Annual Report of the Association EURATOM-CEA
2005
(Executive summary)
Cover: Most of the ITER diagnostics systems are to be integrated in port plugs, which are water-cooled stainless steel structures inserted into the vacuum-vessel ports. The port plug must perform basic functions such as providing neutron and gamma shielding, supporting the first wall armour and shielding blanket material, closing the vacuum vessel ports, supporting the diagnostic equipment. CEA has contributed to the engineering activities (including CAD effort, structural and thermal analyses) on the port plugs and has more particularly focused on the design and diagnostic integration in the representative equatorial port plug EQ01.
This document is the executive summary of the full annual report which summarizes activities performed by the EURATOM-CEA Association in 2005 within 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, which are summarized in an annual report issued at the EU Commission level.

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 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 2005 within the frame of the European Technology Programme (both “EFDA” activities and “Underlying Technology” programme).

Three 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 and magnet and plasma facing components activities.

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

Progress in fusion technology is constant over the years and this report once again highlights a number of important steps that have been accomplished in this domain. 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. On the longer term, progress in technology will improve the vision of an electricity producing reactor and will 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.

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Physics Integration activities

Euratom-CEA activities carried out in the field “Physics Integration” are mainly linked, to heating and current drive, especially on the Ion Cyclotron Range of Frequencies (ICRF) Antenna developments, and to the development of diagnostic components.

For heating and current drive, the European concept for a 1 MeV, 40 A negative ion based accelerator for the neutral beam system on ITER, the SINGAP, SINGLE Aperture (SINGAP), is an attractive alternative to the ITER reference design, the so-called MMuG (Multi-Aperture, Multi-Grid) accelerator. A prototype SINGAP accelerator has been used for several years and produced D- beams of 910 keV, 30 A/m². The measured beam profiles on the target agreed well with those predicted by calculations. However certain design features of the prototype accelerator prevented the production of beams with the optical quality required for ITER [figure 1], i.e. a beamlet divergence of ≤7 mrad and beamlet aiming within ±2 mrad of that specified. Therefore a new “ITER-like” accelerator has been designed and built in order to demonstrate that the beam optics required for ITER can be achieved with a SINGAP accelerator.

The experiments have so far confirmed some aspects of the design of the new “ITER-like” accelerator, but not all. In particular the experimental data show that the beamlets have a bi-Gaussian power density distribution (70% of the power can be described by a beamlet divergence of ~4-5 mrad and 30% is in a halo) as opposed to the single Gaussian with 2.5 mrad divergence of the simulation. The fraction of the total power that is seen as a halo varies between 15% while operating at low current densities without Cs to 30% during cesiated high current density operation.

Concerning the activities of the Association on the Ion Cyclotron Range of Frequencies (ICRF) Antenna, one of the objectives was to propose an update of the ITER ICRH system Reference Design, and number of changes in the ITER FDR ICH design have been proposed [figure 2], to simplify the system and to improve its reliability keeping unchanged the design concept and interfaces. Issues not sufficiently covered in the FD report were also addressed. At present stage this design have to be refine from an engineering point of view. This task implying a more detailed mechanical description of the array including support and cooling, completed by a more accurate EM analysis and a detail thermal analysis will be
pursued in the coming year. In the meantime the development of ad-hoc components is pursued. The main element to develop is the compact tuning system; it is planned within two years to design, build and experimentally demonstrate the performances of this component.

Another part of this activity to be covered during the next two years is to design, build and test a Compact Vacuum Tuner (CVT) compatible for an integration in the ITER Ion Cyclotron Heating and Current Drive launcher. This high power tuning device is designed to fulfil ITER in-vessel EM, mechanical, thermal, nuclear and Remote Handling specifications (RH). In the ITER array two CVTs would be combined in a two-strap “ITER-like structure” (ILS) which is the basic element of the ITER IC array and features a significant resilience to load variations, such as those due to ELMs. The device, however, can be used for general high power cw impedance matching applications [figure 3].
Concerning the part dedicated to diagnostic development, Euratom-CEA has advanced the design of several ITER diagnostic systems for which the EU has developed conceptual designs, to re-evaluate their performance for the most recent analysis of plasma conditions, and to provide support to the ITER IT in the preparation of the relevant ITER documentation. Among others, studies have been performed in 2005 on bolometer cameras, wide-angle viewing/thermography system, calorimetric measurements, Motional Stark Effect (MSE) and reflectometry system.

The Association has also been involved in diagnostic design for ITER, especially concerning port integration and magnetic diagnostics.

On **port integration**, 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. Although the required analysis was associated with the port plug where Europe and more particularly Euratom-CEA may have a certain responsibility in term of diagnostics, Euratom-CEA has so focused on the design and diagnostic integration in the generic equatorial port plug EQ01. The specific contributions were to perform, associated to this port, general engineering (including CAD effort), structural and thermal analysis [figure 4].

![Figure 4: 3D view of the equatorial port plug EQ01 & Displacements of the equatorial port (disruption)](image)

On **magnetic diagnostics**, three kinds of ITER magnetic sensors have been reviewed. Manufacturing procedure and recommendations have been made. In particular, the ex-vessel tangential coils exhibit the more severe constraints: the available radial dimension is very small (7 mm), the requested area is rather big (2 m²) and there exists a big discrepancy in their dimensions.

The design of the external Rogowski coil has been refined [figure 5]. In particular, a former on which grooves making a double screw is proposed to ensure a regular winding. A model has been produced. The path in the TFC casing has also been defined in collaboration with the ITER IT.

![Figure 5: Example of grooves in a Rogowski for uniform winding port (disruption) & View of the Rogowski route on the TFC casing](image)
The inner vessel tangential and partial flux loops set-up is very simple, because it is made of MI cable arranged around the torus or making saddle loops. The work done highlighted the special attention that must be devoted to the mounting, because of big radial dimensions and the obligation to define a horizontal plane in order to measure only the vertical flux. During the plasma, the vessel thermal expansion could damage the loop. Therefore it is recommended to leave an extra length of cable at some places. In general, the vessel thermal expansion effects have been highlighted as a potential source of error. A model of the vacuum vessel representing its thermal behaviour must be developed. In the same way, the eddy current in the structure must be modelled in a future work.

**Related tasks in the full report:**
CEFDA01-645, CEFDA03-1044, CEFDA03-1129, CEFDA04-1140, CEFDA04-1146, CEFDA04-1180, CEFDA04-1182, EFDA05-1271, TW5-TPHI-ICRFDEV, CEFDA02-1003, CEFDA03-1111, CEFDA04-1206, TW5-TPDC-IRRCEER-D03,

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Vessel/In Vessel activities

The final construction of the ITER vacuum vessel will require a multipass welding operation to fully penetrate the entire 60 mm thickness. The ITER welding reference process is the industrially proven (but slow) multi-pass narrow gap TIG method (NG TIG).

For the filling passes, the investigated MIG/MAG + YAG process promises to increase productivity further. Welding tests performed concern hybrid YAG + MIG / MAG welding process to fill 20, 35 and 50 mm depth and 5.5 mm width grooves [figure 1]. Filling demonstration has been assessed for such grooves with the hybrid process, in the flat welding position. Sound welds have been realized with no welding defects. The deposit rate reached is about 4.5 kg/h.

These results have been obtained by optimising the welding torch (contact tubes with a geometry adapted to the grooves and a ceramic protection) and in developing new gas protection devices (to fill in the groove). The measured side wall penetration is about 350 µm, less than the 0.5 to 1 mm for the Narrow Gap TIG Welding process. But the robustness and stiffness achieved make this process useful to industrial applications.

Another field studied concerns the building-up and repair of ITER vacuum vessel which will require the machining of 60 mm thickness steel walls by a cutter saw. The slicing machining system will be coupled with the IWR robot, moving inside the vacuum vessel on dedicated rails. To operate cutting process, a mechanical system has been designed and built [figure 2], able to reach a 360 mm.s⁻¹ machining speed while cutting a 10 cm thickness of stainless steel, leading to 6 passes for cutting the 60 mm required.

Figure 1: Deposits in 50mm deep grooves

Figure 2: Design of machining tool & Used saw
In the context of the ITER vacuum vessel inspection, studies were performed to investigate ultrasonic techniques which are relevant for this inspection. New improved techniques were required to get better the whole inspection of the weld thickness. In particular, the recent tasks were related to the design of phased array probes using simulation and the experimental arrangement for most relevant configurations of the phased array probes. 

The Euratom-CEA Association has pursued the analysis of the effect of the TIG joint weld structure related to the UT inspection using phased array techniques. In particular, a parametric analysis has been carried out using ultrasonic simulation software. This has been achieved using the CIVA software for the simulation and experimental ultrasonic arrangement [figure 3].

![Image](image_url)

**Figure 3:** shows the simulation for 10 mm height inner surface breaking notch and it is compared with the experimental results.

In this framework, a collaborative work has been launched in order to assess three UT methods developed by PHOENIX(UK), SINE/T ECHO (Russian Federation) and CEA (Saclay) for acceptance by the RCC-MR manufacturing code and the French Certification Authorities so that they may be used successfully during the ITER vacuum vessel manufacture.

The properties of materials used for fusion components need to be known in detail by designers, by licensing authorities and by materials specialists. The Association Euratom-CEA’s contribution to Materials Properties Handbook, in addition to supporting ITER IT, EFDA CSU, and ITER database groups, is to provide MPH files for Type 316L(N) steel weld metals and joints. In 2005, the work was focussed on low temperature (316L) and high temperature (19-12-2, OKR3U) weld metals. In 2006 work will be extended to 16-8-2 weld metal.

Manufacturing of the ITER Primary First Wall (PFW) panel by HIP forming is investigated. The Association contributes to this task by several actions, including the improvement of the knowledge on the strength of CuCrZr/SS junctions after different manufacturing conditions foreseen for the manufacturing of first wall panels and at defining an acceptance criteria for the assembly. In the meantime, investigations on
the properties of the CuCrZr itself after the different manufacturing conditions have also been intended. Mock-ups have been realised, by assembling stainless steel tubes inserted between two CuCrZr plates on a thick stainless steel plate, and will be examined by VTT (Finland) [figure 4].

In 2005, the Association has also launched studies on creep-fatigue behaviour of CuCrZr. Some data from the literature show that the linear damage cumulative law is not respected for CuCrZr material.

On the other hand, Euratom-CEA proposed to work in collaboration with CNRS-CRETA to elaborate and characterise two existing compositions, Cu-13wt.%Sn-2wt.%Ni-6wt.%In (=STEMET 1108) and Cu-30wt.%Sn-9wt.%Mn-1 wt.%Ti, the objective being to manufacture and characterize Cu-based braze alloys foils suitable for Be/CuCrZr induction brazing, and to try to avoid the problems encountered with STEMET 1108 ribbons from MIFI-AMETO and identified by EFDA, such as variations of compositions among supplied batches, ageing and too high liquidus temperature.

Related tasks in the full report:
CEFDA04-1202, CEFDA05-1226, TW2-TVV-ROBOT, TW3-TVM-JOINT, TW4-TVV-HYBRID, TW4-TVV-OSEWELD, TW5-TVM-BRAZÆ, TW5-TVM-COMADA, TW5-TVV-MPUT, UT-VIV/VV-Hybrid-Modelli.

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Plasma Facing Components activities

A proposed route to a valid conception of first wall mock-ups implies diffusion welding of Be tiles onto a Cu alloy, but it is difficult to apply traditional analysis techniques to Be tracking because 1) of its toxicity and 2) of its low atomic number Z. Ion Beam Analysis (IBA) has thus been considered, and in particular, Nuclear Reaction Analysis (NRA) and Particle Induced X-ray Emission (PIXE) techniques.

The objective of the work performed in 2005 by the Association was to demonstrate how such techniques can bring valuable information in such investigations, and describe the Be//Cu junction [Figure 1] related to manufacturing parameters of reference mock-ups. Up to now, 10 Be//Cu junctions have been analysed using NRA/PIXE analyses. The results presented above show that these techniques are nicely adapted to this study. However, other specific techniques should be available in 2006 for this study. In particular, field emission gun-equipped scanning electron microscope, coupled with the Electron Back-Scattered Diffraction (EBSD) technique, should allow to select the most complex cases for the heavier IBA techniques, complete the IBA results, and finally allow a precise determination of the composition of the Be//Cu junctions.

The Association Euratom-CEA has also pursued its effort on demonstrating the applicability of the Hot Isostatic Pressing (HIP) technologies for manufacturing Primary First Wall panels made from a combination of copper alloy (as heat sink material) a stainless steel (as structural material) and beryllium tiles (as an armour material). Up to now, several mock-ups and panels were manufactured and thermal fatigue tested. The Be armour material was joined onto the copper heat sink material by hipping or brazing. The objective of this task was to perform a development work programme to optimise the Be/CuCrZr alloy HIP joining conditions and to manufacture FW mock-ups for high heat flux and thermal fatigue testing. In 2005 the development work to further improve the Be/CuCrZr HIP joint performance has been completed.

As thermo-mechanical fatigue is one of the most important damaging mechanisms for the plasma facing components (PFC) of the ITER machine due to the high number of operating cycles (several thousands) and to the expected surface heat loads, an assessment of the behaviour of PFC under cycling heat loads is essential to demonstrate the fitness for purpose of the selected design solutions. Euratom-CEA has also been involved in the monitoring and analysis of thermal fatigue testing of PFC to be performed in the frame of the European R&D programme for ITER. Three mock-ups were thermal fatigue tested at FE200 in 2005: CuCrZr/SS first wall mock-ups [figure 2], W armoured monoblocks and CFC armoured monoblocks.
Among all Non-Destructive Examinations, active infrared thermography by internal thermal excitation is becoming recognized as a technique available today for improving quality control of many materials and structures involved in heat transfer. The infrared thermography allows to characterize the brazing bond between two materials with different thermal physical properties. 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. In order to increase the defect detection limit of the SATIR test bed [figure 3], several possibilities have been assessed. SATIR was partially upgraded (preparation) in 2005 by increasing the inlet pipe diameter and by improving the heating unit. SATIR upgrade (construction) planned in 2006, phase 2 is 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.

Carbon Fiber Composites (CFCs) are considered as an attractive choice for high heat flux components in existing and forthcoming tokamaks such as ITER. The Association has been involved in 2005, in the development of a new process for joining CFC (carbon fiber reinforced carbon composite) to the Cu compliant layer in view of the fabrication of divertor components. After an analysis of the status of art, evaluation of a new solution for CFC / Cu joining [figure 4].
The Association has also worked on two CFCs that are particularly interesting for fusion devices: NB31 from SNECMA which is a 3D CFC constituted by a NOVOLTEX perform, with P55 ex-pitch fibers in the high thermal conductivity direction and NS31 which is a Si-doped 3D N31CFC.

In one hand, the aim was to study dimensional changes [figure 5], density, specific heat capacity and thermal conductivity changes of NB31 and NS31 irradiated in the PARIDE (PlAsma facing materRials for ITER and DEMO) 3 and 4 irradiations, performed in HFR/Petten at 250/260°C with two damage levels (0.24 and 0.83 dpa.g). Moreover the activity of the different radionuclides contained in these two CFCs has been measured. In the other hand, the study of the dimensional, density and thermal conductivity changes of the same CFCs grades irradiated in the RBT-6 nuclear reactor (Dimitrovgrad, Russia) at 90°C with damage levels ranging from 0.002 to 0.13 dpa.g has been carried out.

![Figure 5: NB31 & NS31 dimensional changes after irradiation at 90°C](image)

**Related tasks in the full report:**
CEFDA02-583, CEFDA03-1029, CEFDA04-1138, CEFDA04-1218, CEFDA05-1243, CEFDA05-1248, CEFDA05-1257, CEFDA05-1261, J W5-AEP-CEA-26, TW1-TVP-CFC1, TW5-TVD-C/rfc, UT-VIV/PFC-Damage, UT-VIV/PFC-HIP, UT-VIV/PFC-NanoSic, UT-VIV/PFC-Pyro.

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Remote Handling activities

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 tools for ITER or the radiation tolerance assessment of required electronic components in a fusion environment.

The Euratom-CEA Association performs a R&D program to demonstrate the feasibility of **close inspection of the ITER Divertor and first wall**. The work performed includes the design, manufacture and testing of an articulated multi-purpose tool demonstrator called Articulated Inspection Arm (AIA) [figure 1]. 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).

The feasibility of this carrier has been assessed in 2002. A scale one mock-up was manufactured and tested in air and at room temperature (2003). In 2004, a single module prototype was manufactured, tested in air and at room temperature and finally tested under ITER relevant conditions at Tore Supra.

In 2005, manufacture of a vacuum and temperature module demonstrator was tested in a representative module of Tore Supra called ME60, under temperature and pressure constraints. Promising results were obtained in term of structural resistance of the system. The past year was dedicated to the segment upgrade [figure 2] and to cycling test campaign to validate all the robot components.

The successful results enable to start the whole robot manufacture and procurement.

The manufacture of the final robot is under progress. The assembly is planned for the end of 2006. The AIA storage cask, the deployer and the electronic external bay will be delivered during 2006.

**Figure 1**: AIA deployment in ITER vacuum vessel and design of a single module

**Figure 2**: New elevation jack, new rods and motor cooler
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.

Hydraulic technology can provide powerful actuators in small volumes. For that reason hydraulics becomes an interesting technology to build heavy duty manipulators for maintenance operations in space constrained areas.

Due to potential leaks, oil hydraulic can not be used for maintenance operation in ITER. Pure water hydraulics proposes a good alternative to oil and today’s developments are focusing on that direction.

Previous work focused on the test preparation of a SAMM oil hydraulic vane actuator that was adapted to operate with water [figure 3]. Materials were changed and new coatings were used to check their compatibility with water. A test rig was designed and built and performances of the joint were assessed. Analyses of the test results and actuator’s mechanical state were made.

Due to significant internal leaks in the actuator, performance evaluation was difficult. Although performances didn’t give the expected results and leaks were high, no real corrosion problems were noticed, meaning that the material choice was correct at least for simple testing. Wear analysis after endurance test would be necessary to confirm the option of all selected coatings during the design phase.

It seems difficult to define the exact values of the clearances required in the joint to reach the expected performances. Today among other vane actuators, the Maestro joint seems to be in best position to reach ITER’s requirements.

Next test phase will therefore concentrate on a performance analysis of a reconditioned Maestro joint to define a start point without making any modifications of the existing design. In a second phase, an analysis of the results and proposals to modify the design will be made.

Figure 3: Test Rig

The radiation tolerance assessment of electronic components 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.

Works done from end of 2004 to end of 2005 led to unexpected results. Most of the modules designed and realized as part of functional blocks gave partial interesting results. Moreover, it was expected from industrial printed cards enough quality from signals to validate with a good accuracy the conversion of measures coming from both resolver and LVDT sensors. Unforeseen events during the manufacturing affected the normal use of the cards. Debugging and recovering of the main functions and radiation scheduling became no so easy to drive.
A new resolver card, now realized and validated, is ready for further experiment. The final testbed has also to been realized in order to give some accuracy to the converted resolver measures [figure 4]. LVDT experiments could be extended with accuracy determination.

Figure 4: left - New prototype of LVDT multiplexor
right - Test bed for LVDT measurements

Related tasks in the full report:
TW5-TVR-AIA, TW4-TVR-Radtol, TW4-TVR-WHMAN, TW5-TVR-RADTOL, TW5-TVR-WHMAN,

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Magnets structures and Integration activities

In 2005, the Euratom-CEA Association was involved in the detailed design and manufacture of relevant mock-ups [figure 1] for some critical areas of the Toroidal Field (TF), Central Solenoid (CS) and Poloidal Field (PF) coil windings. Mechanical testing at cryogenic temperatures of the mock-ups under relevant loads and number of cycles will be carried out at FZK Karlsruhe and ENEA Brasimone. The design of the mock-ups has been performed in close collaboration with these two Groups and EFDA/ITER, coordinate the testing activity and report on the final test results. Association Euratom-CEA is responsible for the definition of the testing conditions (loads, number of cycles, temperature, etc.) under review and approval of EFDA/ITER.

The TF inlets, CS inlet and PF tail mock-ups have been manufactured following the design analysis, and tests have been performed. The hydraulic tests of CS inlet should be completed in 2006.

The development and test of a new generation of 'high performance' Nb$_3$Sn superconducting strands [figure 2] has been pursued by the Association Euratom-CEA, by a strong involvement in several studies, including among others the test of six companies strands (Alstom (F), Outokumpu Italy (I), Outokumpu Finland (FIN), EAS (D), SMI (NL), Oxford Instruments (GB)) products by cross measurements of six European laboratories. In 2004, the benchmarking step and a partial characterization of the strand provided by Oxford Instruments (OST) were performed (see 2004 activities report). In 2005, the remaining parts of the program were completed. After comparison with EFDA specifications, the only OST strand was found to meet the totality of EFDA specifications. The OKSC strand deviation from specifications was found in the AC losses (+25%), but this could be due to an anomaly of HT. For the OCSI strand a slight deviation was found for IC, but transport properties still remain in agreement with ITER specifications in terms of critical current density.
The investigation of the **bending strain effect on Nb$_3$Sn strand performances** has been pursued and a bending application method (BEAM) has been developed and validated. Basically the BEAM includes the samples preparation and positioning on the HT mandrel and the strand transfer onto the measuring mandrel after HT.

The main tool used is a home-made one [figure 3]; it is used to maintain the strand during the transfer and soldering stages, requiring a specific caution.

The exploration of the **sensitivity of high performance Nb$_3$Sn strands to stainless steel jacketing** on sub-size samples, regarding the critical properties has also been pursued, as the Association is in charge of the production of about fifty sub-size Nb$_3$Sn conductor samples. These samples are of three different sizes corresponding to various stages of the cabling pattern of a ITER TF conductor petal.

In particular the samples to be produced, by increasing size are of the type (3x3), (3x3x5), (3x3x5x4) [figure 4].

In February 2005, a contract for the cabling and jacketing of the samples has been awarded to the company Nexans. Prototypes were required to confirm the capability of NEXANS to produce samples according to specifications.

Controls have been performed to confirm that the samples comply with the required specifications.

The Association Euratom-CEA has also launched in 2005 a study in order to perform **cryogenic tests at 4 K to characterize the ITER magnet structural materials** and to prepare European laboratories for the large number of tests necessary during the ITER magnet procurement action. This task is performed in collaboration with FzK for the definition of the standard procedures to be followed during cryogenic tests at 4 K, 77 K and room temperature.

Air Liquide / DTA and CEA / SBT have been collaborated for several years. Air Liquide / DTA is in charge of Mechanical tests (Young modulus, tensile test, compression test, materials fatigue ...) thermal tests are performed at CEA / SBT (thermal expansion, thermal conductivity ...).
From a technical point of view, the work is separated in two parts, the first one concerns the upgrade of the tests bench to increase the quality of the measurement and to standardize the measurement procedure.

In the second part, the same tests have been carried out at FzK laboratory and CEA/SBT or Air Liquide/DTA then the results have been compared.

In 2005, activities concerned mechanical and thermal tests using Instron tensile machine [figure 5]. All the thermal tests have been performed; the mechanical tests will be finished in 2006.

**Related tasks in the full report:**
CEFDA03-1015, CEFDA03-1098, CEFDA03-1120, CEFDA04-1127, CEFDA04-1134, CEFDA04-1170, CEFDA04-1201, CEFDA04-1215, CEFDA04-1219, TW1-TMS-PFC ITE, TW5-TMS-HTSMAG, TW5-TMSF-HTSPE.

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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.

On TBM accidental safety study two accidental conditions have been assessed:
- firstly, HCLL-TBM behaviour under accidental pressurization
- secondly, HCLL-TBM behaviour under Loss Of Flow Accident (LOFA)

In 2004, the HCLL-TBM behaviour under Loss Of Flow Accident (LOFA) operating conditions was assessed in one of its worst scenarios, LOFA with active plasma shutdown after delayed accident detection with disruption. In 2005, the study was completed by the assessment of LOFA with no active plasma shut-down and showed that the scenario needs to be better analysed, in particular with regard to the requirements in term of active plasma shutdown. Future work will consist in calculating more precisely the available delay before non-respect of criteria and in checking the capability of the existing protection systems to be activated in this delay. The beryllium layer on the FW face in front of plasma will be also added.

The development of fabrication processes for test blanket modules (TBM)s subcomponents has been pursued. The TBM_s are composed of the following subcomponents: first wall, stiffening plates, caps and breeder units. All these components are cooled with helium thanks to embedded channels. The structural material is a reduced activation ferritic martensitic steel (Eurofer). The complexity of the cooling scheme and the small channel dimensions result in manufacturing difficulties. As several processes are envisaged, potentially applicable processes for the manufacturing of TBM subcomponents have been identified and investigated in 2004. In 2005, this investigation was completed and several recommendations on fabrication processes were made.

The joining technology development, fabrication and testing of mock-ups to qualify manufacturing technologies for TBM assembly has been launched in 2005. The demonstration of the integration of manufacturing technologies will be assessed through the fabrication of a TBM demonstrator.

For TBM manufacturing, three stages must be developed: Stiffening joining, Stiffening/First Wall joining, and Stiffening and First Wall/Cap joining. For this study, first laser welding results in real joint configuration show a laser hot cracking sensitivity [figure 1] in partial penetration depth, observed on other welding processes, such as TIG and Electron Beam, but not in full penetration depth. The first 2006 trimester will be dedicated to solve this problem. Several potential solutions let think positive technical possibilities.

![Figure 1: hot cracking in fusion zone](image)
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 Li$_2$TiO$_3$ fabrication process, and, regarding to its experience in this field, in the investigation on blanket manufacturing techniques of the first wall by Hot Isostatic Pressing.

One interesting advantage of this option is that the tubes and the plates can, in principle, be bent before HIP. During this work, the feasibility of the manufacturing of blanket first wall using rectangular tubes was assessed. In a first step, a route consisting in bending tubes [figure 2] was tried. It appeared that it is not possible to recover the bending deformation upon HIP. Then, an alternative route was studied which consists in building the channels by welding pieces. The laser welding process was used though it was never tried before on Eurofer. Straight parts and bends were welded successfully. The quality of the welds was sufficient for HIP: a short straight mock up was manufactured successfully. On the opposite, butt welds were very brittle and their tightness could not be maintained during the fabrication of a U-shape mock-up. The manufacturing of this mock-up aborted after two repairs. Since the bending route is not satisfactory, the limiting factor as regards the development of the rectangular tube process is the manufacturing of the tubes. Laser welding appeared to be the best solution, but more work is required to develop this process which is not yet mature enough for this application.

**Related tasks in the full report:** TW2-TTBB-002b-D01, TW5-TTBB-006-D04, TW2-TTBC-001-D01, TW2-TTBC-002-D01, TW2-TTBC-002-D02, TW2-TTBC-002-D03, TW2-TTBC-003-D05, TW2-TTBC-005-D01, TW2-TTBC-005-D02, TW4-TTBC-001-D01, TW5-TTBC-001-D01, TW5-TTBC-001-D02, TW5-TTBC-001-D03, TW5-TTBC-001-D06, TW5-TTBC-001-D07, TW5-TTBC-002-D01, TW5-TTBC-005-D01, 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**.

Previous studies have shown that the milling step under argon for the elaboration of the ODS steel reduces the toughness of these materials. The carbides alignments created by the milling are probably one of the reasons of such embrittlement. The impact properties of the milled & HIPed ODS steels can be improved if thermo-mechanical treatments are applied but these ones are not always compatible for near-net shaped applications.

In order to solve the toughness problem, one could perform mechanical alloying under high purity controlled atmosphere (H₂). Hydrogen would reduce both the oxide and carbide formation at the surfaces of the milled powders. However, this solution requires heavy equipment (for safety reasons). Another and more radical solution is to completely avoid the mechanical alloying step. This solution would also make the elaboration process simpler. In that aim, the Association has studied the **elaboration an ODS steel directly by internal oxidation** without milling.

Unfortunately, this initial aim could not be reached because of an intrinsic and unexpected problem at the first step: reactivity of Yttrium during atomisation [figure 1].

From the experience acquired, two points are important to produce Y-supersaturated Eurofer powders which could be used to elaborate of ODS steel by internal oxidation:

- one has to use only yttrium, without any oxygen addition, because Y and O would form an \( \text{Y}_2\text{O}_3 \) skin at the surface of the melt,
- the \( \text{ZrO}_2 \) and \( \text{Al}_2\text{O}_3 \) parts of the equipments should be protected, because Y would react with them. One solution could be to apply a \( \text{Y}_2\text{O}_3 \) protective layer to these parts that could be deposited by Chemical Vapor Deposition (CVD), by slurry coating or by plasma spraying.

The Association has also pursued its effort to propose material properties allowable for design and licensing of components fabricated with the Eurofer steel. In 2005, the database of Eurofer steel was updated, particularly with the RAFM data and analysis resulting from the work done at CEA and NRG.

With the addition of the new Eurofer steel data in 2005, the collection of relational databases for RAFM steels contains:

- **Products database**: 571 records including 118 on Eurofer
- **Compositions database**: 475 records including 26 on Eurofer
- **Tensile database**: 1185 records including 258 on Eurofer
- **Impact database**: 1710 records including 467 on Eurofer
- Impact plots: 161 records including 45 on Eurofer
- Creep database: 205 records including 81 on Eurofer
- Fatigue database: 232 records including 70 on Eurofer
- Fracture toughness database: 261 records, mostly for Eurofer steel, including 8 Master Curves.
- A few fatigue crack propagation test results
- Summary records for all databases

The qualification of fabrication processes is an important part of the Euratom-CEA Association on Structural Material activity. Among others, as fusion reactor blankets are structures cooled by internal channels using reduced activation ferritic-martensitic steel plates and tubes, a study has been pursued in 2005, in order to define and qualify suitable process for the fabrication of the tubes, with a particular emphasis on rectangular tubes that might be used for the fabrication of the first wall of the modules.

At the beginning of the study, rectangular tubes made of T91 steel were manufactured by hot extrusion and expertised [figure 2]. It was shown that tubes had a poor dimensional accuracy, but both the material microstructure and its mechanical properties were satisfactory, despite a rather large grain size. Cold drawing extruded tubes appeared a promising solution in terms of dimensional accuracy improvement and grain refining. This solution was tried in 2005.

The combination of hot extrusion and cold drawing is a promising way to achieve rectangular tubes suitable for the manufacturing of first wall channels. The microstructure and the mechanical properties of the material are satisfactory. Nevertheless, process improvements are required as regards to the surface condition, the dimensional accuracy and the corner radius. There are indeed some possibilities for improvement, such like the use of a tensile pulling to get rid of twist and bow defects, the use of neutral atmosphere for heat treatment and basic precautions to avoid cracking.

Another important aspect of fabrication processes study concerns dissimilar welds development for joining TBM Eurofer tubes with external branch pipes in SS316 steel (cooling channels, PbLi loop), with the aim to characterize laser welding joints of TBM tubes in butt joint design. The existing laser tool will be adapted to the application, respecting the welding requirements. CAO design of TBM tubes and CEA tools are running out to simulate welding sequence [figure 3]. Design modifications of the laser tool will allow to process TBM tube welding. Final version and clamping devices are expected for may 2006.
Advanced structural materials such as SiC-SiC ceramic composites are also studied by the Association. One of the studies performed is the modelling of the mechanical behaviour of advanced 3D SiC/SiC composite. In 2005 the implementation in CAST3M of two new constitutive laws adapted to SiC/SiC composite have been done. These laws were developed at ONERA (Office National d’Etudes et de Recherches Aérospatiale). These new models, with convenient material parameters, obtained from 0° and 45° traction compression tests on woven SiC/SiC, allow to reproduce the experimental macroscopic behaviour of this composite. Comparison of CAST3M and ZeBuLoN (ONERA finite element code) results were performed [figure 4].

In the next years, implementation in CAST3M of multiscale modelling adapted to SiC/SiC will be continued.

![Image: Comparison CAST3M & ZeBuLoN](image)

**Figure 4: Comparison CAST3M & ZeBuLoN**

**Related tasks in the full report:** TW2-TMMS-001b-D02, TW2-TMMS-004b-D01, TW4-TMMS-006-D03, TW4-TMMS-007-D02, TW5-TMMS-004-D02, TW5-TMMS-004-D04, TW5-TMMS-004-D06, TW5-TMMS-004-D07, TW5-TMMS-005-D01, TW5-TMMS-006-D01, TW5-TMMS-007-D04, TW5-TMMS-007-D21, TW3-TMMA-001-D04, TW3-TMMA-002-D04, TW5-TMMA-001-D08, UT-TBM/MAT-LAM/Opiti, UT-TBM/MAT-MICRO, UT-TBM/MAT-Modpulse.

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Safety and Environment activities

The computer codes which are used for the analysis of the accidental sequences in ITER should have good quality assurance level. The EVITA facility [figure 1] has been designed for the simulation of the physical phenomena occurring during a coolant ingress into the cryostat of ITER reactor, which is one of the identified accidental sequences in the ITER safety report. Studied physical phenomena are namely ice formation on a cryogenic structure, heat transfer coefficient between walls and fluid, flashing, two-phase critical flow. The comparison between calculations and experiments allows the ability of the computer codes to treat the relevant physical phenomena to be assessed.

In 2005, the main experimental results are the pressure evolution in the vacuum vessel, the different heat exchanges and the ice formation on the cryogenic surface.

To achieve the experimental program, ten tests have been scheduled on the EVITA facility this year. The results of these complementary tests have improved the knowledge of the formation of ice layers and the condensation of water during injections of steam or pressurized water.

In association to experiments performed in EVITA facility, PAXITR2 is a spot calculation computer code used to evaluate the pressurization or the depressurization of containment. In 2005, the Association has also pursued the validation of cryogenic calculations that have been performed on the EVITA experimental facility in the frame of the validation and qualification of the PAX ITR 2 code used in fusion safety assessment.

Due to inherent PAX ITR limitations (spot calculation, steam and water in thermal equilibrium), non-equilibrium phenomena ($P > P_{\text{sat}}(T_m)$) cannot be calculated.

Moreover, the experimental vacuum vessel pressure evolutions are different even if the incoming flow rates are almost the same. The tests analysis has shown that this could be due to some liquid droplets mixed with the steam injection. This phenomenon cannot, of course, be simulated with PAX ITR code. The tests are supposed to be performed in the same conditions but the reproducibility seems to be difficult to obtain [figure 2].

The test analysis should go further to identify the tests to be performed again. The actual tests are still not sufficient to qualify a computer code on this kind of transients.

In the framework of fusion waste management, detritiation processes could allow the waste level to be reduced. Studies have been pursued to determine different
processes that could be used for tritium removal from housekeeping materials (mainly plastics), vacuum oil and organic liquids.

In 2002-2004, the Heat Transfer and Fluid Mechanics (LTMF) laboratory of CEA Nuclear Energy Division in Saclay performed H₂ risk related studies for the ITER vacuum vessel. Numerical simulations were performed with the CAST3M code and code to code comparisons were made with the GASFLOW and COM3D/DETi3D codes of FZK. In 2005, no new numerical simulations were performed but instead, a synthesis of the performed numerical work was made, focussing on the development and validation needs for the CAST3M code, with respect to hydrogen distribution and combustion, dust mobilization and explosions. Finally, with the realization that unmitigated accidents pose unacceptable risks, a thorough bibliographic review of inerting mitigation techniques was made, together with a proposal for performing experiments in the MISTRAL facility of CEA Saclay [figure 3] to evaluate the effectiveness of the method to inert hydrogen clouds, as well as to prevent dust explosions.

Finally, the status of CAST3M code development and validation with respect to hydrogen and dust mobilization and explosions was made. Flow models developed for hydrogen risk in fission reactors as well as the associated validation efforts were described. A synthesis of the simulation work performed for ITER accident scenarios was also made, identifying the specific development and validation needs that are required, for both hydrogen distribution and combustion, as well as dust mobilization and explosion.

In 2006, modelling work in the area of dust mobilization will be performed, with the development of a two-fluid (gas and dust particles) model, and validation on appropriate experiments such as FZK’s dust tube and ENEA’s STARDUST experiments. This work will be performed in close collaboration with FZK.

In fusion devices such as the International Thermonuclear Experimental Reactor, ITER, neutron activation would also produce **Activated Corrosion Products** in the divertor, blanket and vacuum-vessel cooling loops, as well as in any other auxiliary cooling systems. These corrosion products are basically released from Cu-alloy (as CuCrZr)
constituting part of the piping under flux in the divertor and from SS 316L (N) IG constituting the first wall/shield blanket. These ACP could be responsible for a large percentage of the Occupational Radiation Exposure (ORE) of personnel during the reactor operation, inspection and maintenance. By consequences, the precise determination of ACP inventories and the estimation of the resulting doses to personnel is thus an important safety task.

The computer code PACTITER has been used for the calculation of generation and transport of ACP in the various Primary Heat Transfer System (PHTS) or Tokamak Water Cooling System (TCWS). In the last version, PACTITER V3, the release rates of the different materials in contact with the cooling water are input data of the code. Thus they have to be accurately determined in various representative ITER operating conditions prior to any calculation.

Corele loop [figure 5] devoted to the measurement of the release of industrial tube belongs to the test facilities used to validate PACTITER code.

The tests campaign has underlined an important temperature effect: the total release rate is higher at 150°C than at 200°C and negligible at 100°C. Last test 2005-01 has pointed out the major role of the first days (<300 hours) on the release rate process.

These effects have been interpreted as the result of the competition between thermodynamics (chemistry coolant, pressure, temperature, material composition) and elementary diffusion (including H2O and/or O2). Depending on the magnitude of the thermodynamically equilibrated concentration of the released element in the bulk of fluid, the coolant velocity (and the hydrodynamic diffusion) may play a role in the release rate.

However such experiments have been conducted for a water chemistry, which could not match the final ITER specifications. In that aim it could be interesting to specify the release rates values using perfectly deionized water.

**Related tasks in the full report:**
TW3-TSS-LT4, TW4-TSW-002, TW5-TSS-SEA3.5, TW5-TSS-SEA5.3, TW5-TSS-SEA5.5, TW5-TSS-SEA5.6, JW5-FT-3.2, UT-S&E-LASER/DEC, UT-S&E-LiPbwater, UT-S&E-Tritium-Impact.

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System Study activities

In the roadmap of the development of fusion energy, ITER is the following major step. The main objective of ITER is to demonstrate prolonged fusion power production in deuterium-tritium plasma. Half way through the scientific exploitation of ITER, it is plan to begin the design of the demonstration fusion reactor, DEMO. Construction of DEMO should start when ITER is fully exploited.

The development of the technologies which will be required for this demonstrator have already started in the aim of testing some DEMO-relevant components inside ITER. The breeding and high-grade energy extraction blanket modules are some of these elements. Indeed, these components facing plasma are exposed to very high thermal loads, so it is necessary to replace them at given periods.

Due to the radiation level, it is necessary to make the operations fully by Remote Handling. This critical maintenance needs to be done in acceptable times in regard to the availability of the reactor. In that field, the Association has been launched in 2005 an activity concerning the Remote Handling of the blanket elements during the maintenance phases.

The objectives are to review the different possible segmentations to allow the selection of a reference concept for the forthcoming DEMO conceptual design for three reference design:

- Large modules [figure 1]
- Banana segments
- Multi-module-segments (MMS)

In the first phase of the work, we have reviewed the different Remote Handling Equipments usable in the fusion context. The main bases are the studies made for ITER and the solutions used in JET.

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 technology 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 an non-inductively driven current density profile. This task is a first attempt to analyze this problem by means of the integrated modeling code CRONOS. CRONOS is a suite of numerical codes for the predictive/interpretative simulation of a full tokamak discharge. It integrates, in a modular structure, a 1-D transport solver with general 2-D magnetic equilibria, several heat, particle and impurities transport models, as well as heat, particle, current and momentum sources.

The preliminary part of this study, consisting of the 0-D design and the development of important modules of the CRONOS suite of codes, has been completed in 2005.

Figure 1: Illustration of the Large Modules concept proposed by FZK
The code is now ready to be exploited to find a viable scenario of steady-state operation for DEMO. It should be stressed that the existence of such a scenario is not a priori guaranteed. Therefore, the 1-D modelling is an essential step for the first design phase of DEMO and of course also of the following commercial reactor.

Within the framework of the European Power Plant Conceptual Studies (PPCS), one of the reactor models, the model AB, is based on a Helium-Cooled Lithium-Lead blanket (HCLL) blanket [Figure 2]. A study has been performed by the Association in order to check the validity of the analyses performed for DEMO when extrapolated to the PPCS specifications and to assess the T-management and control in the HCLL blanket and associated systems. The maximum authorized T-release to the environment is assumed to be 27 Curies/day, equivalent to 1 g/year. However, this allowance will be for the whole plant, including other sources of tritium like reactor refuelling, divertor pumping etc. Even taking the whole allowance for the breeding blanket system, the tritium isolation ratio J1 / J5 is to be as high as 200,000, with the assumption of a reactor availability of 82% (300 operating days per year). The 2005 performed activities focused on the importance of taking into account the T transport in the Pb-17Li.

Figure 2: Detailed views of a HCLL blanket module and of a Breeding Unit

Related tasks in the full report:
CEFDA05-1285, TW4-TRP-002-D02b, TW5-TRP-002-D03a.

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ITER Site Preparation

The ITER site preparation is performed within the framework of the European ITER Site Studies (EISS). These studies have their own steering process with regular meetings and exchange of information with EFDA and the European Commission. The most important fact in 2005 was the announcement, the 28th of June in Moscow, by the ITER Project’s partners of the decision to build ITER in Cadarache. During the second semester of 2005, the President of the French Republic Mr Jacques Chirac, the European Commissioner for Science and Research Mr Janez Potocnik and the Director General of the future ITER Organisation Mr Kaname Ikeda visited Cadarache.

SAFETY & LICENSING:
As it was the case in previous years, the writing of the first version of the main technical document in support of the Safety Authority’s instruction, the “Rapport Préliminaire de Sûreté” (RPrS) was followed. The writing of this document by EFET is supported by many studies, performed at European level in parallel. All chapters have been finalised and the RPrS needs to be reviewed in 2006, following a wider project review. The R&D to be launch within the framework of EISS5 will also allow to complete and detail the writing of several chapters. The risk analysis studies and the DAC and DARPE specification have been continued. According to the French regulation and the 10th of August 1984’s Order, the Safety and Licensing process concerns these 3 main documents RPrS, DAC and DARPE. Several studies were performed to support these Safety and Licensing area, for example the aspects concerning an ITER waste management strategy, in particular for tritiated waste.
PUBLIC DEBATE:
The relevant authorities, the “Commission Particulièr e du Débat Public” (CPDP) was put on hold in 2004, until a decision on the site choice. The CPDP has been reactivated in July 2005. The final version of the file of the Public Debate has been completed and sent to the “Commission Nationale du Débat Public” (CNDP). The Public Debate will take place during the first semester of 2006, in particular the CPDP will organise about sixteen public meetings in all the Provence region. Different media, prepared for the Public Debate (e.g. multimedia interactive terminal, different types of models) with the financial help of the local authorities completed by EFDA within framework of the TW4-TES-COLABA task, were completed.

IN-FENCE STUDIES:
The site drawings were updated, taking into account the interfaces with the ITER International Team. A survey of the hydrogeology is performed, with a synthesis report every year. This survey will be used to design the draining system. A new interactive model of the ITER site was also realised. The technical specification for the “First office building”, for the arrival of the International ITER Team has been completed and a call for tender was launched, in order to build this 100 offices building during the first semester of 2006. This building will have around 100 offices, several meeting rooms and will host the ITER International Team before the construction of the main office building on ITER site.

OUT-FENCE STUDIES (TRANSPORT OF THE HEAVY COMPONENTS):
The five diagnostic studies concerning the transport of the ITER components (Initial status of the environment, Characterisation of bridges, Detailed maps and longitudinal profiles, Construction of an unloading dock and Feasibility of a localised dredging) were finalised within the framework of the Regional Steering Committee. All the files were transmitted to the “Direction Régionale et Départementale de l’Equipement” (DRDE) in charge of the realisation of the improvements of the itinerary: a call for tender was launched by these State Services to choose a “maître d’oeuvre” to study in detail all the modifications needed.

SOCIOECONOMIC ASPECTS:
Just after the decision, the implementation of a “Welcome Office” was launched at “La Bergerie”, an ancient building close to the “Château de Cadarache”. This “Welcome Office” is composed of 7 offices and a meeting room. It has been presented to the Negotiators the 12th of September during the N10 Meeting.

Related tasks in the full report
CEFDA04-1161, TW4-TES-COLABA

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CEFDA03-1098 TW3-TDS-MAG: Detailed engineering and manufacturing studies of the ITER magnet system: Poloidal Field (PF) coil windings and cold test assessment
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JW5-FT-3.2RHpart Design and construction of the system for laser detritiation of JET co-deposited layers Remote handling expertise for laser detritiation

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CEFDA04-1182 TW4-THHN-IITF2: First ITER NBI and the ITER NB test facility Progress in the design
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