martensitic (RAFM) steel, the Eurofer, as structural material, eutectic lithium lead as breeder material, neutron multiplier and tritium carrier and high temperature helium as coolant. In 2003, the HCLL design has been developed and optimized with regard to its multiple objectives.

The generic DEMO blanket module consists of a box formed by an U-shaped First Wall / Side Walls cooled by He circulating in toroidal/radial/toroidal channels and closed by two cooled caps (top and bottom). A Back Plate acting also as He collector and distribution system closes it, at the rear (Figure 23). It is stiffened by He cooled poloidal-radial and toroidal-radial stiffening plates forming square radial cells. In each cell is inserted a cooling Breeder Unit (Figure 24) consisting of 5 parallel horizontal cooling plates, connected to a breeder unit back plate ensuring the insert rigidity. He at 300°C/500°C inlet/outlet temperature and 8 MPa pressure has been chosen as coolant. One circuit is used for cooling both the box and the breeder zone. Since the first wall is the more solicited component, both in terms of load and in terms of neutron irradiation, it has
been chosen to cool it with high velocity low temperature He. The liquid metal is fed from the back at the top of the module and recovered in the back, at the bottom. It circulates radially-poloidally between the cooling plates of a unit cell and it passes from one cell to the one below through appropriate openings placed alternatively at the front and at the back of the horizontal grids. Thermal, thermal-hydraulic and mechanical calculations have been carried out showing that this structure is able to recover the deposited heat with a good thermodynamic efficiency without exceeding the thermal and mechanical limits in both normal and accidental conditions. Neutronic calculations show a tritium breeding ratio (TBR) close to 1.22 for a radial thickness of 75cm and an Li6 enrichment of 90%. Although this concept seems viable, fabrication techniques and assembling sequences should be investigated with the help of experts and industry. Relevant experimental activities will be launched in 2004, including also specific experiments dedicated to the He technology (tribology of sliding parts, leak tightness between static parts, behaviour of inner thermal barriers in static helium).

In the frame of the HCPB concept led by Euratom-FZK, Euratom-CEA is involved in the procurement, characterization and optimization for cost reduction of the Li2TiO3 fabrication process. A 6 kg batch of Li2TiO3 pebbles (with a size distribution in the range 0.6 to 0.8 mm, produced by pre-industrial means in 2002) was characterized in 2003, the pebbles achieving the required specifications. A new 2 kg batch of pebbles, with a size distribution in the range 0.6 to 0.8 mm, was also produced. The feasibility of lithium recycling is being investigated in collaboration with the Euratom-ENEA Association.

Concerning the manufacturing of the Test Blanket Modules first wall, one of the options is joining rectangular tubes and smooth plates by Hot Isostatic Pressing. The bending process for square tubes has been assessed. The bending reproducibility is not very good, but anyhow better than the accuracy (Figure 25).

Related tasks in the full report: TW2-TTBB-002b-D01, TW2-TTBB-002b-D03, TW3-TTBB-002-D02, TW2-TTBC-001-D01, TW2-TTBC-002-D01, TW2-TTBC-002-D03, TW2-TTBC-005-D02, UT-TBM/BB-He.

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Structural Material activities

The European material fusion programme is mainly dedicated to the development of Reduced Activation Ferritic Martensitic (RAFM) steel, the EUROFER.

Irradiations experiments at high doses (up to 70-80 dpa) and for irradiation temperatures lower than 400°C, where materials are susceptible to reach a high level of hardening and embrittlement, are conducted in the BOR60 reactor. Post irradiation experiments of sample irradiated at 32-42 dpa are ongoing. The preliminary results show that EUROFER 97 has a quite low irradiation-creep deformation. As expected, this material hardens during irradiation but it should keep a reasonable level of ductility.

Different studies on the qualification and fabrication processes have been performed. The production of a solid Eurofer alloy by HIP (Hot Isostatic Pressing) compaction of EUROFER powder offers the great advantage to allow the manufacturing of parts with a complex design. In order to qualify the global process, different possible defects have been investigated. The first one, studied in 2002, was gaseous contamination during storing/handling operations or during degassing operations. The second one, studied in 2003, is particulate contaminations originating from the powder atomization and their influence on mechanical properties. A special Eurofer material has been elaborated by adding nickel powder to the Eurofer powder, in order to simulate the frequently observed nickel contamination (Figure 26). Hardness and tensile properties show only a low effect but Charpy impact properties at 20°C indicate a strong drop and fracture surface observations indicate a brittleness of the powder particle surface.

The EUROFER weldability is also investigated. A mock-up including a representative welding junction has been defined for the next experimental welding investigation, according to the HCLL Blanket module design. A metallurgical model has been developed for EUROFER-97. A first numerical simulation, for a 14 mm thick plate, has been successfully compared to experiments. An additional 2D simulation was carried out on the same HCLL mock-up, showing the good capability of the model to represent the thermal effect of TIG. The data generated by the Eurofer European material program are collected, validated, harmonized and compiled in a database in order to propose material properties permitted for design and licensing of components fabricated with the EUROFER steel. Most of the conventional design limits can be derived with adequate reliability and used for the ITER test blanket modules at moderate irradiation doses.
Additional work is needed to substantiate the fatigue and fracture toughness databases and in all cases for higher dose irradiations.

Modelling activities on microstructural evolution have been performed by the Euratom-CEA Association since 1994 in the frame of the underlying programme. Since 2003, modelling activities are also included in the EFDA programme. The Euratom-CEA Association activities aim at providing a database at the ab initio level (i.e. in the framework of the Density Functional Theory, DFT) of energies and structures for a set of characteristic atomic configurations involving helium atoms and vacancies in the a-Fe lattice. These data can be used directly for a better characterisation of small helium-vacancy (He-V) complexes in a-Fe or as input data in a kinetic model or for the fit of semi-empirical interatomic potentials, which can in turn be used for the simulation of larger He-V clusters and/or for molecular dynamics simulations. An ab initio calculation scheme for the study of helium-vacancy defects in a-Fe has been successfully set up in 2003.

Advanced structural materials such as SiC-SiC ceramic composites and tungsten alloys are also considered in the frame of the European fusion program. A multi-scale model to simulate the mechanical behaviour of a SiC/SiC composite has been studied and its implantation in the finite element code CASTEM has been proposed.

The behaviour of these advanced materials under irradiation needs to be address. A common irradiation (5 dpa equivalent Fe) of SiC-SiC ceramic composites and tungsten alloys samples at two temperatures, 1000°C and 600-650°C, will be performed in the Osiris reactor at CEA-Saclay. The 2003 activities were mainly focused on the design of the irradiation rig. Start-up of the irradiation is foreseen in 2004.

To test and fully qualify candidate materials up to the expected doses of a fusion power reactor, a high flux source of high energy neutrons, presently not existing, has to be build and operated. A neutron source from the deuterium-lithium (D-Li) stripping reaction has been selected as the basic concept of the International Fusion Materials Irradiation Facility (IFMIF). This device requires the generation, by a linear accelerator (linac), of a 250 mA continuous deuteron current at a nominal energy of 40 MeV, with provision for operation at 30 MeV and 35 MeV. The Euratom-CEA Association is involved in the development of accelerator components (Electron Cyclotron Source, Radio Frequency Quadrupole,...). Work performed in 2003 dealt with the development of critical accelerator components and analysis of possible design alternatives. On the other hand, the H2+ ions acceleration has been investigated, which would help the commissioning of such a high intensity deuteron facility, especially by limiting neutron production before the industrial running mode.

Related tasks in the full report: TW1-TTMS-002-D16, TW1-TTMS-003-D12, TW2-TTMS-001b-D02, TW2-TTMS-002a-D17, TW2-TTMS-004a-D01, TW2-TTMS-004a-D04, TW2-TTMS-004b-D01, TW2-TTMS-004b-D02, TW2-TTMS-005b-D03, TW2-TTMS-005b-D05, TW2-TTMS-005b-D09, TW3-TTMS-006-D05, TW3-TTMS-007-D02, TW2-TTMA-001a-D10, TW3-TTMA-001-D04, TW3-TTMA-002-D04, TW3-TTMI-001-D01, TW3-TTMI-001-D03, TW3-TTMI-001-D05, UT-TBM/MAT-LAM/DBTT, UT-TBM/MAT-LAM/DBTTpred, UT-TBM/MAT-LAM/Opti, UT-TBM/MAT-Modpulse, UT-TBM/MAT-ODS.
Safety and Environment activities

In the frame of the validation of the codes used for the safety assessment of ITER or future fusion power plants, a benchmark is under progress to compare experimental tests achieved on the EVITA facility and calculations performed by different pressurization codes (CONSEN, PAX, MELCOR, MAGS). The objective of the EVITA facility is mainly the investigation of the pressurization rate and heat transfer characteristics of water injection into the vacuum vessel of ITER when a cryostat event occurs (Figure 28). The analysis of the tests performed in 2002 and 2003 has resulted in important data (pressure evolution, ice layer formation) on the different phenomena occurring in the vacuum vessel during water or steam ingress. The second part of the year was devoted to set up the incondensable gas injection on the EVITA facility. Several tests have been realized to demonstrate the feasibility of the experimental program.

In 2003, some improvements were...
made on the PAX ITR 2 computer code in order to model EVITA experiments. The most important work was carried out on the cryogenic structure where a specific development was performed to evaluate the kinetics of the pressurization and the ice growth. Finally, EVITA PAX ITR 2 calculations are globally satisfying. Possible improvements have been identified to obtain a better modelling of EVITA transients.

In the framework of fusion waste management, detritiation processes could allow reducing the waste level. Studies have also been performed to determine different processes that could be used for tritium removal. For steel detritiation, melting with gas bubbling seems to be one of the most promising methods. Concerning carbon dust and flakes, full combustion seems to be the best detritiation process. Even if large quantities of secondary wastes are produced, this process can be selected because the concerned volumes are small. Thermal process under Hytec gas (Argon + 5% H2) allows graphite detritiation without any destruction. Optimal detritiation temperatures are between 573K and 1073K. An experimental program is also ongoing to validate modelling of mechanisms involved in tritium trapping and desorption. Different thermal treatments in the solid phase have been investigated to lower the residual tritium concentration (by desorption reheating), to reduce the diffusivity of residual tritium (by changing the microstructure), and to reduce the kinetic of residual tritium desorption (by forming barrier films).

In order to limit the tritium inventory inside the vacuum vessel of fusion devices, in-situ detritiation processes of plasma-facing components could be used. Detritiation processes using lasers are envisaged. This may provide fast heating of an exposed surface (1000K on a thin, 1 - 100 µm, near-surface layer), thus resulting in detritiation either by hydrogen desorption from the surface or by ablation of this near-surface layer. Low and high repetition rate laser benches with a sufficiently complete control and measurement equipment were developed and applied for graphite heating and ablation studies (Figure 29). The ablation thresholds for graphite (≈ 1 J cm-2) and for co-deposited layer (≈ 0.4 J cm-2) surfaces were determined experimentally. The obtained ablation efficiency was estimated to allow a 40 µm thick co-deposited layer ablation with a rate of ≈ 1 m2 per hour with 500 W laser power. The feasibility tests carried out with Glow Discharge Optical Spectroscopy diagnostics demonstrated that heating by nanosecond lasers is sufficiently efficient to perform detritiation of thin co-deposited layers. Theoretical models are also developed to compare simulation data with experimental results on laser surface heating and ablation. These investigations are essential both to obtain the optimal interaction regime and to improve the laser detritiation method performances.

Related tasks in the full report: TW1-TSS-SEA3.5, TW1-TSS-SEA5, TW3-TSS-SEA5.5-D03, TW1-TSW-002, TW3-TSS-SEA5.3, TW3-TSS-SEA5.5-D03, J W3-FT-3.14, J W0-FT-2.5, UT-S&E-LASER/DEC, UT-S&E-LiPbwater
System Study activities

The European **Power Plant Conceptual Study** (PPCS) was launched in January 2000. Its objectives are to assist in assessing the fusion energy status and in establishing coherence and priorities in the EU fusion program. The Power Plant Conceptual Study is scheduled to be completed early in 2004 with the publication of an overall final report. In 2003, only some limited activities have been performed to improve specific points, such as the waste management of reactor models A and B and the design of the water cooled divertor (reactor model A). This divertor is based on a Eurofer/Tungsten divertor with a high temperature compliance layer (soft graphite “PAPYEX”) and a flux repartition device (pyrolytic graphite coating). The effects of irradiation have been investigated as the loss of conductivity, the significant swelling for orthotropic graphite and the enhancement of the Young modulus. The effect of the loss of conductivity on irradiated graphite has been quantified with steady state temperature computations. It appears that the temperature distribution through the structural material is not affected by the loss of conductivity. The main modification of the temperature distribution with irradiated properties is an average enhancement of 300°C in the tungsten plasma facing component.

Concerning **socio-economic studies**, the opinions and attitudes of the population with regard to future energy, fusion, research in fusion and the siting of ITER have been studied via meetings with various groups of people in the region near Cadarache (different focus groups as “general public”, “associations”, “local authorities”, “emergency workers”). It has been demonstrated once again that training and communication are important elements in the process of social acceptability.

**Related tasks in the full report**: TW1-TRP-PPCS1-D02b, TW1-TRP-PPCS2-D02b, TW2-TRP-PPCS15-D03, TW1-TRE/FPOA
## INTRODUCTION

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**Breeding Blanket**

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