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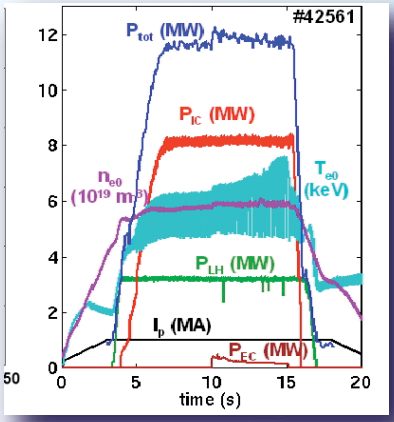
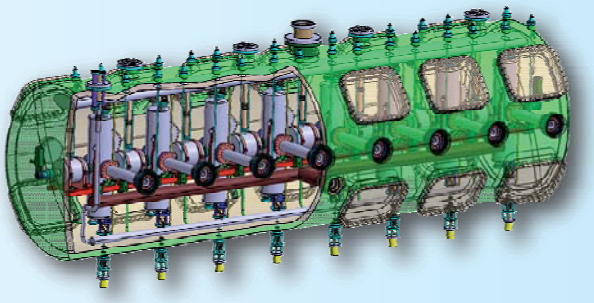
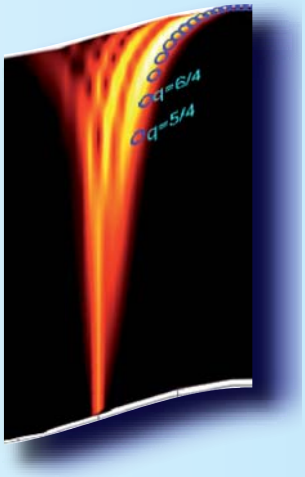




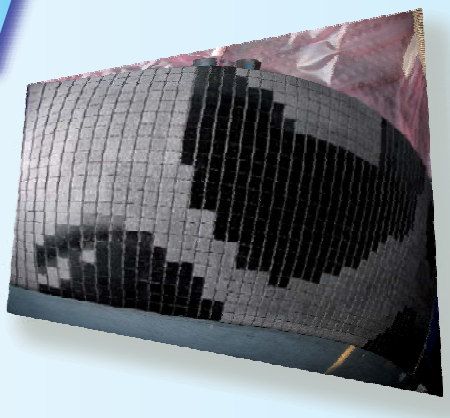


Institut de Recherche sur la Fusion par confinement Magnétique

Association Euratom-CEA

Annual report 2008

(Full Report)



Association EURATOM-CEA

Institut de Recherche sur la Fusion

par Confinement Magnétique

CEA/DSM/IRFM

Association EURATOM-CEA pour la fusion

ANNUAL REPORT

2008

Content

1 Introduction

During the year 2008 and in the frame of the European Programme of research on controlled fusion, the Euratom-CEA Association has pursued vigorous developments in different areas in which it possess key competences, with particular emphasis on the physics and technology of long duration, high injected power discharges.

The 2008 experimental campaign on Tore Supra has been especially fruitful with the completion of multi-annual programmes bringing important answer, notably on operational aspects of long duration discharges:

- In the frame of the "Deuterium Inventory on Tore Supra" (DITS) project, the analysis of a first set of the samples, taken after the specific long duration discharges campaign conducted in 2007, allowed a considerable progress in the comprehension of hydrogen isotopes retention in an environment of plasma facing components (PFC) with carbon based armour material. Solving the discrepancy (of the order of 10) between post-mortem and particle balance analysis, it was possible to conclude that 90% of the deuterium is trapped in redeposited carbon layers.
- During the DITS specific long duration discharges campaign, a growing concern had been the increasing difficulty for coupling heating power to the plasma, with a correlated increase of the number of disruptions. Analysis of the disruptions, observed during the 2005-2006 high injected power campaign and the 2007 DITS campaign, strongly suggested that such behaviour was caused by the overheating of carbon redeposited layers that are badly thermally connected to the actively cooled PFC. In order to ascertain this hypothesis, an extensive cleaning of the PFC of Tore Supra was conducted during the 2008 winter shutdown, in which 800 grams of carbonaceous materials were scraped off. At the plasma restart, it was particularly easy to increase the coupled power up to the record level (for Tore Supra) of 12 MW without the occurrence of any disruption, a very encouraging results in view of the ongoing lower hybrid current drive power (LHCD) upgrade (CIMES project).
- Taking advantage of the long pulse capability of Tore Supra, integrated scenarios have been developed with real time control of the stationary states of the current profile. This includes control of fast ion stabilized sawteeth, transition and sustainment of hot core plasma without MHD activity and recovery from a voluntary triggered deleterious MHD regime.

Besides these highlights, a substantial experimental programme has been conducted on Tore Supra, notably on:

- MHD stability of long duration discharges,
- MHD activities driven by fast particles with the characterization of the modes structure and excitation threshold of β Alven eigenmode and electron fishbones,
- Scaling laws of turbulent transport with for the first time comparison between the measured turbulence spectra and the one simulated by the gyrokinetic global / full distribution code GYSELA,
- Wall conditioning in the presence of a permanent magnetic field using Ion Cyclotron Resonant Heating (ICRH),
- Characterization of the scrape of layer plasma with simultaneous measurement of the electron and ion temperature,

- Disruption mitigation studies with comparison of massive injection of different noble gas.

Concerning JET, the participation of the Euratom-CEA Association has remained high with involvement in the Close Support Unit, the JET Operation Contract and the scientific exploitation. Regarding the scientific exploitation of JET, the Association concentrates its participation on the subject with strong synergism with its specific competences. Particular efforts have been devoted to:

- Hydrogen isotopes retention studies,
- Advanced scenarios studies such as JT60-JET physics identity experiments in ITB regimes, high triangularity hybrid confinement optimisation, development of steady-sate scenarios at high β_N and confinement,
- Commissioning of the ITER-like ICRH antenna,
- High power level commissioning of the LHCD antenna and long distance coupling studies.

A strong involvement in enhancement projects has also to be noted with the successful commissioning on plasma of the fast pellet injector for ELMs mitigation, the procurement of 3 reflectometers (V-, W- and D-bands), tests of interferometer electronics for fringe jumps correction, and participation to the ITER-like wall project, trough detachment of key experts to the operator as well as Beryllium tiles modelling.

Regarding plasma theory and modelling, the global / full distribution function gyrokinetic code GYSELA has been upgraded with the inclusion of ion-ion collisions, through a reduced Fokker-Planck collision operator and with a heat source. These modifications allow exploring more realistic tokamak turbulence regimes. The non-linear MHD code JOREK has been applied to the simulation of the evolution of ballooning modes driven unstable by the large pressure gradient in the H-mode edge pedestal. The reduced MHD model currently implemented has been extended to include the parallel flow velocity. Edge modelling has focused on the study of control of ELMs by resonant magnetic perturbation (RMP), in particular on the response to RMP of a MHD rotating plasma that could change significantly the RMP penetration into the plasma. Concerning integrated modelling, the development of the CRONOS code suite has been carried on, with particular emphasis on increasing its reliability and international coverage. This year, CRONOS has been extensively used to investigate steady-state scenarios for ITER and DEMO. Finally, he participation to the Integrated Modelling Task Force remains a high priority of the Association with 6 Project Leaders or Deputies and a deputy Task Force leader.

On the technological side, the LHCD power upgrade project for Tore Supra (CIMES project) has considerably progressed. The prototyping phase is now completed: the manufacturing of the first series of nine klystrons is progressing satisfactorily and the first five klystrons have passed their factory acceptance tests at the end of 2008. All components (wave guide, mode converter, passive active multijunction (PAM) modules) necessary for assembling the actively cooled PAM antenna are now in house. The PAM launcher and the first half (8 klystrons) of the LHCD emitter should be operational on Tore Supra for the 2009 autumn campaign.

Most of the technological developments carried out by the Association concentrates on key domains that apply to ITER:

- Various studies have been conducted to characterize, qualify and prepare the procurements of the superconducting conductors for ITER. Analyses of the cryoplant operation modes have also been performed.
- Activities on actively cooled PFC have concentrated on the definition of acceptance criteria using samples manufactured with calibrated defects. Such studies have been particularly useful for testing, by transient infrared thermography (SATIR) technique, high heat flux units manufactured by European industries in the frame of the prequalification phase for the ITER critical procurement packages.
- Concerning heating and current drive systems, mechanical studies have been initiated for the ICRH antenna for ITER, in the frame of the CYCLE agreement. The use of innovative materials (composite, meta-materials) has been investigated for simulating plasma load on ICRH test beds. Following the assessment made in 2007 for a "day 1" LHCD system on ITER, the design phase is being prepared in view of an EFDA task that should be initiated in 2009. Activities on neutral beam injection have concentrated on fundamental aspects of negative ions production through modelling as well as experiments with new diagnostics as cavity ring down spectroscopy.
- Regarding diagnostics, activities have ranged from design studies for ITER, as for the visible infrared wide angle viewing system and its integration in the equatorial port plug 1, to test of innovative measurement techniques on Tore Supra with the experimental validation of fibre optic current sensors, insensitive to electronic integrator drifts, that could complement for ITER classical inductive magnetic sensors.

Still on the technology side, one highlight in 2009 was the qualification of an 8 meter long multipurpose carrier prototype, the articulated inspection arm (AIA). After baking at 200°C, the AIA, equipped with its vision system, was fully deployed in the Tore Supra vessel under ultra high vacuum ($1.4\ 10^{-5}$Pa) and high temperature (120°C) conditions, a world first of its kind performance. Following the AIA deployment, no degradation of the vacuum vessel conditioning was detected and plasma operation restarted without needing any additional conditioning. The AIA design is compatible with a complete close inspection of one sixth of ITER plasma vessel.

Outside the scope of the Euratom-CEA contract of association, a strong activity has been conducted in the frame of the broader approach agreement between Japan and Europe:

- CEA has been strongly involved in the redesigning of the JT-60SA project aiming at a significant cost decrease. In this framework, the EU packages partially (9 out of 18 TF magnets, Power supplies) or fully (cryoplant) dedicated to CEA were considerably optimised. Besides, a new package was added in CEA responsibility perimeter, the TF cold tests facility, while the 140 GHz gyrotron procurement package was cancelled due to the project decision to postpone the ECRH installation.
- Concerning the Engineering Validation and Engineering Design Activities of the International Fusion Materials Irradiation Facility (IFMIF-EVEDA), CEA contributes to the project team, located in Rokkasho, Japan and is involved (Irfu, in Saclay) in the accelerator by ensuring the European coordination and the delivery of major systems for the accelerator prototype. In 2008, the main activities of the project have been focused on the prototypes and the engineering design of the Test Cell.

- Concerning the highly parallelised computer, called IFERC-CSC, CEA participates into this project via the IFERC Special Working Group number 1 that is preparing the benchmarking process for the procurement phase. A set of high-level benchmark codes, which are to be sent to the bidders for answering the call for procurement, has been selected.

The Euratom-CEA Association is still strongly involved in collaborative activities that are summarized in annexes 9.1 for international collaborations and 9.2 for national collaborations. Among Euratom Associations, there is an important activity with the 3 German Associations totalizing 9 active topics, and with the Czech Association IPP, with scientific work on 4 topics. Collaboration with other associations is not less active, and 26 collaborations are reported from 11 different countries. Outside the European Union, the Association have collaboration with almost all ITER partners (except Korea). The strong development of collaborative activities with China, observed in 2006-2007, continues to increase and reaches now a level of 9 topics. Within France the number of academic institutions involved in the EFDA fusion programme continues also to increase, in the frame of the "Fédération de Recherche sur la Fusion Magnétique". The number of involved laboratories reaches now 31 with 50 research projects conducted.

Overall, 2008 has been a very fruitful year with many achievements on the experimental, modelling and technological sides. A key feature of the research conducted by the Euratom-CEA Association is the integration of the physical and technological constraints. This allows complementarity and cross-fertilization between the various fields necessary to progress in magnetic confinement research. Such an approach relies on preserving a good balance between fundamental, experimental and project oriented activities, balance that the Association intends to maintain in the coming years.

2 Tore Supra status

2.1 Operation

2008 was a very fruitful year, in terms of plasma operation and results, even if the global availability was slightly reduced compared with those achieved in 2006 and 2007. In the framework of the preparation of the future Tore Supra program once the CIMES project completed, the 2008 restart program focused on the establishment of the wall conditions required to routinely couple to the plasma RF power above 10 MW. During the part of the 2007 experimental campaign, dedicated to the Deuterium Inventory on Tore Supra (DITS) project, a clear link had been found between the growth of the deposited carbonaceous layers that takes place in the shadowed areas of the actively cooled plasma facing components (PFC) and the disruptivity of the RF heated plasmas. During the winter shutdown 2007-2008, it was thus decided to clean up all the PFC and to remove the carbonaceous layers that have grown since the completion of the CIEL project in 2002. This allowed, for the first time on Tore Supra, to easily couple additional heating power up to 12 MW power level. This result also allowed the reassessment of the wall conditioning procedures used on Tore Supra. The operation of a series of new diagnostics, as well as the routine operation of older one, has greatly contributed to the success of the experimental campaign. In 2008, the in-situ validation of the AIA robot was also an important achievement.

2.1.1 Winter shutdown 2007-2008

The winter shutdown was mainly dedicated to the periodic maintenances of the cryogenic and cooling water systems and to the standard annual regulatory controls that impose a warming up at 300 K of the superconducting magnet. In addition, corrective maintenances of the RF heating systems were conducted on both the LHCD system, intensively used in 2007 in the framework of the DITS program, and the ICRH system on which the degradation of the surface property of the vacuum RF feed through had limited the injected power all along the 2007 year, as well as on the ECRH. This intensive maintenance program allowed restarting with the full capability of the RF heating systems. In parallel of this maintenance program, the cleaning of all the PFC was conducted inside the vacuum vessel: it consisted in removing, with hand scrapers the deposited layers without damaging the CFC or the graphite of the PFC. This operation required 186 hours of work performed by four teams of two operators equipped with the adequate personal protective equipments along 10 working days. The result was quite successful (see Figure 2-1). Finally, 144 g of carbonaceous materials were collected from the Toroidal Pump Limiter and 646 g from the guard limiters and the other PFC. For the safety of the operators the dust concentration in the atmosphere of the plasma chamber during this operation was monitored and found below the quantification limit (0,15 mg/m^3).

2.1.2 Plasma restart and commissioning

In addition to the standard operation of commissioning the tokamak subsystems, the 2008 restart has been dedicated to the characterisation of the effects of the cleaning of the PFC on the plasma properties. The standard wall conditioning procedure was unchanged in

order to well discriminate it from effect of the PFC cleaning. This resulted in two remarkable results:

- all along the campaign it was possible to breakdown the plasmas at low voltage,
- the increase of the coupled heating power was performed in less than five days, in July, reaching very easily a total injected RF power close to 12 MW (see Figure 2-2).

During the restart a series of high power discharges (such as shot 41942 11.8 MWx10 s) was realized: it allowed cumulating four minutes with coupled power level in the range of 10-12 MW (see Figure 2-3). Contrary to what happened during the 2007 campaign no disruption was associated to the high power discharges. All along the campaign the oxygen concentration remained low and only one boronisation was performed. These results clearly demonstrate the link between the carbonaceous layers and the disruptivity of the discharges at high hating power. They also confirmed the validity of the wall conditioning procedures applied on Tore Supra.

2.1.3 Global availability and statistics

In 2008, the 109 operation days originally planned were reduced down to 69 effectively realized due to a series of incidents. The most significant ones were:

- an in vessel water leak occurring on an infrared endoscope (7 days lost) due to a faulty assembly following a curative maintenance,
- a depressurisation of the superconducting coils (25 days lost) that occurred during the cooling down of the coils, due to an unusual sequence of events.

It should be noted that the teams in charge properly managed the consequences of these incidents and that the time delays were kept at the minimum achievable.

Finally the availability of Tore Supra for 2008 is 52 %. This figure is below what was achieved in 2006 and 2007. However, the 2008 availability was the same than what was obtained in 2004 and 2005 (see Figure 2-4) with a higher total number of discharges (see Figure 2-5).

Figures

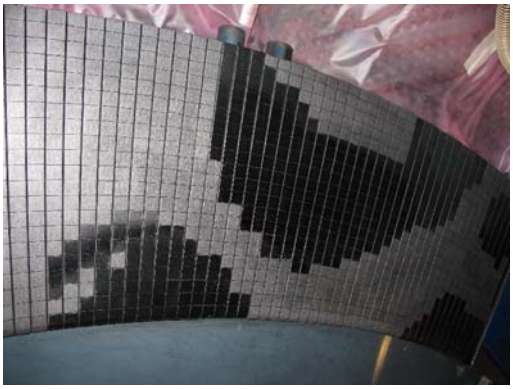


Figure 2-1: a TPL sector after the cleaning procedure; the bright areas correspond to treated zones that recover the bright aspect of CFC; the dark areas correspond to eroded zones that have not been treated.

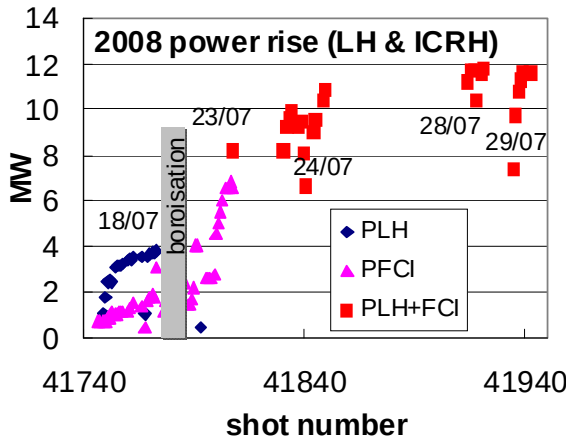
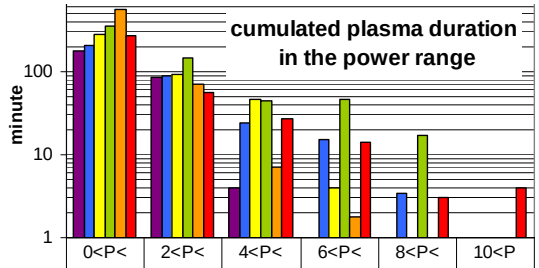


Figure 2-2: 2008 power increase.



	0<P<	2<P<	4<P<	6<P<	8<P<	10<P
2003	177	85	4	1	0,1	0,3
2004	210	90	24	15	3,4	0,5
2005	284	92	46	4	0,7	0
2006	350	146	45	46	17	0,3
2007	556	70	7	1,76	0	0
2008	269	56	27	14	3	4

Figure 2-3: cumulated plasma duration for a given power range.

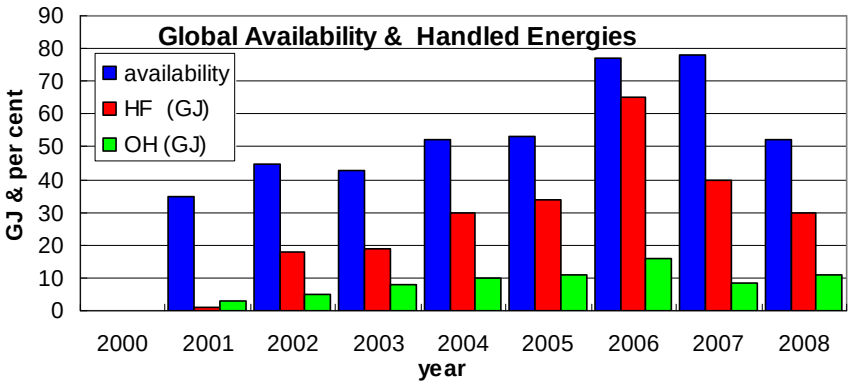


Figure 2-4: global availability of Tore Supra.

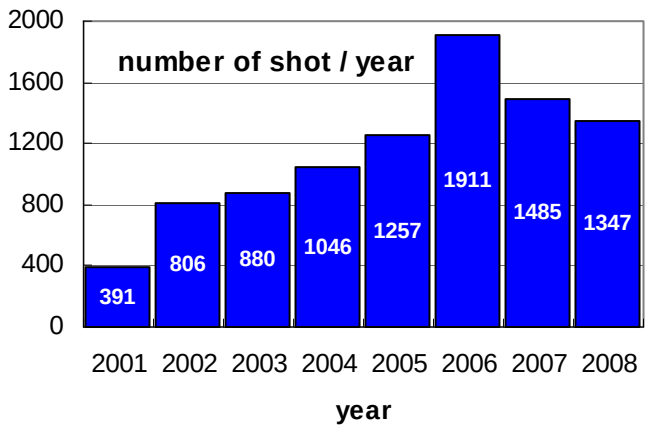


Figure 2-5: total number of plasma discharges performed per annual experimental campaign.

2.2 CIMES enhancement

2.2.1 Klystrons

The manufacturing of the first series of nine klystrons (see Figure 2-6) progressed satisfactorily. The first five klystrons respectively passed their factory acceptance tests at the end of 2008. These five klystrons have demonstrated high reliability and important margins towards operational limits for the three modes of operation: diode mode, operation on perfectly matched load, and operation on unmatched load (VSWR=1.4). Figure 2-7 illustrates the final test of the factory acceptance of the 3rd klystron (10 successive pulses of 1000 seconds duration every 3000 seconds at 720 kW power level on matched load). The five first klystrons are already in house. The delivery of the first series (nine klystrons) is expected for April 2009.

After the factory reception, all tubes has been installed and tested on the Tore Supra test bed. The transport procedure has been checked by verifying the internal vacuum of the tube, the good behaviour of the filament used to heat the cathode and the nominal performances on matched load. Figure 2-8 shows the klystron output power obtained on Tore Supra and on TED tests beds.

The second series of nine klystrons has been launched in May 2008. The fabrication, test and delivery of these nine klystrons will directly follow the ones of the first series without off-load period.

The first half part of the transmitter will be equipped with the first 8 tubes for the 2010 experimental campaign. In a second step, the full transmitter (16 klystrons) will be equipped for the second part of 2010 experimental campaign.

2.2.2 C4 launcher

All components necessary to assembly the C4 passive-active-multijunction (PAM) launcher have been delivered to Cadarache. The 16 mode converters were delivered in March and the 2 passive-active multijunction modules in September for the ½ PAM INF and in November for the ½ PAM SUP (see Figure 2-9). They have been tested (low level RF test and

leak test of the water channels) and validated. The low level RF tests results are in very good agreement with expectations from calculation.

Concerning the PAM launcher, the brazing of the first bank was conducted during December 2007. Unfortunately, a large number of unacceptable brazing defects (lack of brazing as shown on Figure 2-10) were found, when opening the brazing oven on December 26th 2007. Following an in depth analysis, it was concluded that the problem comes from the fact that the assembly to be brazed is composed of a large number of bi-material plates. Errors due to the fact that the bi-material plates are not perfectly plane and that there is discrepancy between theoretical value of the dilatation coefficient and the experimental one, cumulates so that a reliable prediction of the truss rods length is not possible.

An innovative procedure to cope with this difficulty has been proposed by CEA. It consists in using pressurized bellows (Figure 2-11) between the plates and the truss rods so that the pressure can be controlled during the whole brazing process. Various tests have been conducted at the supplier facilities to validate the repairing procedure. The reparation has been conducted in July 2008. The second part of the PAM launcher has been brazed after the successful reparation of the first part.

Following the PAM modules delivery, a new schedule has been established foreseeing that the launcher will be ready for operation on Tore Supra in October 2009.

Figures

Figure 2-6: new klystron TH2103C.

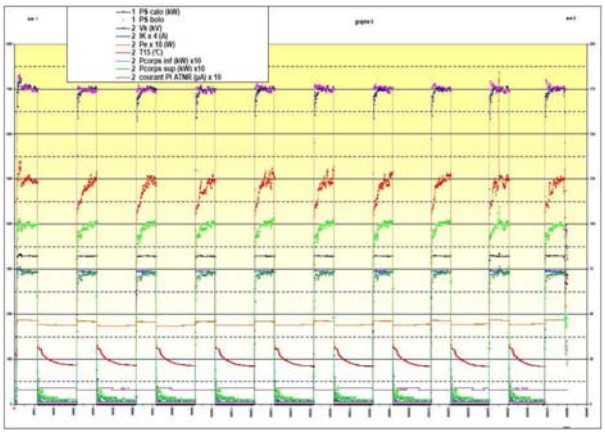


Figure 2-7: run of ten successive pulses during the 3rd klystron factory acceptance tests.

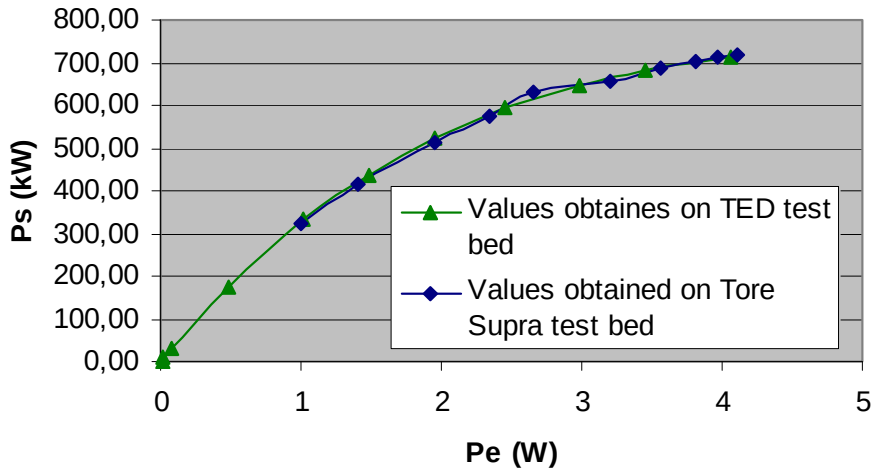


Figure 2-8: TH2103C klystron prototype output power on matched load versus input power.

Figure 2-9: the top (left) and bottom (right) banks after final machining.

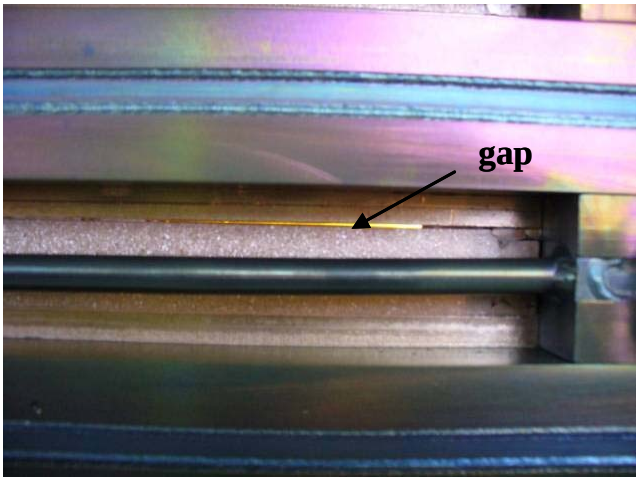


Figure 2-10: lack of brazing between two plates.

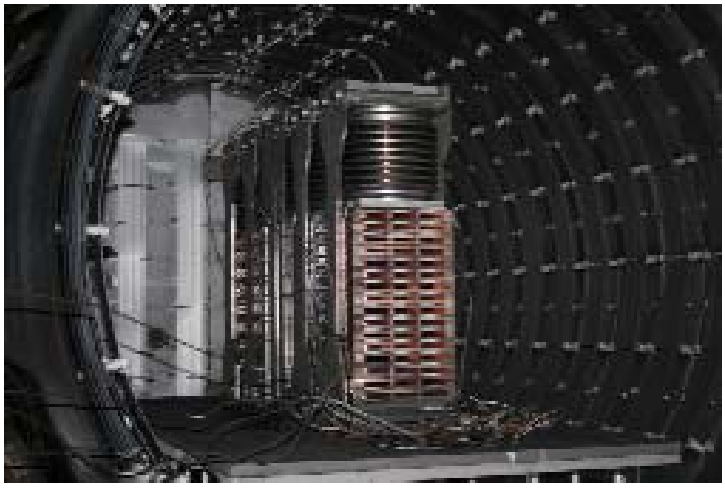


Figure 2-11: new cycle of brazing with pressurized bellows.

3 Theory & modelling

3.1 First principles based theory

This section deals with classic magnetic confinement theory. It covers work aimed at improving our understanding of physics processes occurring in fusion machines, code development based on fundamental plasma physics equations ("first principle") and their application to experiments, and work aimed at simplified models or formulas to be used in integrated modelling.

3.1.1 Transport/turbulence

The global gyrokinetic code GYSELA has been upgraded with the inclusion of ion-ion collisions, through a reduced Fokker-Planck collision operator and with a heat source. These modifications allow exploring more realistic tokamak turbulence regimes. Code development has been pursued in collaboration with specialists from INRIA on matters like parallelisation and, in general, code performance.
In the absence of turbulence, below the micro-instability threshold, the code is able to recover known results of neoclassical theory, like the magnitude of the heat diffusivity and the proportionality of poloidal ion rotation v_θ to the temperature gradient [1] (see Figure 3-1). Above the instability threshold, a two-parameter scan as a function of turbulence drive and collisionality provides a cartography for the turbulent heat diffusivity and v_θ in this two-dimensional space [2], [3]. The positive dependence of turbulent transport on collisionality occurs already at modest temperature gradients and persists well above the effective collisionless non-linear threshold. Turbulent generation of poloidal momentum is also observed. The observed dependence of flow shear on collisionality, primarily governed by turbulence, indicates that its effect could be important at ITER-like collisionality.

Validation and benchmarking of turbulence codes has been carried out through a EU-wide effort coordinated by the EFDA Integrated Tokamak Modelling Task Force, as well as through bilateral collaborations. Within the ITM-TF effort, the poloidal spectra of the electrostatic potential fluctuations have been compared between a dozen of fluid and gyro-kinetic codes with some success [4].

Research aimed at identifying optimal choices of coordinates for turbulence simulations was pursued. The goal is to reduce the computer resources (CPU time and storage) needed for a simulation of given accuracy and duration (in physical time units) by a large factor (perhaps 20-50 for ITER parameters), by aligning one of the coordinates to the direction of small gradients. Building on previous work, it was demonstrated that for a large class of electrostatic turbulence models, it is possible to employ a generic, arbitrarily fine mesh in the poloidal plane, independent of the magnetic geometry, while at the same time retaining a coarse mesh in the toroidal coordinate, which is adequate to compute gradients along the ambient magnetic field.

First principle physics based and numerically fast reduced transport models are necessary to improve the scenario designing capability. In order to test the validity of the quasi-linear approach, a transport model based on an eigenvalue gyrokinetic code, QuaLiKiz [5], has been intensively compared to both nonlinear simulations and fluctuation

measurements. The possible choices in the free parameters of the electrostatic potential wave number spectra are weighed in the light of a comparison between turbulence measurements, obtained by reflectometry in Tore Supra, and nonlinear gyrokinetic simulations, using GYRO and GYSELA [6]. The total quasi-linear fluxes (combining the quasi-linear response with the choice on the fluctuating potential) given by QuaLiKiz, are compared to the nonlinear fluxes computed by GYRO [7], while varying various dimensionless parameters such as the normalized temperature gradient R/L_T (see Figure 3-2). A single coefficient is adopted to renormalize the quasi-linear predictions to the nonlinear fluxes. Interestingly, both the ratios between the transport channels (ion energy, electron energy and particle fluxes) and the parametric behaviour of the fluxes agree well for coupled ITG-TEM turbulence. This transport model is now ready to be integrated in CRONOS to predict temperature and density profiles.

3.1.2 MHD

The non-linear MHD code JOREK has been applied to the simulation of the evolution of ballooning modes driven unstable by the large pressure gradient in the H-mode edge pedestal. The reduced MHD model currently implemented has been extended to include the parallel flow velocity. This allows the study of the outflow of energy and particles along the open field lines to the divertor [8]. The simulations show the formation of a fine scale (order of centimetres) structures in the temperature profile at the divertor, forming spiral structures in the toroidal direction (see Figure 3-3). The structures are due to the temperature perturbations induced by the ballooning mode and are in qualitative agreement with the 'stripes' observed in the experiments. Meanwhile, the JOREK code has undergone further development along two lines:

- Extension of the physics model to two fluids reduced MHD with a separate evolution of ion and electron temperatures. First results indicate that the filaments forming during the ELM are likely to have a larger ion temperature relative to the electron temperature due to the lower parallel conductivity of the ions [9].
- The implementation of the full MHD model. This requires the definition of a coordinate system. Here a new approach is being studied: using the local basis vectors of the Bezier finite element formulation as coordinate system (aligned to the magnetic field).

The operational version of the full MHD code XTOR [10] was used to study double tearing instabilities in the so-called "giga-Joule" non-inductive TORE-SUPRA discharges. The code reproduced successfully tearing instabilities with large saturated islands. However, the model was unable to reproduce some of the dynamics observed experimentally, such as saturated 2/1 tearing instabilities with small island sizes. This suggests that other effects (e.g. bi-fluid) play an important role, probably because these modes are in the vicinity of the stability threshold. As a further development, XTOR was equipped with a new pre-conditioned fully implicit Newton-Krylov time stepping method, and systematic tests were carried out. As a first step towards the bi-fluid version of the code, XTOR-2F, different formulations of this model were investigated. The main difficulty is to take into account the drift effects in toroidal geometry, at the same time keeping the numerical scheme as close as possible to the implicit method already used in XTOR. In parallel, a completely implicit time advance method was developed and successfully tested for the hybrid MHD-particle code XTOR-K.

Experimental and theoretical work was conducted towards understanding MHD instabilities driven by fast particles, and in particular modes in the BAE/GAM frequency range, triggered by fast ions, and modes in the fishbone frequency range, triggered by fast ions or electrons [11]. Following the observation of Beta Alfvén Eigenmodes in Tore-Supra, the study of their linear characteristics and stability has been undertaken. This work has led to a detailed theoretical computation and understanding of the mode frequency and structure [12]. The mode destabilization threshold in the presence of a population of fast ions was derived analytically in order to get insight into the parameters of interest in the mode drive. To validate this calculation, experiments were conducted in Tore-Supra and compared with theory, using simulations of the fast ion population and equilibrium parameters. This comparison confirmed the expected role of most macroscopic parameters in the mode destabilization and revealed less intuitive features of the mode onset with a varied shear or heating localization. Furthermore, an extensive study of fishbone-like modes in LH-heated Tore Supra plasmas has characterized the role of fast electrons. Redistribution of suprathermal electrons in both energy and radial spaces has been correlated to periodic mode frequency jumps using the hard X-ray emission measurement [13]. Numerical resolution of the MHD mode dispersion relation using electrons distributions calculated by the Fokker-Planck code LUKE has confirmed the electron-driven instability at the frequency observed experimentally.

3.1.3 Heating & current drive

The code LUKE solves the 3-D linearised relativistic bounce-averaged Fokker-Planck equation in the zero-banana width limit using the finite difference method. It is particularly fast and robust for heating and current drive calculations in tokamaks or any axisymmetric magnetic configuration with arbitrary poloidal cross-section. The kinetic solver is linked to the universal ray-tracing C3PO designed for electron waves at Lower Hybrid and Electron cyclotron frequencies, and to the R5-X2 fast electron Bremsstrahlung code that simulates non-thermal radiation emitted in the heated plasma in the X-ray energy range. Together, the tools C3PO, LUKE and R5-X2 form the LUKE electron heating and current drive package [14] which itself is incorporated into the CRONOS tokamak modelling package for scenarios studies. A version of LUKE for the European Integrated Tokamak Modelling task force is also developed. The code developments in 2008 were oriented principally towards:

- the full implementation of the distributed computing capabilities for reducing time consumption of irreducible calculation loops,
- the implementation of a self-building on-line documentation, necessary in the context of the international use of LUKE (China, India, Czech Republic, Switzerland),
- a simplified maintenance through standardised code installation,
- a link to a new ray-tracing code AMR for Electron Bernstein wave studies developed at IPP-Prag (Czech Republic),
- a new module for investigating the role of a non-Maxwellian electron distribution function on the onset of fast electron driven MHD instabilities observed on Tore Supra, an important issue for ITER scenarios,
- a new graphical user interface for easy simulations.

In the framework of numerous collaborations, LH or EC current drive studies have been carried out for SST-1 and ADYTIA (IPR, India), EAST and HT-7 (ASIPP, China), COMPASS (IPP-Prag, Czech Republic), JET (EFDA), TCV (EPFL, Switzerland), besides Tore Supra. Extensive activity was devoted to parametric studies of LH on ITER [15], [16],

and of LH assisted current ramp-up studies for JET and ITER. Comparisons between fast electron Bremsstrahlung experimental observations and LUKE predictions have been carried out for the Tore Supra tokamak [17].

The evolution plan of the kinetic parallel full wave code EVE has shifted from a phase devoted to fundamental developments to a new phase, in which the code is exploited and any new development is driven by the direct needs of the simulated physics phenomena. The first stable version of the code, numbered 1.0.0, was released in 2008. The most noticeable developments in the code in the previous year are:

- Integration: an upgraded interface of the code with the equilibrium code HELENA The code has also been integrated in the ITM framework and is now capable of working with external equilibriums directly read from the ITM data tree.
- Order Reduction Algorithm (ORA): EVE is a second order FLR code that, by construction, is limited to the description of damping up to the second harmonic of the cyclotron resonance. However, to model Fast Wave Electron Heating (FWEH) / Current Drive (FWCD) experiments, it is necessary to incorporate damping effects of fast particles, potentially occurring at higher harmonics. The implementation of an ORA has allowed including this phenomenon into the code (see Figure 3-4).
- A Python Post-processor to deduce the full 3-D solution depending on the antenna phasing as a superposition of toroidal Fourier modes.

In parallel to these developments, the stable version code has also been exploited to model several experiments, among which:

- Simulation of Fast Wave Current Drive in Tore Supra and ITER [18].
- Modelling of ICRF heating in JET and its influence on impurity transport [19].
- Extensive simulations of ICRH scenarios during the activated phase of ITER have been performed. In particular, the influence of the antenna phasing was thoroughly investigated [20].

3.1.4 Edge modelling

One of the promising methods to control Type I ELMs is the installation of dedicated coils that achieve this goal by modifying the edge magnetic field by Resonant Magnetic Perturbations (RMP). RMPs have been shown to be effective in eliminating Type I ELMs in DIII-D or significantly mitigating them in JET. At present, ELM control by RMP is recommended for ITER since it could increase the lifetime of the divertor by reducing heat and particle fluxes due to Type I ELMs and hence reducing surface erosion. IRFM is actively participating in experimental and theoretical studies of RMPs on the existing machines (DIII-D, JET, COMPASS, MAST, AUG) with the aim of better understanding of underlying physics of ELM suppression by RMPs for the further optimisation of RMP coils design proposed for ITER [21]. This work was initially supported by EFDA Tasks [22], later on by ITER Design Review activity and at present by ITPA and EFDA MHD Topical Group preferential support. The progress done in physics studies has permitted to propose the present design of 9 3-row RMP in-vessel coils for ITER (see Figure 3-5) with optimized spectrum (see Figure 3-6) for all ITER reference scenarios to create ergodic zone at the plasma edge for $r/a>0.9$ similar to DIII-D scenarios with ELM suppression by I-coils. The present studies are focused essentially on MHD rotating plasma response on RMPs (such as screening of RMPs, braking of plasma rotation etc) that could change significantly the RMP penetration into the plasma [22].

The effort on the theoretical investigation of plasma wall interaction has been pursued in collaboration with the M2P2 team in Marseille. The framework of this work is to build two codes for the modelling of plasma-wall interaction with ab-initio descriptions of plasma turbulence with a 2-fluid description. A simplified version, the code SOLEDGE-2D is being used to test numerical schemes and to model key physics issues on the basis of diffusive transport. This code has been used to investigate the penalty technique to model complex wall configurations while allowing one to use the extremely efficient pseudo spectral numerical schemes. The application and validation of the penalty method that allows one to recover the Bohm boundary conditions has been achieved. The physics that has been addressed with this code is that of the departure of the SOL profiles from the standard exponential fall-off. In particular 2D effects have been shown to lead to two separate decay regions in the vicinity of the limiter. In the deep SOL the usual e-folding length is observed while in the near separatrix region a sharp decrease is found. The latter region is in fact a region bridging the edge plasma with near toroidal symmetry and the SOL with relatively strong parallel gradients. The investigation of the SOL flow imprint on the edge flows has also been investigated. A specific scaling for the edge flow magnitude and structure has been predicted and validated numerically, the SOL flow spreading into the edge with an e-folding length scaling with a power ¼ of the Schmidt number (ratio of the particle diffusion coefficient and the viscosity). A specific aspect of plasma-wall interaction and the effect of flows is the occurrence of Kelvin-Helmholtz turbulence in the vicinity of the limiters. This effect has been analysed and modelled with the SOLEDGE-3D code. Kelvin-Helmholtz has been observed when the Schmidt number departs from 1 and appears to be generated close to the limiter as expected from the linear theory.

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Figures

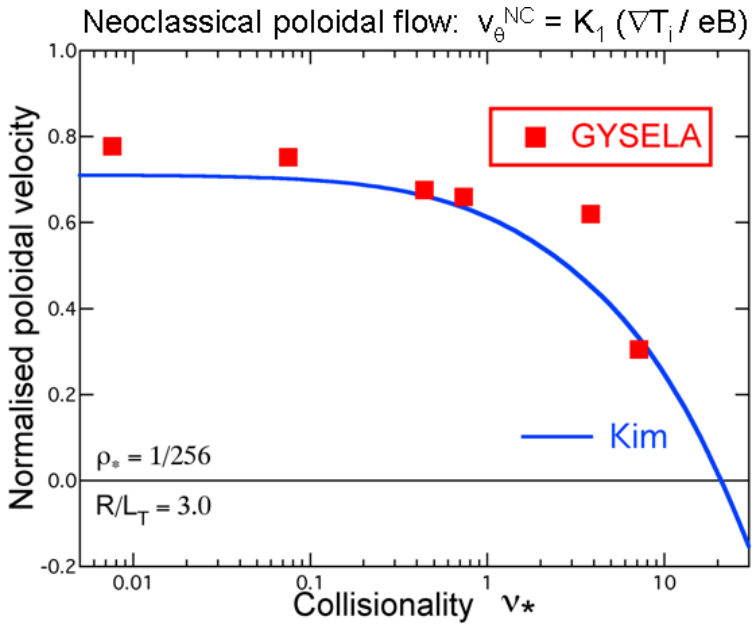

Figure 3-1: in the absence of turbulence, the poloidal flow computed in GYSELA exhibits the same proportionality with the ion temperature gradient as predicted by neoclassical theory [Y. B. Kim et al., Phys. Fluids B, 3 (1991) 2050], both at low (banana) & intermediate (plateau) collisionality

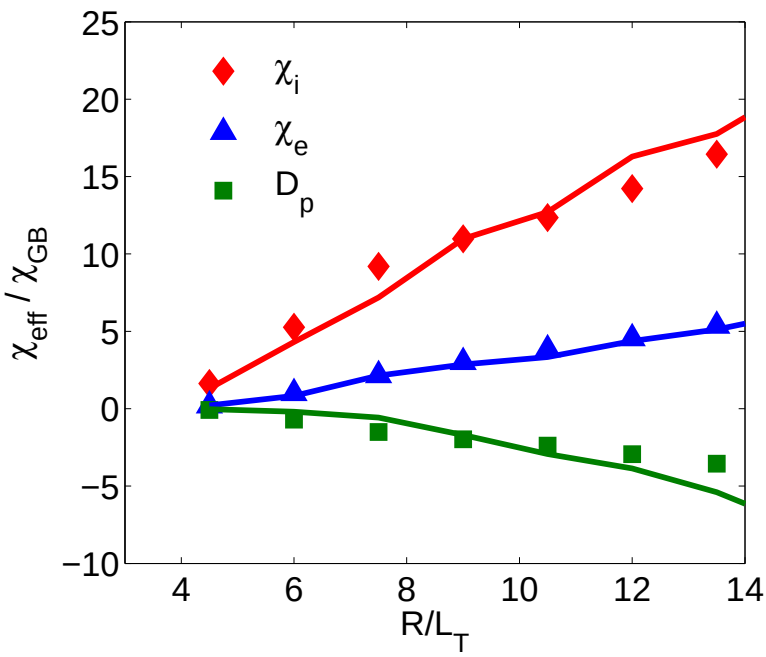

Figure 3-2: effective conductivity vs. normalised temperature gradient

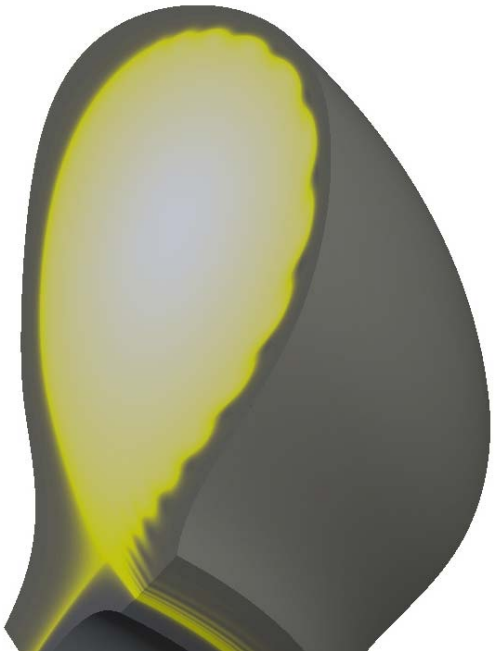

Figure 3-3: the temperature profile perturbed by a ballooning mode leading to multiple stripes at the divertor target.

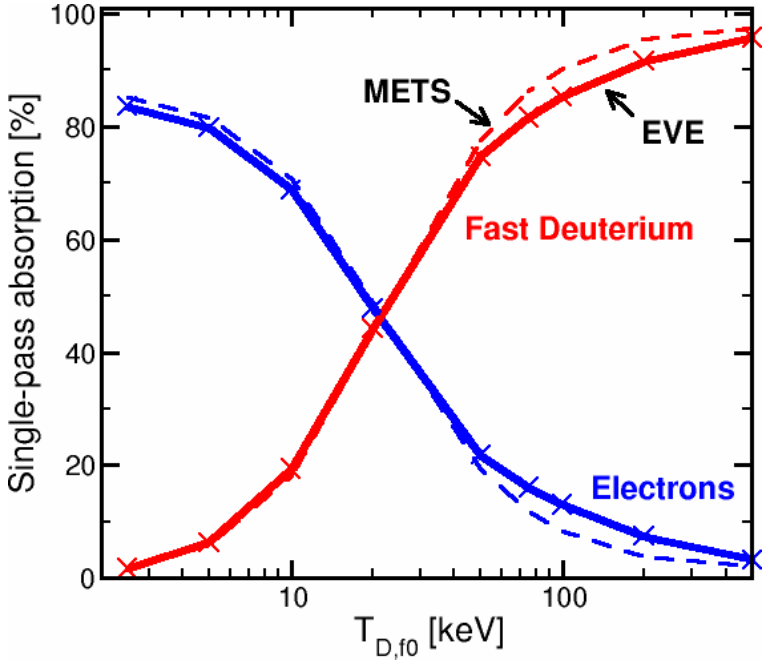

Figure 3-4: power damping competition between electrons and fast deuterium (D) ions at the 3rd harmonic of the corresponding cyclotron resonance versus the fast ions temperature. EVE and the 1D all-orders code METS yield similar results.

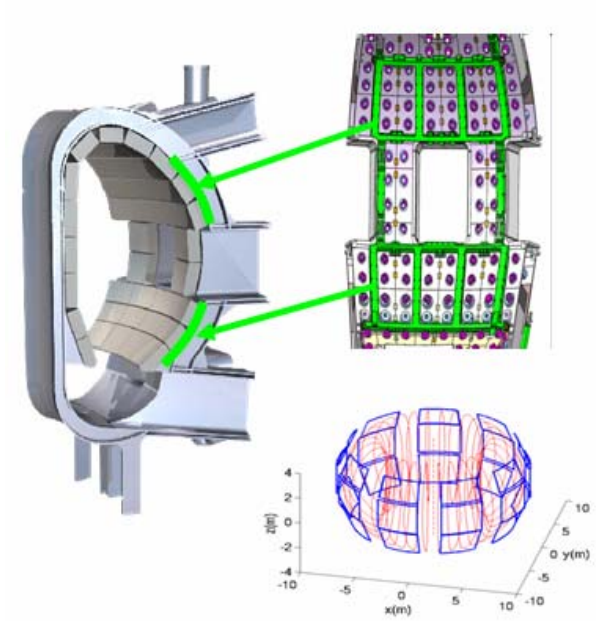

Figure 3-5: *schematic view of RMP coils in ITER.*

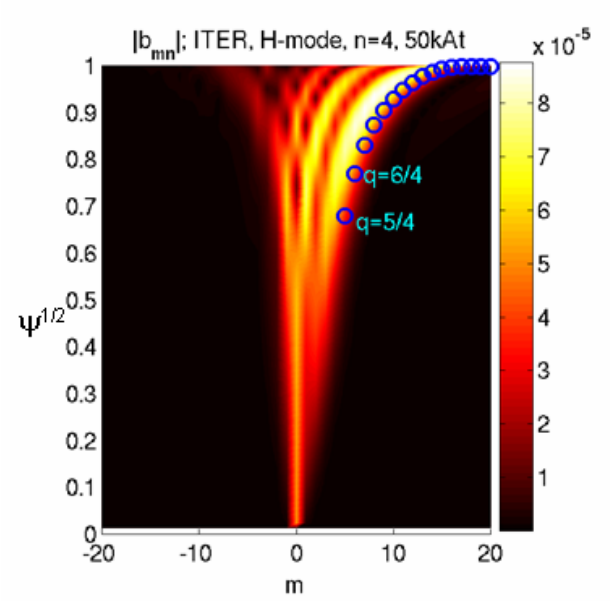

Figure 3-6: *poloidal spectrum of magnetic perturbation strength in Hamada flux coordinates for ITER coils (n=4,50kAt).*

3.2 Integrated modelling

3.2.1 CRONOS development

The CRONOS integrated modelling suite has been developed along the following lines, in order to increase its reliability and international coverage:

- Ported to Matlab 2007b on the new cluster Cephee
- The CRONOS suite has been installed in the two main Chinese fusion laboratories: SWIP (Chengdu) and IPP (Hefei), and in the Indian fusion laboratory: IPP

(Ahmedabad). Our foreign colleagues have been taught how to use the code locally.

- Website for project members installed on the "partenaires" zone: our foreign collaborators can now check the simulation catalogue, submit information or hotline calls from anywhere in the world: https://wikicronos.partenaires.cea.fr/wiki/index.php/Main_Page
- The graphical interface has been redesigned to be more "physicist-friendly"
- Several new non-regression tests added

In addition, a new neutral beam injection (NBI) module has been developed: NEMO (Figure 3-7), integrated to the CRONOS code and benchmarked with TRANSP NBI solver. NEMO calculates the source of fast ions in the plasma resulting from NBI. It is based on the principles of the former SINBAD module, but completely reshaped and upgraded to more recent computing features (fortran 90, dynamic allocations, etc) and includes also new physics features (Finite Larmor Radius correction, use of ADAS cross sections). NEMO is a stand-alone fortran program also integrated within the CRONOS integrated modelling suite, where it is coupled to external Fokker-Planck calculations (fast analytic calculation module or SPOT orbit following Monte Carlo code) in order to simulate the trajectories of fast ions inside the plasma and the resulting heat, particle, current, and torque sources to be used in the core transport equations.

A new free-boundary equilibrium solver, FREEBIE, has been also developed for the CRONOS suite. This feature is a key axis of development of integrated modelling codes today, allowing them to simulate the self-consistent evolution of poloidal field coils controllers and plasma transport equations. Based on the well-known EFIT equilibrium solver, FREEBIE can handle both the direct (knowing the current in the PF coils, determine the plasma separatrix) and inverse problems (knowing the plasma profiles and separatrix, calculate the currents in the PF coils). The next steps to be done in 2009 are to further validate FREEBIE against existing codes / experimental data and to build a plasma shape controller for CRONOS.

3.2.2 Integrated modelling taskforce activities

The IRFM was in 2008 the first European association in terms of manpower allocated to the ITM Task Force, with 6 Project Leaders or Deputies and a deputy Task Force leader. Among the main activities and achievements of last year:

- Development and maintenance of the code platform: KEPLER, Universal Access Layer, Data Structure [23], Integrated Simulation Editor
- Development of tools to link to and import experimental data: machine description, time-dependent data mapping: thanks to these generic tools it becomes possible to import pulse-based data from any tokamak to the ITM database, with the relevant mapping of signal names / definitions.
- The core transport equation solver of CRONOS has been completely rewritten in Fortran, inserted in a KEPLER workflow and run under the ITM environment. This is the first prototype of the European Transport Solver workflow (Figure 3-8).
- Extension of the ITM data structure to Radio Frequency physics (RF waves propagation and absorption, LH grill and EC mirror description)

- Benchmarking of various turbulence codes (see chapter on transport and turbulence)
- Within the IMP#1 project, the EQUAL equilibrium reconstruction code (formerly called EFIT-ITM), the HELENA equilibrium code and the ILSA MHD stability code have been adapted the ITM data structures and are available as actors in Kepler.
- Participation to the ITER Scenario Modelling activities, focusing on the simulation of current ramp-up and ramp-down phases of tokamak discharges. The main activity last year has been to validate a transport model relevant for current ramp-up / ramp-down scenario design against experimental data from JET and Tore Supra, then to apply it to the simulation of ITER current ramp-up phase [24],[25].

3.2.3 Scenarios studies

The scenario modelling studies, outside of the ITM/ISM group (see above), has been focused on the steady-state scenarios for ITER and DEMO.

The possibility of steady-state scenarios with zero loop voltage and lasting for 3000s has been analyzed by means of the CRONOS suite of codes [26]. A scenario for ITER with Internal Transport Barrier (ITB) has been found with pure Radio Frequency heating and current drive systems, Electron Cyclotron Heating and Current Drive (ECH/ECCD), Ion Cyclotron Heating (ICH) and Lower Hybrid (LH), as shown in Figure 3-9. This scenario provides a solution to the current alignment problem, which caused the shrinking and erosion of the ITB in previous studies performed with Neutral Beam Current Drive (NBCD). In this new scenario, the ECCD locks the ITB at mid-radius and avoids its erosion and shrinking. However, there is a clear power threshold for this feature. The LH power deposition is located at ρ=0.7, and the current drive obtained ($\approx$ 0.6 MA) contributes to the total non-inductive current fraction (fni$\approx$97%). The main feature of this scenario is that a critical behaviour exists for the global process of ITB formation and sustainment in tokamaks by means of the suppression of turbulent transport due to negative magnetic shear. This critical behaviour, different to that existing for the ITB creation, has significant consequences for ITER. Since the critical shear needed is strongly negative, s=-0.8, large current drive inside the ITB (e.g., NBCD) destroys it after some current diffusion times due to misalignment of the currents.

The CRONOS suite of codes has been also used to simulate and analyze the DEMO design in the case of a regime with high non-inductive current fraction [27]. The main results show that a large Q=26.5 can be obtained with low NBI input power (= 98 MW) and a high pedestal temperature, 7.8 keV. This scenario leads to a hybrid-like q profile with q_0>1. The non-inductive current fraction is rather high, 88%, with a high amount of bootstrap current, 10.0 MA mainly coming form the pedestal, and 6.8 MA of NBI current drive, which is possible due to the 2 MeV launcher system considered. It has been shown that, due to the large density considered for these scenarios, a 1 MeV NBI system could not be enough to drive sufficient current, as well as the redistributed current inside the plasma could be a handicap to control the q profile. In this scenario, 30 MW more of NBI can be added to attain the steady-state regime with 100% non-inductive current as shown in Figure 3-10. However, due to the large amount of current coming from the NBI system, the time evolution of the q profile cannot be stopped. In order to fix this issue, a more flexible system of ECH or LH has been considered. With this configuration, the control of the q profile is easier; however, due to the high densities considered in these DEMO scenarios (for maximizing the fusion power) the

current drive obtained is much lower than in the NBI case. This issue can be overcome by adding more power, but the final Q drops considerably.

Therefore, the possibility of a DEMO device with advanced scenarios, i.e. high fusion gain with high bootstrap and non-inductive current fraction and relatively small size, will require strong physics performance: high density peaking, high pedestal height, Greenwald limit fractions much higher than 1 or ITB's lasting for very long times. These features are out of the scope of the present day fusion devices. Thus, integrated modelling can provide important key points for the establishment of DEMO scenarios that could, in practice, be also worth for the research plan of ITER.

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Figures

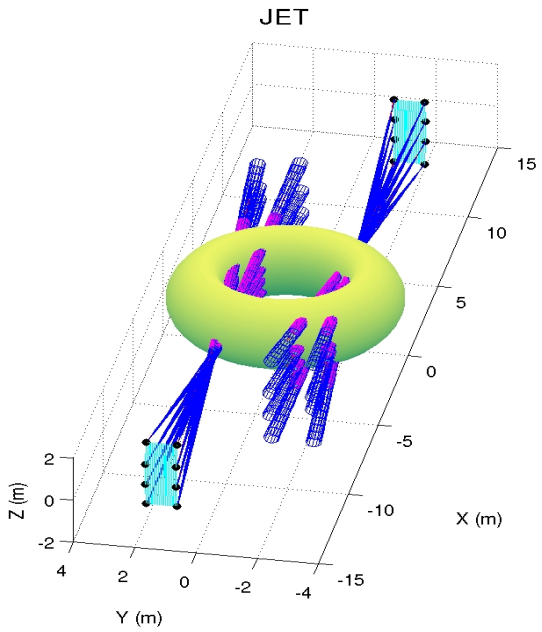

Figure 3-7: 3D geometry of the JET Neutral Beams and Injectors and Beams as implemented in the NEMO code. Each injector is represented by a black point (JET is equipped with two boxes of 8 injectors). The plasma torus is represented by the yellow-green contour. The beam divergence cones are the blue contours

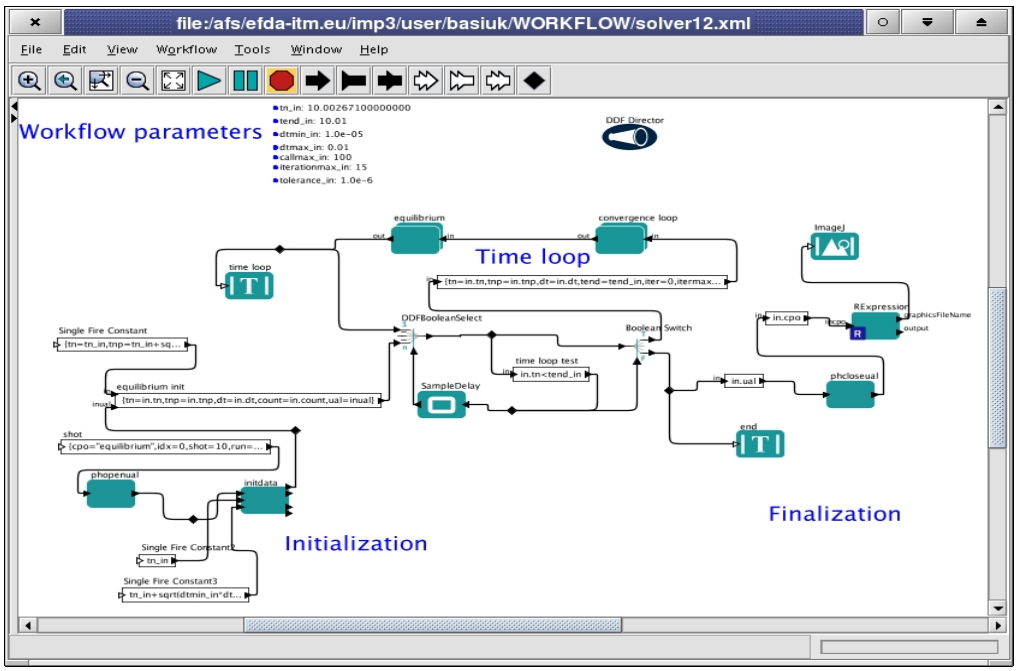

Figure 3-8: European Transport Solver prototype workflow. The workflow design and modules are based on the CRONOS core transport equation solver.

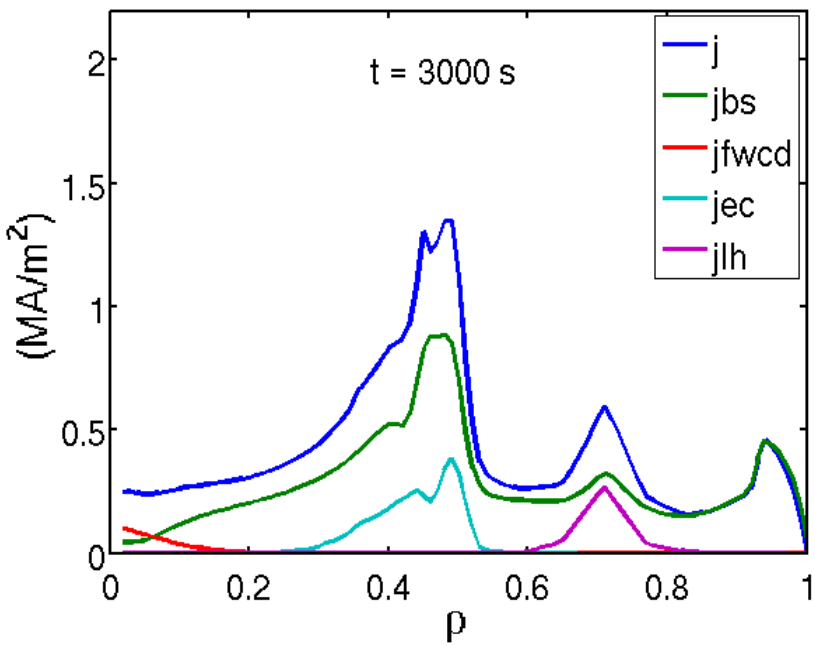

Figure 3-9: total current (j), bootstrap current (jbs), fast wave (jfwcd), electron cyclotron (jec) and lower hybrid (jlh) current drive density profiles at t=3000s for the ITER steady-state scenario.

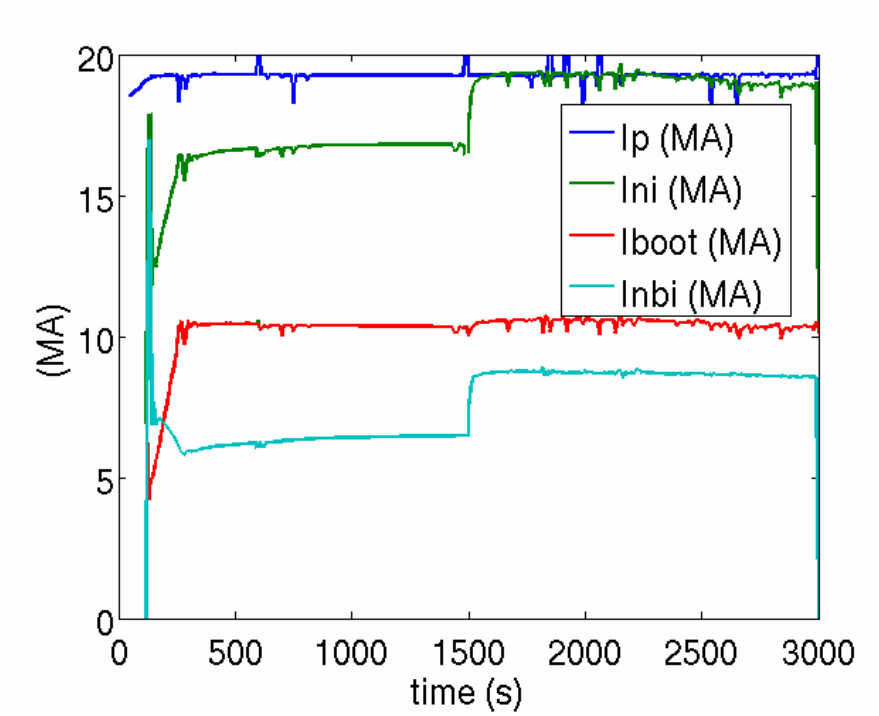

Figure 3-10: evolution of the total current (Ip), total non-inductive current (Ini), bootstrap current (Iboot) and NBI current drive (Inbi) for the DEMO scenario.

4 Experimental physics

4.1 Scenario developments

In this section, EURATOM-CEA activity related to the plasma scenario developments performed on Tore Supra (sections 4.1.1, 4.1.2) and on EFDA-JET (section 4.1.3) is summarised. A complementary approach has been used to develop scenarios on both devices focusing our activity on issues related to steady-state tokamak operation [28] where control of the plasma profiles (current and pressure) plays a key role in optimising/sustaining the fusion performance.

On Tore Supra, a dedicated Task-Force (Action Programme-1, AP-1) has been set-up in 2008 with the objective of developing high power scenarios in view of preparing the coming long pulse operation after the completion of the CIMES project (higher LHCD power with a new set of klystrons and a new LHCD antenna) [29]. During the 2007-2008 winter shutdown, the carbon deposits on the PFCs have been completely removed. This thorough cleaning procedure had a dramatic positive impact on the operation of the machine. After a very fast ohmic restart, characterised by rather low impurity content and quickly decreasing radiated power fraction, LH, then ICRH power increase phases have been performed. As a result, no particular effort was required to obtain discharges with nearly 12 MW of total heating power, for duration of 10 s, very close to the maximum available power on Tore Supra. In 2008 power handling in Tore Supra scenario was not limited by the low power disruption phenomena encountered since 2005 [29]. Consequently during the 2008 experimental campaign, Task Force AP-1 activity has been focused on the following scenario issues:

- Development of integrated scenario with real time control of the stationary states of the current profile as reported in section 1.1.1 [30]
- Real time control of sawtooth activities with ECCD as reported in section 1.1.2 [30][31]
- High power combined LH and ICRH operation (B_T=3.8 T, I_P = 1 MA) and related coupling studies as reported in section 1.3
- Plasma breakdown assisted by ECRH. In 2008, the emphasis was to study ECRH breakdown at the second harmonic X-mode as requested by ITER for reduced toroidal field operation. These experiments nicely complement the 2007 experiments performed in the first harmonic O-mode [33]. It was found that higher power delivered by 2 gyrotrons were necessary to reach the plasma pre-ionisation whereas one was necessary in O1. The lowest loop voltage to initiate the plasma was ~0.5V/m (perpendicular injection) compared to 0.15V/m in O-mode first harmonic.
- Exploration of ICRH second harmonic Hydrogen minority heating scenario at reduced toroidal field (2T) as an ITER relevant RF heating scheme (on ITER second harmonic of Tritium will be used). An ICRH power up to 5MW has been reached combined with 2MW of LHCD. The main difficulties were related to the observation of a degraded state of B4C layers on the lateral protection of the ICRH antenna (antenna Q5) producing localised over-heating limiting the available powers for antenna safety reasons (Infra-red protection).

- Very first experiments in view of accessing a transition to an H-mode at low toroidal field (at B_T=1.9T) with magnetic configuration of reduced TF ripple. These experiments were carried at the end of the experimental campaign and have not lead to the triggering of the H-mode since the maximum coupled power was ~6MW below the expected threshold to get an H-mode.

4.1.1 Real time control of stationary states of the current profile on the Tore Supra tokamak

The current density profile plays a leading role in plasmas physics and tokamak scenario development since it governs the global stability (MHD activity) of the discharge and micro-turbulence (energy confinement). Real time control of current profile offers the possibility of achieving and sustaining required plasma states with given confinement and MHD properties. This is of utmost importance in the context of developing steady-state fusion reactor-grade scenario with improved confinement while approaching the MHD stability limits (advanced tokamak concept).

In most present day tokamaks, the plasma duration is for technical reasons limited to a few current resistive times. As a consequence, the current profile is only controlled transiently (for instance during the plasma current ramp-up phase as on JET). This is usually done by acting on the resistive diffusion across the plasma radius of the ohmic electric field that remains in most cases the dominant source of current. Demonstrating the control of successive stationary states of the current profile throughout a plasma discharge requires long-pulse capability (i.e. $\tau_{plasma} >> \tau_{resistive}$), which is a distinctive feature of Tore Supra.

We report in this sub-section innovative Tore Supra experiments in which real time control of successive stationary states of the current profile has been successfully achieved [30]. These experiments involve a combination of Lower Hybrid Current Drive (LHCD) and Ion Cyclotron Resonance Heating (ICRH), using up to 8 MW of power with plasma durations up to 40 s (which represents about 20 resistive times). The plasma current profile is controlled by varying the level of LHCD power, i.e. replacing part of the ohmic current by a non-inductive source with a different location. The safety factor profile can thus be varied as required from an ohmic like monotonic profile with $q_{min} \leq 1$ (sawtoothing plasma) to a non-monotonic one profile with $q_{min} \sim 3/2$. Within this range of q_{min}, the q-profile evolves through five distinct states, characterised by specific MHD activity. These q-profiles states are detected by real time analysis of the electron temperature relaxations resulting from the MHD activity, observed on the central chords of the Electron Cyclotron Emission diagnostic. Every 2 s, the controller checks whether the plasma is in the requested plasma state and, if not, modifies the LHCD power level in the relevant direction. Prior to the discharge various plasma states characterised by different q-profile are pre-set by the physicists. During the discharge evolution, the feedback controller that compares in real time the actual to the requested state will steer the plasma towards the pre-set state. Successful experiments reported in this sub-section include [30]:

- the control of the sawteeth activity as illustrated on Figure 4-1, where the plasma jumps from a sawtoothing to a non-sawtoothing state (and vice-versa) as requested,
- the transition and sustainment of a “hot core” plasma regime without MHD activity as illustrated on Figure 4-2 where high electron temperature (6keV) is sustained between 20-25s

- the recovery from a voluntary triggered deleterious MHD regime as illustrated on Figure 4-2 where at t=25s the ICRH power is suddenly stopped to trigger a fast MHD event. As a consequence, the controller switches off the LHCD power and tries to recover the required plasma state when the plasma has reached a quiescent phase.

4.1.2 Real time control of fast ion stabilised sawteeth by ECCD on Tore Supra

In tokamak plasma, sawtooth oscillations in the central temperature, caused by a magnetohydrodynamic (MHD) instability, can be partially stabilized by fast ions. The resulting less frequent sawtooth crashes can trigger unwanted MHD activity. Recent Tore Supra experiments have shown for the first time that modest electron-cyclotron current drive power, with the deposition positioned by feedback control of the injection angle, can reliably shorten the sawtooth period in the presence of ions with energies ≥0.5MeV (provided this ECCD is localized correctly with respect to the q=1 surface) [31].

In the Tore Supra experiments, central ICRH (57 MHz) was used to create a significant central pressure of fast ions with energies in the MeV range. The effect of co- and counter-ECCD on the sawtooth period has been explored in discharges where the capability of the Tore Supra ECCD system of varying toroidal and poloidal injection angles over a wide range was exploited to sweep the ECCD deposition from outside the q=1 surface to the plasma centre. An overview of one such discharge with co-ECCD is shown in Figure 4-3. For reference, the traces of a similar discharge without ECCD are also shown. As can be seen, the sawtooth period increases from about 25 ms in the Ohmic phase to around ~80 ms when ICRH is applied, i.e., when fast ions are created. During the ECCD deposition sweep, there is a sudden drop in sawtooth period almost to the level in the Ohmic phase; when the ECCD deposition moves closer to the centre, the sawtooth period goes back up to 80 ms again. Based on the above experimental results, real time sawtooth period control was implemented varying the poloidal ECCD injection angle in real time to modify the ECCD absorption location. Figure 4-4(a) shows an attempt at simple feedback control of the sawtooth period. The sawtooth period error (the difference between requested and measured sawtooth period) is used as input to a proportional-integral (PI) controller, which determines and controls the ECCD absorption location. As the requested sawtooth period in this case is between the two achievable values, the controller causes the measured sawtooth period to oscillate between these two values. To overcome this problem, a ‘‘search and maintain’’ control algorithm has been implemented. This algorithm initially varies the ECCD absorption location, in search of a location where the sawteeth are sufficiently short; once this has been achieved, the controller maintains the distance between the ECCD location and the measured sawtooth inversion radius, thereby maintaining short sawteeth throughout the ECCD pulse as seen in Figure 4-4(b).

4.1.3 CEA participation to Scenario Development at JET-EFDA

The EURATOM-CEA Association has extensively participated to the EFDA-JET experimental campaigns in 2008 C20-C25 (from April up to December 2008). Up to 31 physicists have been at JET during this period, corresponding to a total manpower of ~5.0ppy under order. They have participated to these campaigns covering a large range of scientific

expertise. The scientific activity has been focused towards the participation to the scenarios task force dealing with the development and exploitation of scenarios for ITER. The EURATOM-CEA Association is highly involved at the coordination level of the scenario Task forces with the leader and deputy leader of Task Force-S2 (hybrid [33] and steady-state scenarios [35]), and the deputy leader of Task Force-S1 (Baseline ELMy H-mode scenario). In addition, physicists from the EURATOM-CEA have lead the following experiments at the level of Scientific Coordinators (SC) carried within the scenarios Task forces:

- S2-2.3.4 Development of Steady-State scenario at high β_N and high confinement (SC from CEA)
- S2-2.3.6 JT-60-JET physics identity experiment in the ITB regime-joint JET & JT-60U experiment carried out within the ITPA (SC from CEA)
- S2-2.2.3 High triangularity hybrid confinement optimisation (Deputy SC)
- S2-3.2.1 Fuelling of hybrid and steady state scenarios (Deputy SC)
- S2-2.3.5 Current and pressure profile control (back-up experiments not executed in 2008-SC from CEA)

In this report, we focus on the description of the experiments S2-2.3.4 & S2-2.3.6.

The ITER non-inductive scenario requires high performance steady-state plasmas ($Q\approx5$, $\beta_N\approx3$, $H_{IPB98(y,2))}\approx1.5$, $n/n_{Gw}\approx0.8$, $q_{95}\approx5$) with edge conditions compatible with the plasma facing components [36]. This scenario aims to operate at high pressure and modest plasma current with a large bootstrap current providing the conditions for steady-state operations of ITER NBI-only experiments have been performed in JET at reduced B_T, I_p, ($\leq$2.25T, $\leq$1.6MA) to study the plasma stability and confinement at high normalised $\beta_N>3$ with various target q-profiles. The focus of our studies concerns experiments done at higher B_T/I_p (2.7T/1.8MA) and power (higher electron heating ICRH and LHCD) to reach T_i/T_e, ρ^* and ν^* nearer to ITER values for steady-state operation [37]. With $P_{NBI}\approx20$-23.8MW, $P_{ICRH}\approx2$-6MW, $P_{LHCD}\approx2$-3MW, in plasmas with weak magnetic shear and $q_0\sim2$, the following has been obtained: $H_{IPB98(y,2)}\approx1.2$, $\beta_N\approx2.7$, $n_l\approx4$-$4.5\times10^{19}m^{-3}$ at $q_{95}\approx5$ with an ITER-like plasma shape. These plasmas have core $T_i/T_e\approx1.4$, $n/n_{GW}\approx0.6$, $\rho^*/\rho^*_{ITER}\approx2.1$, $\nu^*/\nu^*_{ITER}\approx2.2$. In view of using and studying the full capability of heating & current drive mix in JET, these experiments rely on optimising the edge for good RF coupling while maintaining good core and edge confinement. Various techniques (gas dosing, vertical field kicks, Error Field Correction Coil) have been tested to mitigate the impact of the ELMs on the core confinement, by reducing the ELMs size and penetration depth into the plasma core. Improved core confinement with Internal Transport Barriers (ITB) has been sustained in addition to the good edge confinement, raising $H_{IPB98(y,2)}$ up to ~1.2. According to interpretative modelling (CRONOS), the bootstrap current fraction is ~30%, NB current fraction is ~30% and LHCD~20%.

Internal transport barriers (ITBs) are considered as a candidate for enhancing the confinement in Advanced Tokamak scenarios [38]. Different physical mechanisms are thought to enable the formation of ITBs in plasmas by causing a local suppression of turbulence. The dominant mechanism in one device may, however, may differ from that in others. Hence, a series of experiments have been carried out at JT-60U and JET in order to find common characteristics and differences in the process of ITB formation, with the aim to better understand the physics behind the triggering and sustainment of ITBs [39]. These experiments have been carried out with near identical magnetic configurations, similar heating waveforms, plasma current, magnetic field and safety factor. The main goal was to match normalised parameters, q(r), $\rho^*(r)$, $\nu^*(r)$ and $\beta(r)$, at the time when the ITBs are triggered, which was achieved in these experiments. ITBs formed in plasmas with various target q-profile shapes were studied, from reversed central magnetic shear to optimised

positive shear. The plasma rotation profiles were varied in both devices. In JT-60U, the toroidal torque on the plasma was tuned by applying different amounts of co- and counter-current Neutral Beam Injection (NBI) fractions. In the JET experiments the toroidal field ripple, which is known to affect the toroidal plasma rotation, was changed, from the standard value of δ=0.08% at the outer separatrix, to higher values matching the TF ripple in the JT-60U experiments, δ =0.3%, and to even higher values of δ=0.8%. In both devices similar ITBs were triggered, in plasmas with matching normalised parameters and foremost identical q-profiles. The main differences observed were the difference in the density profile during the ITB phase (more peaked in JT-60U), edge characteristics and plasma rotation (higher Mach numbers in JET). Even with a match in TF ripple, a difference in toroidal rotation profiles still remained. The experiments suggest that the q-profile seems to play an important role in the triggering mechanism of ITBs in JET and JT-60U.

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Figures

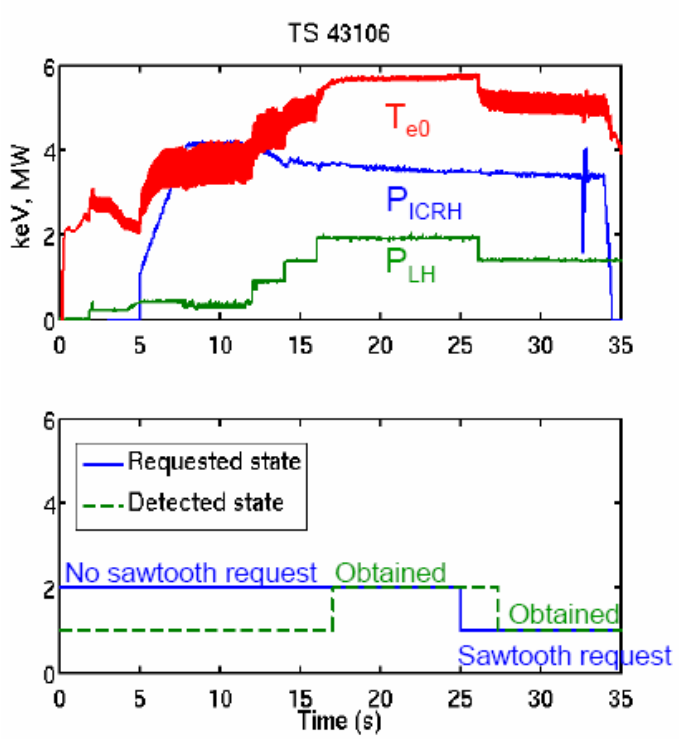

Figure 4-1: plasma real time control of the sawtoothing activity with monotonic q-profile. Tore Supra #43106. Time evolution of the core electron temperature (T_{eo}), ICRH and LHCD powers, requested and measured plasma states as labelled 1-5.

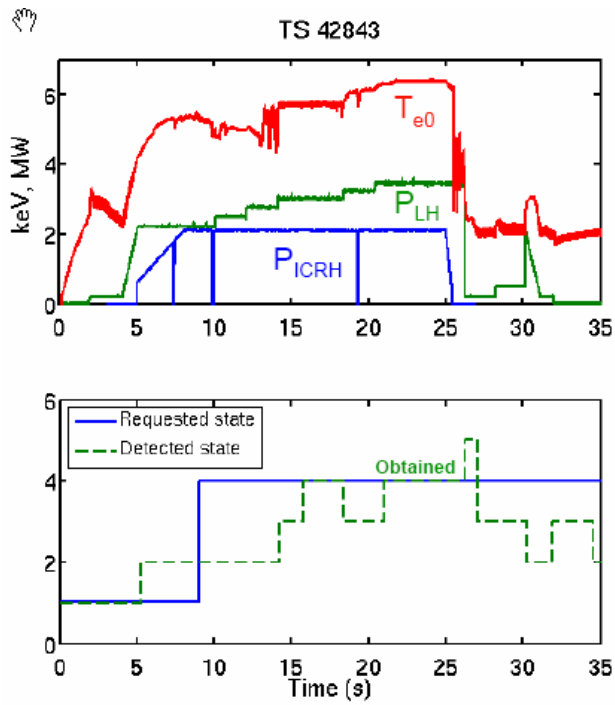

Figure 4-2: plasma real time control of the hot core phase and MHD activity with non-monotonic q-profile (with q_{min}~3/2). Tore Supra #42843. Time evolution of the core electron temperature (T_{eo}), ICRH and LHCD powers, requested and measured plasma states as labelled 1-4.

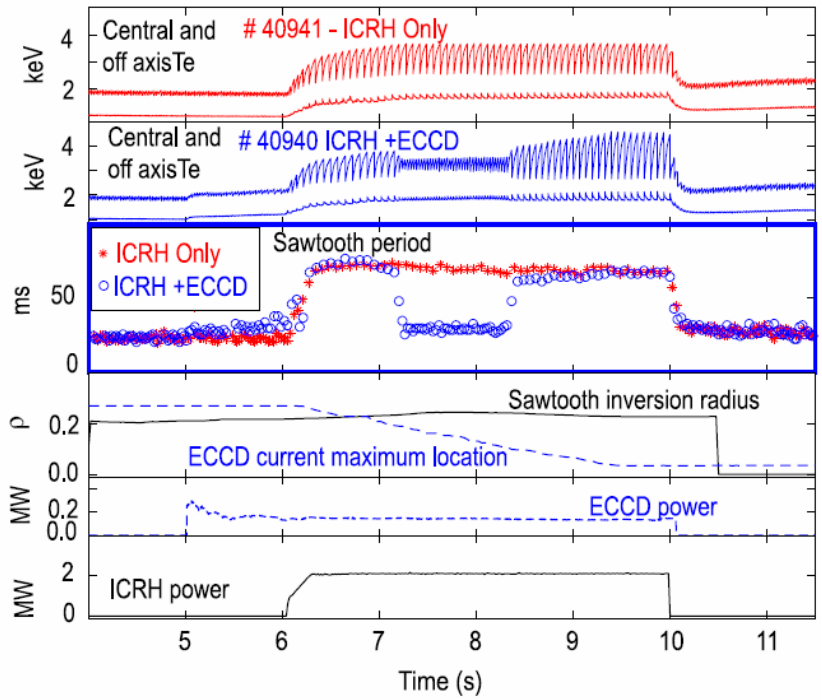

Figure 4-3: demonstration of sawtooth destabilization showing two consecutive shots with 2.3 MW of ICRH with and without ECCD. The radial ECCD location was scanned from outside the sawtooth inversion radius to the plasma centre. From 7.2 to 8.3 seconds, the sawtooth period drops to 30 ms.

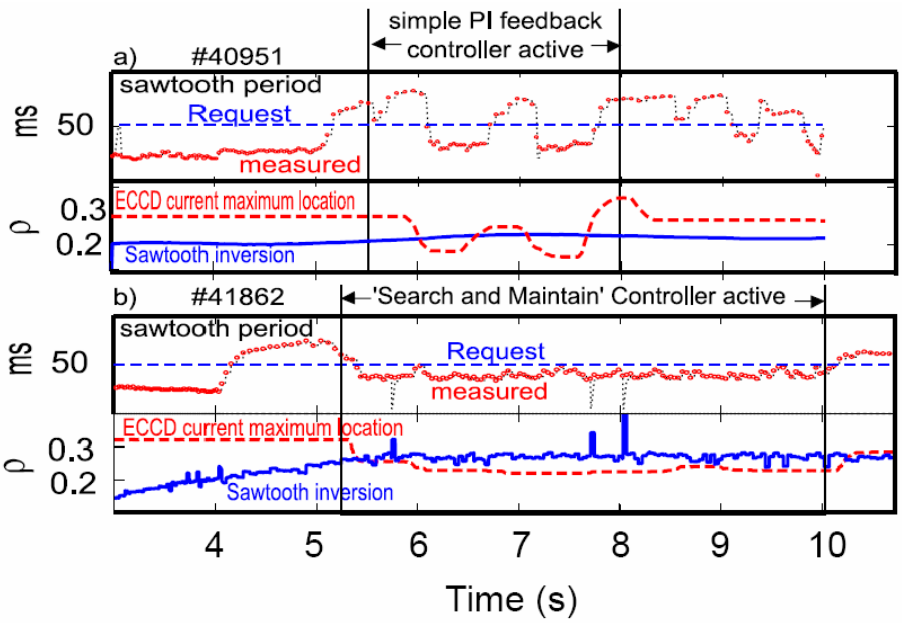

Figure 4-4: (a), (b) Real time sawtooth period control with the poloidal injection angle as the actuator. (a) Oscillation with proportional-integral controller. (b) Short sawteeth obtained and maintained with search and maintain controller.

4.2 Transport, turbulence, MHD

4.2.1 MHD stability

Several studies have been carried on, which are related to the topic "MHD stability and plasma control topic" (section 1.3 of Work Programme).

The analysis of resistive MHD limitations for non-inductive discharges on Tore Supra has been focussed on the issue of the non-linear evolution of resistive MHD instabilities [40]. More specifically, the aim was to find the relevant parameters for discriminating regimes with core localised crashes and the so-called MHD regime, which corresponds to a soft resistive limit resulting in quasi steady confinement degradation. We found that the position of the q=2 surface, where the mode causing the crash develops, is such a relevant parameter (see Figure 4-5). When this surface is inside a normalized radius of about 0.3, the non-linear regime consists of core localized crashes, while more external modes evolve toward the MHD regime. This has been interpreted using the nonlinear MHD code XTOR [41], where the q=2 surface is artificially moved, allowing to recover a regime with core localised crashes and a regime similar to the MHD regime when increasing the q=2 radius. This study also showed that the quiescent non-inductive regime, where no MHD activity can be found on the q=2 surface, is not recovered in MHD simulations. Additional physics coming from a more comprehensive MHD model, such as a two fluid model, may account for this discrepancy.

Modelling of resistive instabilities has also been started at the end of 2008 for experiments on the JET Tokamak. Resistive modes are found to be linearly stable in these experiments, but they can be nonlinearly excited by a seed island (these are NTM, Neoclassical Tearing Modes). Modelling is done again with the XTOR code, and compared to a simple analytic model: both find that the nonlinear threshold is getting smaller at the time a resistive mode (on q=2) is triggered in the experiment.

The issue of disruption mitigation and runaway control has been addressed in Tore Supra experiments, and the study also includes experiments at JET. Successful control of the runaway beam has been demonstrated, and massive gas injection has been performed. These experiments have been very promising, and will be further addressed. Massive gas injection on a normal plasma (intended to be used when the risk of disruption is suspected to be large) has been studied using Helium, Neon and Argon. Although the characteristic time of the disruption is similar for the different gases, the runaway generation is found to be much lower when using Helium.

4.2.2 MHD modes driven by fast particles

Energetic particles driven modes are one of the concerns for burning plasmas. On Tore-Supra, the combined use of ICRH and LHCD systems at significant power generate fast ions and electrons. High sensitivity diagnostics like reflectometry or fast and correlation ECE are able to detect and localize the MHD modes driven by energetic particles [42].

4.2.2.1 Beta Alfvén Eigenmodes

Coherent modes with a macro-scale localized structure have been observed repeatedly in the acoustic frequency range with minority heating ICRH power. Those observations have been reported with reflectometry [43] and Electron Cyclotron Emission (ECE) correlation [44]. Both their structure and the characteristics of their excitation, which appear to be intrinsically linked to the fast particle population, advocate for an identification of these modes as Beta Alfvén Eigenmodes (BAEs) [45] rather than Geodesic Acoustic Mode (GAM). With this identification in mind, the excitation threshold of BAEs has been calculated. The exciting energy comes from fast trapped ions while the mode is stabilized by Landau damping on bulk ions [46]. The condition for linear excitation can then be calculated [47][48]. A careful analysis of their conditions of excitation has been carried out experimentally. Figure 4-6 displays the existence of an excitation threshold depending on various parameters such as the ICRH power, the minority fraction, the density, and the B-field. This threshold agrees with the theoretical calculation obtained from [48].

4.2.2.2 Electron fishbones

In LHCD plasmas, electron fishbones can be excited [49], [50]. Periodical mode frequency jumps were observed in electron fishbones frequency range (≤20kHz), see Figure 4-7. The mode disappearance when LHCD power drops at t~18.8 s points to fast electron driven modes. These modes where identified as precession electron fishbones. Using fast ECE diagnostic it was shown that the mode structure changes after each frequency jump [51]. A redistribution of fast electrons was also observed with hard X-ray diagnostic. The precession fishbone is excited by electron with a given energy so that they are resonant with the modes. When the mode develops, part of these electrons is redistributed from the centre to the gradient zone. When a new mode at a different frequency develops, electrons in another energy band are affected. This behaviour causes periodical oscillation of the fast electron distribution function as on Figure 4-7.

4.2.3 Scaling laws of turbulent transport

In Tore Supra, an extensive study of β, ρ^* and ν^* dependencies has been started in L-mode plasmas, hence without interfering with H-mode edge physics. The goal is to combine local transport analysis by CRONOS with density fluctuations measurements, providing two independent ways to evaluate dimensionless dependencies. These detailed scaling are then confronted to non-linear turbulence simulations [52]. The ultimate objective is to constrain the transport model QuaLiKiz based on quasilinear theory [53]. Density fluctuations are measured from the edge to the core, using fast sweep, Doppler and fixed frequency reflectometers and hence give further insight into possible different local transport scaling.

A methodology for the local transport analysis with the CRONOS code is being proposed in order to account for ripple losses, radiative losses, etc. in a robust way. Four experimental sessions have been dedicated to these studies in 2008 focusing on ρ^* and ν^*. The high level of additional power reached has allowed extending the previous scans, in particular to lower ν^*. Nevertheless, matching the density profile shape at various magnetic fields has been hardly achieved. The impact of this mismatch is under investigation. A first evaluation gives a very weak variation of the confinement time with ρ^* of the order

of $B\tau \propto \rho^{*-1}$. The turbulence measurements by Doppler reflectometer during these shots give access to spectra covering a wide k range as show on Figure 4-8.

Concerning the extension of the ν^* scan to lower ν^*, the confinement time dependency is weaker than the one observed at higher ν^* in 2007. A possible switch between ion dominated turbulence and electron dominated turbulence at lower collisionality is under investigation on the Doppler frequency spectra.

Finally measured k spectra in one discharge have been confronted to the simulated one (see Figure 4-9). The dimensionless profile analysis is ongoing. Once the methodology for the local transport analysis approved, the energy diffusivities will be compared to the turbulence levels measured. And more non-linear simulations will be confronted to measured fluctuation levels and inferred energy fluxes.

4.2.4 Transport of particles and impurities

Experiments on particle transport have been performed in the HL2A tokamak in collaboration with SWIP (Chengdu, China) [54]. A plasma region of reduced particle transport (position $r/a \approx 0.7$, width $\Delta r/a \approx 0.025$) is systematically observed above a line integrated density of about 2.2×10^{19} m^{-3}. The diffusion coefficient and the (inward) convection velocity deduced from the plasma response to a series of supersonic molecular beam injections are strongly reduced at the barrier. In the region enclosed within the barrier they are also reduced compared to the external region. The convection velocity is compatible with the predicted Ware pinch (Figure 4-10). A new series of experiments has been performed jointly in HL-2A and Tore Supra to investigate the effect of ECR heating, the nature of the so-called non-local phenomena and the role of microturbulence in the observations.

Concerning impurity transport, a new injection technique has been validated, namely the supersonic pulsed injections of trace gaseous impurities, similar to the SMBI technique. It was shown [55] that combining the analysis of such injections with that of stationary levels of the same impurity reduces the error bars to about 15% (Figure 4-11). The experimental results in ohmic discharges with weak MHD activity, in agreement with quasilinear gyrokinetic simulations by QuaLiKiz, show that the turbulent diffusion coefficient does not depend on the impurity charge. The profile shape and the order of magnitude of the predicted turbulent convection velocity are also consistent with the observations.

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Figures

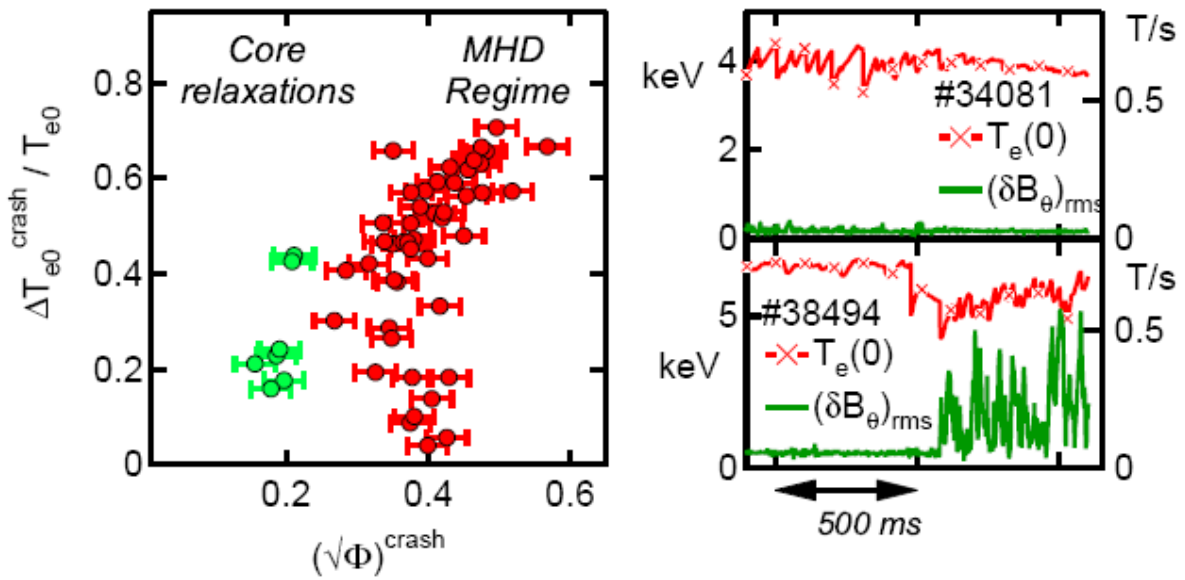

Figure 4-5: Left: Relative temperature drop as a function of the inversion radius of the crash. Right: Examples of relaxations (top) and transition to MHD Regime (bottom).

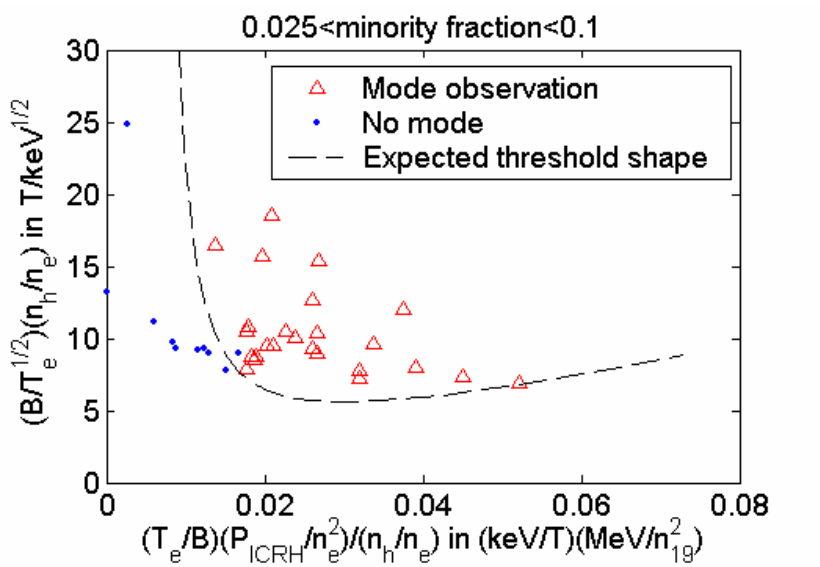

Figure 4-6: Threshold for the excitation of BAE modes on Tore Supra

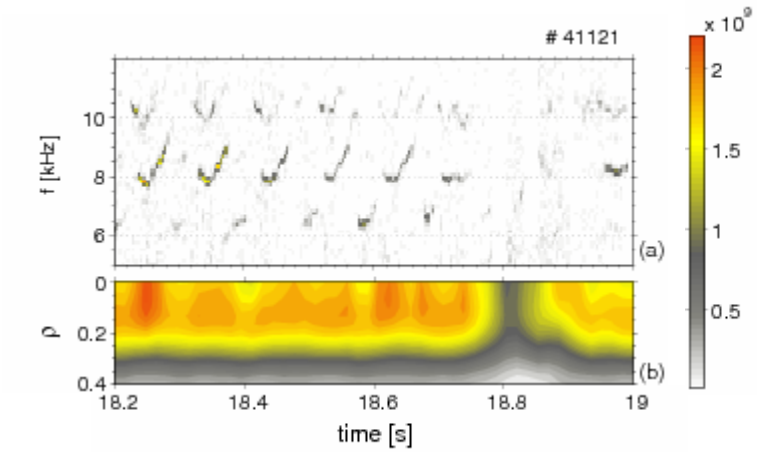

Figure 4-7:(a), Spectrogram obtained by cross-correlating δn and δTe measured respectively by reflectometry and electron cyclotron emission at r/a= 0.2. (b), Filled contour plot of the electron photon emission with energy between 60 and 80 keV obtained by hard- X ray measurement. At t=18.8s, a drop of LH power makes the modes disappear and the fast electrons vanish.

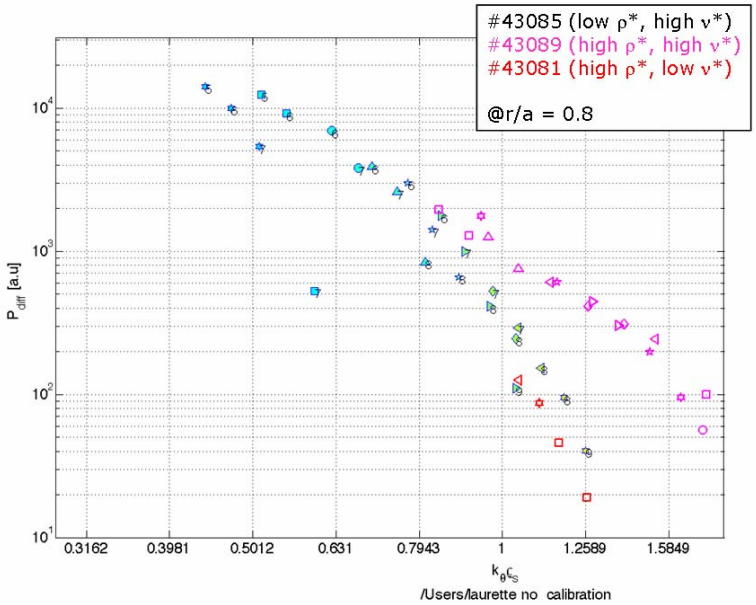

Figure 4-8: Measured spectra of density fluctuations for different values of normalised gyroradius and collisionality.

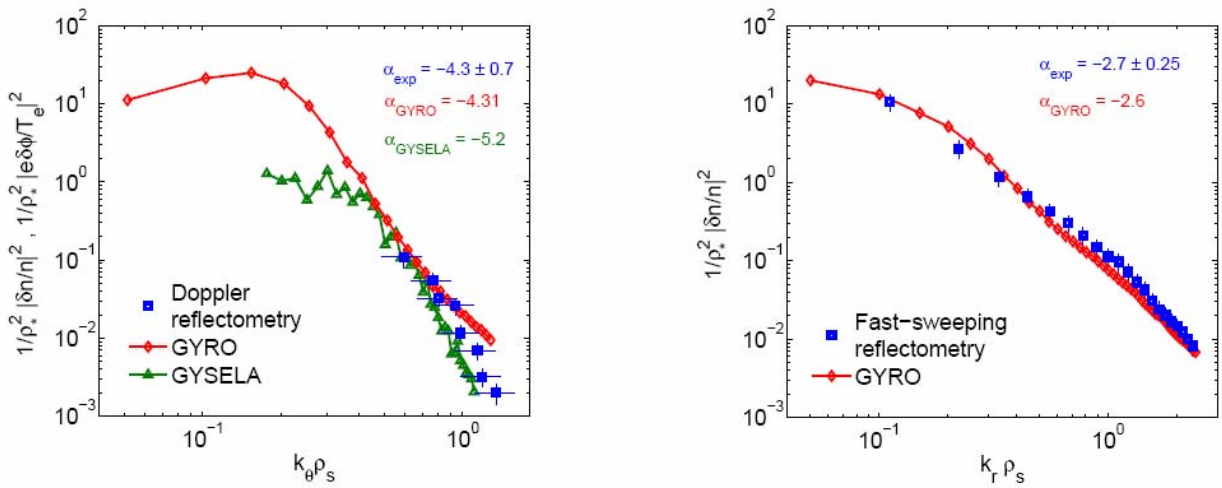

Figure 4-9: Comparison of measured and simulated spectra. The poloidal wavenumber spectrum (left) is measured with Doppler reflectometer and the radial wavenumber with fast sweeping reflectometer (right).

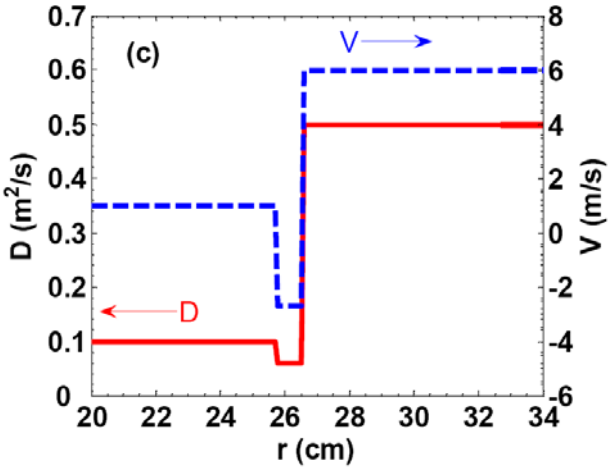

Figure 4-10: Diffusion coefficient and convection velocity profiles in the situation of a spontaneous particle transport barrier in the HL-2A tokamak.

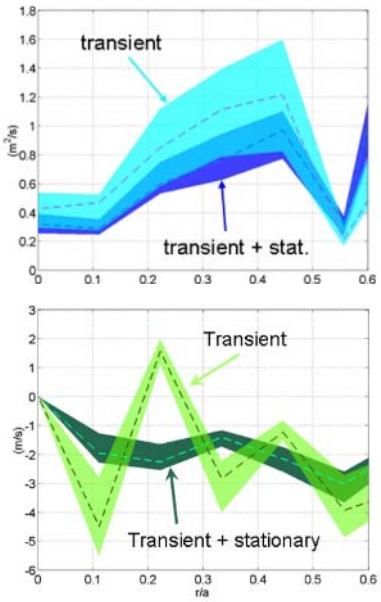

Figure 4-11: Diffusion coefficient (top) and convection velocity (bottom) from the analysis of a supersonic injection alone (light colours) and of the same injection combined with a stationary level of impurity.

4.3 Heating and Current Drive

4.3.1 Tore Supra

Wave-plasma interactions in the plasma edge were documented during the tests of an ITER-like ICRF antenna prototype (ILA) on TS in 2007 [56]. Heat loads on the ILA and RF-induced modifications of LH wave launcher properties suggest localized DC biasing of the plasma in the upper and lower parts of the ILA. Near field simulations with the antenna code TOPICA could reproduce this trend and interpret it in terms of parallel RF currents flowing on ILA frame [57]. Figure 4-12 shows the poloidal variation of the real part of the RF voltage

for the ILA antenna and two modified designs aiming at reducing the high voltage in the corners. On left of the figure, a sketch of the ITER-like prototype antenna indicates that main contributions of V_{RF} are originated from top and bottom antenna box (quite)-parallel structures specifically where image parallel currents $j_{//}$ circulation is favoured. This study provides guidelines for subsequent front face optimisation.

Electron density profile of the ITER scrape-off layer is uncertain and modelling predicts that the ICRH cut-off layer could be too far from the antenna to ensure a high enough coupling resistance for coupling 20MW with one antenna. Gas injection from a pipe magnetically connected to the ICRH could increase locally the density provided ionization is enhanced by ICRH power dissipation in the SOL. Coupling resistance was compared for two cases of gas injection: i) poloidally distributed gas inlet module ('local fuelling') magnetically connected to the antenna ii) mid-plane gas inlet module ('standard fuelling') not connected. Density was feedback controlled to $n_l=4.85\times10^{19}m^{-2}$ in both cases. No significant difference is observed in the coupling resistance for a wide range of antenna strap to LCFS distance d. However the cut-off layer is, on Tore Supra, in the confined plasma or the 'close' SOL whereas on ITER this layer will be more peripheral and local fuelling could be more effective.

The new LH coupling code ALOHA was upgraded (Aloha_v2): CVS management, replacement of Fortran77 by Fortran95 and of NAG libraries by free libraries, user-friendly interface. The 2D version of the code, which allows taking into account the full structure of the electromagnetic modes in the plasma, was validated [58]. LH coupling measurements was compared in details to code predictions. In particular, the shadowing effect of the antenna guard limiters, which can be precisely described by considering a two-layer plasma model with different gradients, has been evaluated [59]. As predicted a minimum of the Reflection Coefficients (RC) of the MarkII launcher is found on several modules when $n_e\sim3\times10^{17}m^{-3}$ (Figure 4-13). This effect, not observed and not predicted on MarkI antenna, is attributed to the passive waveguides inserted between the modules.

The 2006 and 2007 campaigns have been particularly intense in terms of injected power and energy but an increasing operational difficulty has emerged, limiting the high power and long pulse performance. Analysis reveals that the MARFE and disruption in several cases was preceded by a localised hot spot in a region of thick carbon re-deposition on the Toroidal Pumped Limiter [60]. This component was therefore completely cleaned during the winter shutdown 2007 (see section 2.1.1). This thorough cleaning procedure had a dramatic impact on the performance in terms of coupled power. Less than 10 discharges were needed to attain an injected power level of >10MW without a single disruption. In a subsequent day, the same power level could be attained in three discharges. No signs of the previously observed phenomenology (hot spot, flaking, MARFE, disruption) have been observed during all the restart and power increase phase in 2008. A comparison of this power increase phase with one the presence of carbon deposits (2006) is shown in Figure 4-14, left. The filled points correspond to discharges ending in disruption or early termination. In 2008, the power increase phase was much faster, due to the absence of disruptions. As a result, a discharge with close to 12MW of total heating power with duration of 10s has been obtained, which is a new record for Tore Supra (Figure 4-14, right).

4.3.2 CEA participation to JET-EFDA experimental campaigns (Task Force H)

The EURATOM-CEA Association has participated to the 2008 EFDA-JET experimental campaigns in the frame of Task Force H for the following topics: commissioning of the ITER-like ICRF antenna (ILA), modification of the plasma edge by RF sheath effects, high level commissioning of the LHCD antenna, effect of gas injection from different Gas Introduction Modules on LH coupling and CD efficiency, determination of LH power deposition at different plasma densities.

- Commissioning of the ILA on JET plasmas from May 2008 to March 2009 in different conditions (33, 42 and 47 MHz, L- and H-mode, antenna strap-plasma separatrix distances of ~9 to 17 cm) has provided relevant information for future antenna design and operation. New tools to aid the functioning of the ILA have been developed and optimized during the commissioning: feedback control for matching of mutually coupled resonant double loops, feedback control on the phases of the strap voltages, prediction tools for the setting of impedance transformer to optimize the ELM resilience, arc detection system based on sub-harmonics analysis. The maximum power density achieved has been 5.9 MW/m^2 in L-mode with strap to plasma separatrix distance of ~ 9-10 cm at 42 MHz on the lower half of the ILA and limited by problems on the RF generator. This extrapolates to 8 MW/m^2 if the full generator power were available. Efficient (trip-free operation) ELM tolerance was obtained both at 33 and 42 MHz on a large range of ELMs with strap voltages up to 42 kV and a maximum power density of 3.9 MW/m^2.
- Plasma flow induced by RF sheath effects was documented by analyzing the RCs of the LHCD antenna. As expected the coupling of the bottom part of the LH antenna which is magnetically connected to the upper part of the ILA is affected and the RCs of the three bottom rows of waveguides increase for P_{ILA}>1MW indicating a decrease of the edge density. Modification of the density in the flux tubes connected to the LH and ILA antennas was also deduced from the infrared thermography data.
- High level commissioning (HLC) of the LHCD antenna has been extended over the standard JET requirements (4MW in L-mode). 5.5 MW (22.5MW/m^2) has been coupled for the preset time (1s) in L-mode (Figure 4-15). The reference value of power density (25MW/m^2) for the design of the ITER LHCD system was achieved for 4 rows of waveguides over 6. Only three sessions (23 pulses) were dedicated to the HLC in H-mode and coupled power did not exceed 3MW. Metallic impurity release and arc detection from bolometry diagnostic were investigated. It is concluded that this diagnostic on JET is reliable for arc detection and its use could be improved for further reduction of metal contamination and disruptions [61].
- Gas injection from a poloidally distributed module (GIM6) located near the LH antenna is very efficient to raise the density in the SOL and improve the LH coupling [62]. Other gas injection modules, identical to the configuration of the ITER gas injection system, were tested. No effect on the LH coupling was observed but this could be the result of the position of the gas injection with respect of the separatrix and an other experiment has been proposed with a more favourable magnetic equilibrium. No deleterious effect on CD efficiency is observed when gas is puffed from GIM6, at least for gas rate lower than $6x10^{21}$el./s [63].

- LH power deposition was investigated from high frequency (~40Hz) LH modulation experiments (B_t=2.7T). LH power de position is found to be centred on a normalized radius varying between r/a~0.7 and r/a~0.9 when the line-averaged density is increased from 1.7×10^{19} (in L-mode) to $4.2\times10^{19}m^{-3}$ (in H-mode). Modelling of the LH wave propagation/absorption with the C3P0/LUKE code was performed for several pulses. The wave undertakes several reflections on the wall until the parallel index is sufficiently up-shifted to be absorbed (Figure 4-16 left).. These simulations indicate that most of the power is absorbed in the outer part of the plasma (Figure 4-16, right)

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Figures

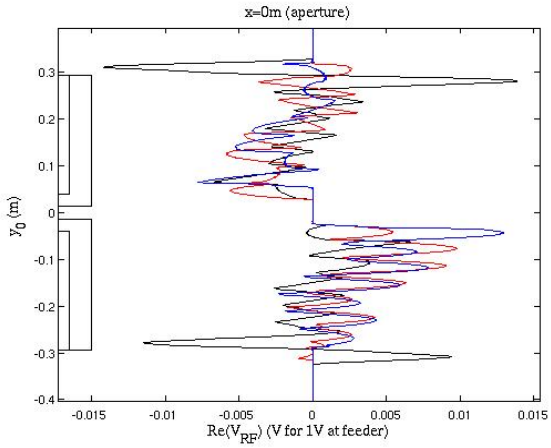

Figure 4-12: Re(V_{RF}) for three different antennae: Tore Supra ITER-like prototype (black) model, and two modified models with box structure following the magnetic field lines. V_{RF} is evaluated at the antenna aperture (virtual surface symbolizing the interface between antenna and plasma) and is plotted versus the altitude y_0 where tilted (7°) field lines intersect the antenna middle-plane (poloidal, radial).

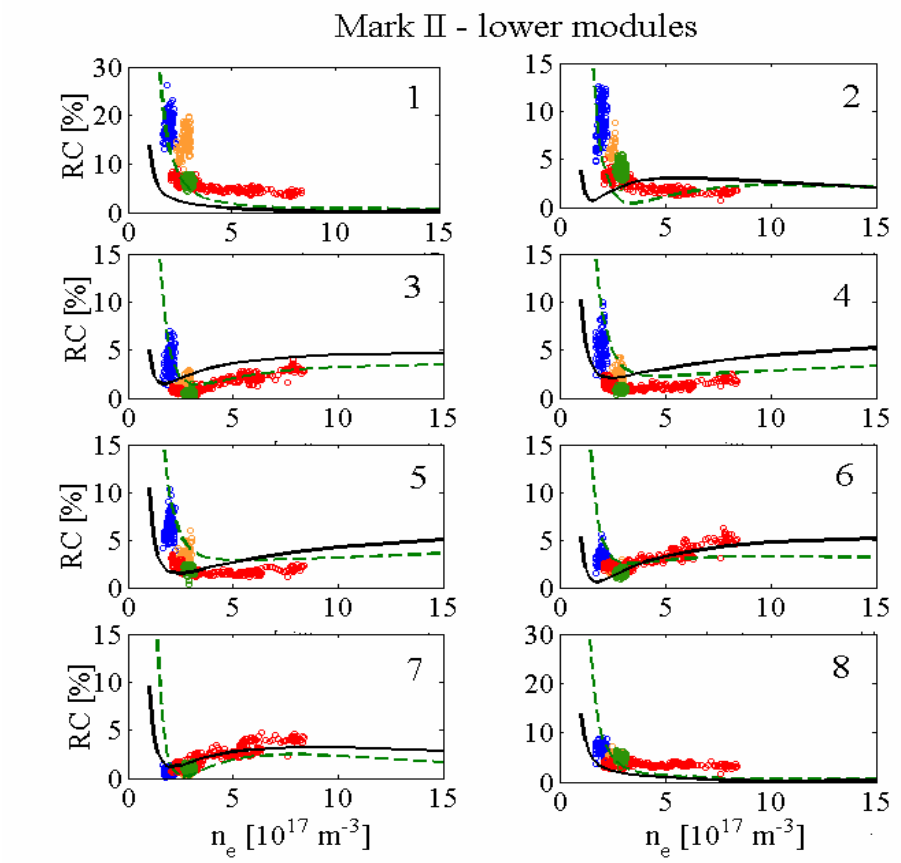

Figure 4-13: RCs of the 8 upper modules of the MarkII launcher versus the edge electron density. Results of the ALOHA code are plotted for two density profiles: 1 gradient with λ_n=20mm (dotted lines) and 2 gradients with λ_n=2mm near the launcher and λ_n=20mm further (solid line)

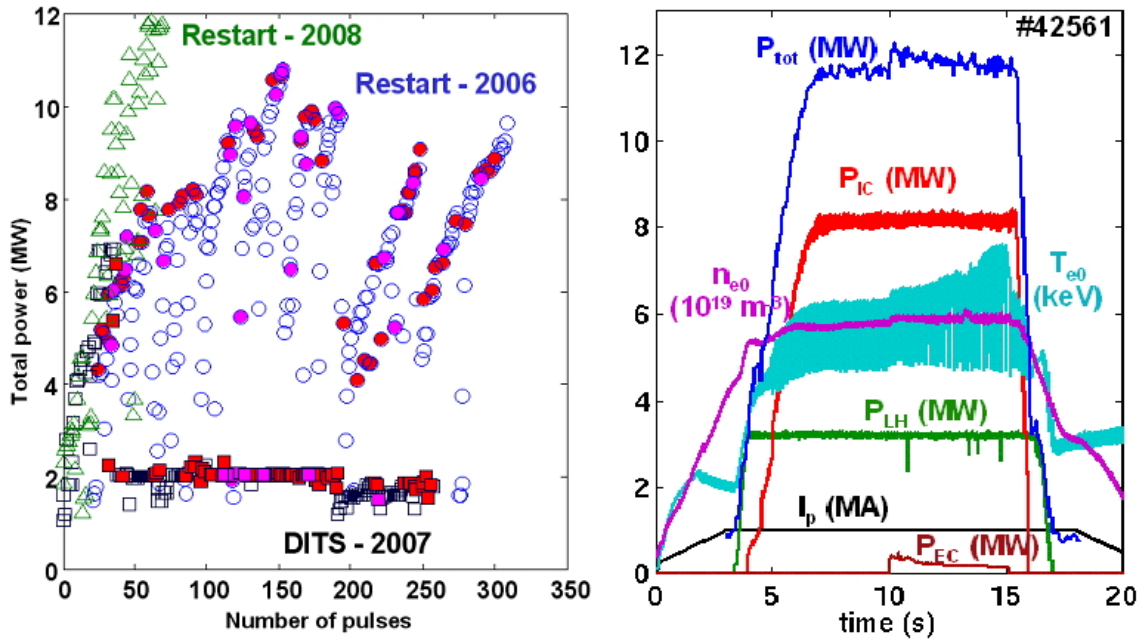

Figure 4-14: Left: injected power versus number of pulses for two plasma restart campaigns: Before cleaning (2006) and after cleaning (2008). Right: time evolution of plasma parameters in the highest power discharges obtained in 2008, with close to 12MW during 10s.

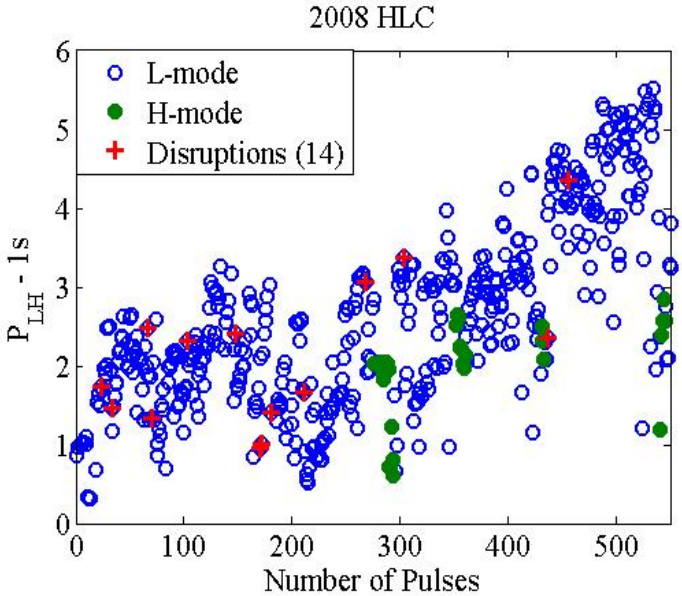

Figure 4-15: high level commissioning of the JET LHCD antenna. Include LH pulses, which were not dedicated to this programme.

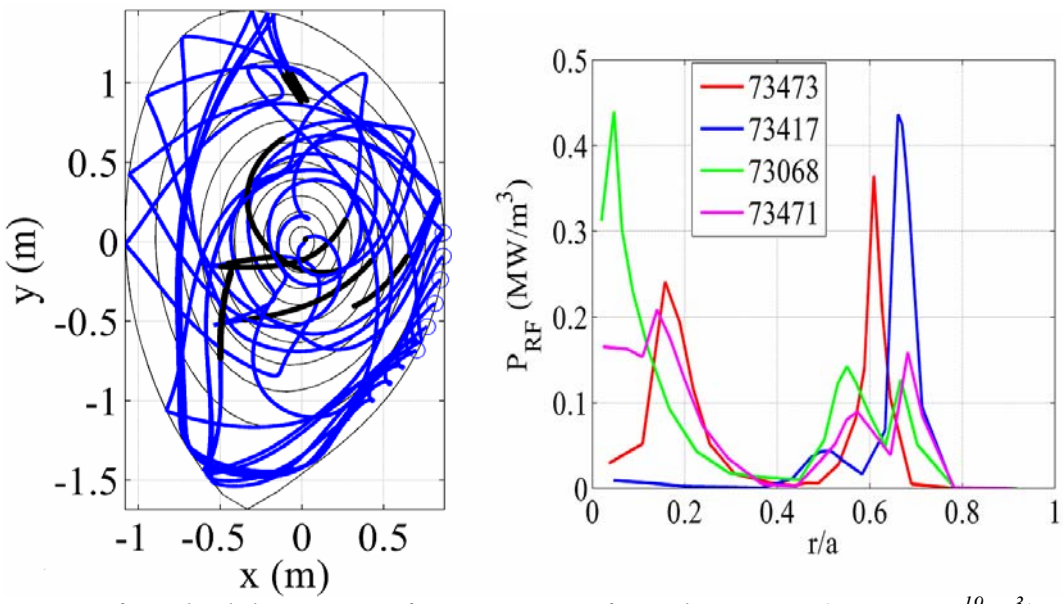

Figure 4-16: *Left: poloidal cut view of a ray tracing for pulse 73417 (n_l=2.8×10^{19}m^{-3}). Most of the wave absorption occurs along the black path. Right: power density deposition profiles for pulse 73473 (n_l=1.7×10^{19}m^{-3}), 73417 (n_l=2.8×10^{19}m^{-3}), 73068 (n_l=3.8×10^{19}m^{-3}), 73471 (n_l=4.2×10^{19}m^{-3}).*

4.4 Plasma wall interaction

4.4.1 Fuel retention

The DITS project (Deuterium Inventory in Tore Supra), aimed at better understanding fuel retention in tokamaks, has the two following objectives:

- comparing the deuterium inventory deduced from particle balance and post mortem analysis,
- identifying the retention mechanisms at stake.

After a dedicated experimental campaign performed in 2007, during which the deuterium (D) inventory was monitored by particle balance, the first phase of the post mortem analysis took place in 2008, on a restricted set of 10 tiles, out of the 40 extracted from the main plasma facing component, the toroidal pump limiter (TPL). Those 10 tiles were chosen in the three zones of interest identified on the TPL: the erosion zone, the thin deposits and the thick deposits, as shown on Figure 4-17.

In complement to TDS (thermodesorption), giving the global D content of the samples, local measurements, such as NRA (Nuclear Reaction Analysis) and SIMS (Secondary Ion Mass Spectroscopy) were carried out, giving additional information on the D profile within the samples for NRA, as well as on impurities for SIMS. These analyses were performed in collaboration with European partners (IPP Garching, VTT in Finland), in the frame of the EU PWI TF. The comparison of the sample D content deduced from NRA and TDS has proven to be consistent, showing that the thick deposits layers contain D beyond the NRA probing range (30-40 μm). The SIMS data are still under analysis. From a combined analysis of TDS and NRA data, it was possible to derive an integrated D inventory for the TPL, with the contribution of both the tile surface and the gaps in the different zones of interest, as shown in Figure 4-18.

As far as retention mechanisms are concerned, this first phase of post mortem analysis would lead to 90 % of the wall inventory related to codeposition (in the thin and thick deposits zone as well as in the gap deposits of the erosion zone). Gap deposits contribute significantly to the global D inventory (38% of the total), in particular in the erosion zone. However, it remains that the D concentration in the tile surface in erosion zones is still significant and higher than expected from simple implantation, leading to 10% of the total D inventory.

The total wall inventory deduced from post mortem analysis of the TPL at the present stage of the study is ~1.5 10^{24} D (error bars estimated to be ± 20%), corresponding to ~ 5g of D trapped in the TPL. This accounts for ~50% of the 3.1 10^{24} D derived from the particle balance, and represents a significant progress with respect to previous studies, which concentrated on deposited layers and did not include yet the gap contribution.

In addition, the microscopic structure of the deposits was studied, in collaboration with the LPIIM laboratory in Marseille, showing elongated tip structure in the poloidal direction.

Those results were presented as oral papers in the PSI [64] and IAEA 2008 [65] conferences. In order to progress further, future work will include the analysis of the remaining tiles to consolidate the present balance, as well as the analysis of deposited layers not yet studied, in particular on the TPL leading edge as well as on other PFCs, such as the inner bumpers.

4.4.2 Carbon balance

In order to obtain a coherent picture of the retention processes, efforts have started to derive the global carbon balance associated with codeposition, using complementary measurements (visible spectroscopy for gross erosion, confocal microscopy for net erosion, PFCs cleaning for net redeposition), as well as modelling, in collaboration with the LPIIM laboratory and FZJ.

Dedicated experiments have been performed to refine the scaling law of gross carbon production versus power coupled to the plasma. Results are still under analysis. However, using the previously established scaling law for plasma operation, as well as estimates for the contribution of disruptions and conditioning procedures, the total carbon gross erosion was estimated to be ~ 2 kg for the whole CIEL duration.

From the extensive cleaning of the PFCs performed after the DITS campaign, the net redeposition was estimated to be around 800 g of carbon, roughly consistent with the net erosion (~1-1.4 kg) as estimated from surface mapping obtained from confocal microscopy, shown on Figure 4-19.

Preliminary applications of these results would show that roughly 10 μm (respectively 1 μm) of the thick (respectively thin) deposited layers could be attributed to the DITS campaign. This would correspond to 2/3 of the D content found in post mortem analysis. More work is needed to confirm these data, in particular using the SIMS analysis to locate the markers (^{13}C and ^{11}B) that were injected at the beginning of the DITS campaign.

4.4.3 Measurements of SOL temperatures

SOL T_i was measured for a first time in detached plasma. $T_i \cong T_e$ (both below 10 electron volts in the SOL) was recovered (Figure 4-20). Thermal equipartition was found to be consistent with relatively strong ion-electron collisionality. In addition, the transition to detached state was found to be related with the flattening of the SOL profiles and a drop of both, the electron and ion saturation current densities in the SOL by about 30% [66].

4.4.3.1 Scaling of SOL T_i and T_e in the ICRH and LH heated plasma:

The aim of this experiment was to measure SOL T_i and T_e in gradually increasing ICRH and LH power. The main questions addressed in this experiment were following. How does the SOL T_i and T_e scale with P_{ICRH} and P_{LH}? Do the SOL temperatures depend on a heating method? In addition, it was planned to repeat the measurements for three different values of the plasma density in order to vary the fast ion orbit loss from the edge plasma and compared it with the theoretical prediction. q=8 (I_p = 0.54 MA, I_T = 1250 A), at which the perturbing effect of the LH antenna on the RFA T_i measurements is reduced, was chosen. In addition to the scaling with P_{ICRH}, unexpected and interesting scaling of SOL T_i and T_e with the radiated power fraction was obtained.

Figure 4-21 shows the radial profiles of SOL T_i and T_e measured for different P_{ICRH}. A factor of 3 increase of P_{ICRH} (open symbols) coincides with the increase of SOL T_i by only ~50%, which is smaller compared to the scaling of SOL T_i with P_{ICRH}, suggested from earlier measurements. However, it must be taken into account that within the range of P_{ICRH}, P_{rad} = 1→1.5MW (earlier measurements were characterized by much smaller f_{rad}), indicating that the increase of P_{rad} could decrease SOL T_i. This is clearly evident from almost a factor of 2 different SOL T_i measured at fixed P_{ICRH} (0.9 MW) but different P_{rad}. The influence of P_{ICRH} and, surprisingly, P_{rad} on SOL Te was found to be much weaker.

Similar effect of P_{rad} on SOL temperatures was obtained from the modelling of T_i and T_e just inside the LCFS (referred to as "edge") using a power balance model (collaboration with P. Tamain, UKAEA). These results suggest that the decrease of SOL T_i with the increase

of f_{rad} could be explained by the increase of the edge ion-electron collisionality, caused by the drop of edge T_e due to radiation. Numerical modelling with the fluid code TECXY is in progress.

4.4.3.2 Simultaneous measurements of SOL Ti by RFA and CXRS

This experiment was aimed at independent experimental validation of RFA T_i measurements, which was not performed in any tokamak before. A special plasma configuration in which the most outer CXRS viewing lines intersect the SOL was proposed. The measurements were performed separately, although in very similar ohmic plasma configuration.

Figure 4-22 shows the radial profiles of T_i measured by RFA and CXRS. The RFA T_i increases towards the LCFS, reaching ~45 eV at the LCFS. The CXRS T_i decreases with the increase of radius in the confined region (with T_i = 200 eV at the LCFS) and increases with radius in the SOL. A factor of 4-5 difference in RFA and CXRS T_i almost certainly cannot be explained by the difference in the plasma conditions in both discharges (in ohmic plasma, RFA T_i does not exceed 100 eV in any configuration explored up to now) or by the uncertainty in ion charge. In addition, such difference cannot be explained by the difference in temperatures of impurities and fuel ions, which are expected to be at thermal equilibrium at the densities studied here. Also difficult to explain is the increase of CXRS SOL T_i with radius. Although the uncertainty in RFA T_i measurements may play a role here, this experiment shows that the analysis of the CXRS spectra measured in the SOL might be very complex (the spectral line widths might not be Doppler dominated), and indicate that CXRS may not be appropriate for SOL T_i measurements as it is widely believed.

4.4.4 Wall conditioning with ICRH

In ITER, the presence of permanent high magnetic field will prevent the use of conventional DC-glow discharge conditioning, the efficiency of the latter being drastically reduced. Conditioning techniques fully compatible with the magnetic field have to be developed, which allow to limit the flux of impurities coming from the walls and ease plasma ignition, to control the desorption of hydrogen and hydrogen isotopes (H, D, T) and finally to minimise tritium inventory in ITER in surface layers of plasma facing components - one of the major issues for ITER operation. Among the alternative conditioning techniques, discharges created at the Ion Cyclotron Resonance Frequency seem to be the most promising one.

Ion Cyclotron Wall Conditioning (ICWC) experiments have been carried out in Tore Supra, in order to assess the efficiency of this technique for fuel removal and recovery after disruptions. The ICRF discharges were operated in He/H_2 mixtures at Tore Supra nominal field (B_T=3.8T) at a RF-frequency of 48 MHz, which is within the range of ITER RF frequencies. Both continuous and pulsed mode operations were performed, using the unique capability of Tore Supra to sustain long duration discharges. The ICRF power was applied through a standard two straps antenna, without any change in the hardware. RF powers ranging from 30kW to 150kW were used in order to avoid too high reionization of desorbed species. The antenna was operated either in π or 0-phasing, with a noticeable improvement of the RF-coupling in the latter case. The outgassing of hydrogen and deuterium, as well as the

implantation of H into the walls, was measured by means of mass spectrometry. Two main results were obtained:

- in order to modify the wall isotopic ratio, the ICRF discharge was operated in a continuous mode. The D outgassing was found to increase with the hydrogen percentage in the mixture at low RF power. After 15 minutes of ICWC in He/H_2 the isotopic ratio measured during ohmic reference shots was altered from 5% to 50%, at the price of an important H implantation into the walls (Figure 4-23).
- the capability of ICWC to recover from disruptions was tested. The disruptions were provoked at high current on the outboard poloidal limiter. Only using pulsed ICWC in pure Helium, two out of two successive attempts were successful, whereas plasma initiation would not have been possible without conditioning (Figure 4-24).

4.4.5 Disruption mitigation studies

Disruptions are a major issue for future tokamak reactors. They generate strong heat deposition on PFCs, induce currents and electromagnetic forces in the vessel structure, and accelerate runaway electron beams up to several MeV. Massive Gas Injection is one of the techniques currently studied to minimize theses three effects. Using Tore Supra's massive gas injector, different noble gases were compared: Helium, Neon and Argon. Electromagnetic effects are reduced with all the three gases. Helium is very efficient regarding runaway electron suppression, on the contrary to Neon and Argon (Figure 4-25). Up to 30% of the thermal energy is radiated by the gas before the final loss on PFCs. Some of those results are confirmed by JET similar experiments. Helium injections were also performed on different plasmas, to assess gas mixing efficiency and its dependencies to the plasma conditions. The penetration of the cold front was measured using the newly installed fast framing camera. Preliminary analysis shows that the depth of penetration may be linked to the safety factor profile rather than the temperature or density profile inside the plasma. The q=2 surface seems to play an important role regarding gas jet mixing.

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Figures

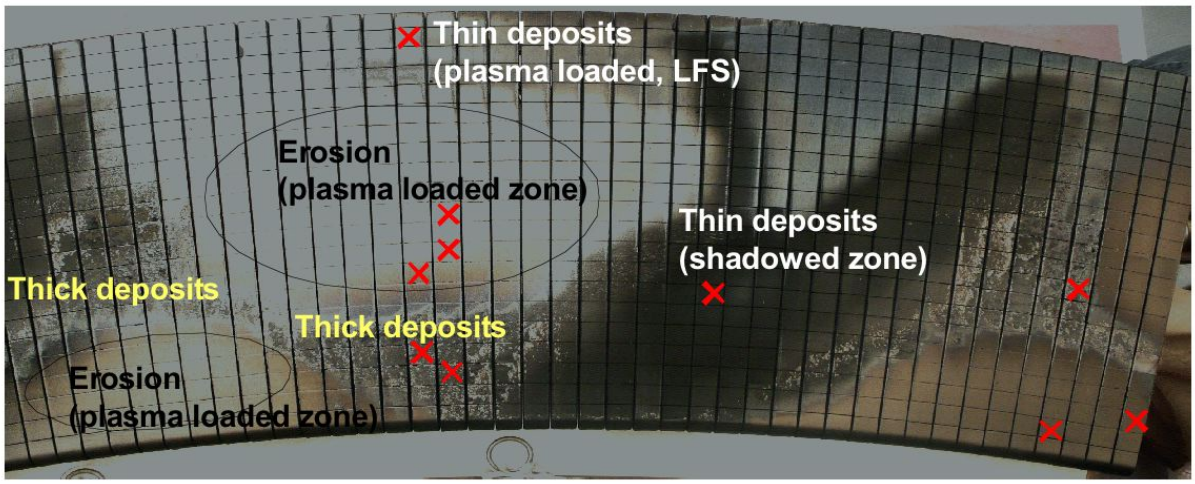

Figure 4-17: sector Q6A of the Tore Supra limiter, extracted after the DITS campaign. The different zones of interest are indicated as well as the 10 tiles analysed during the first post mortem analysis campaign (crosses).

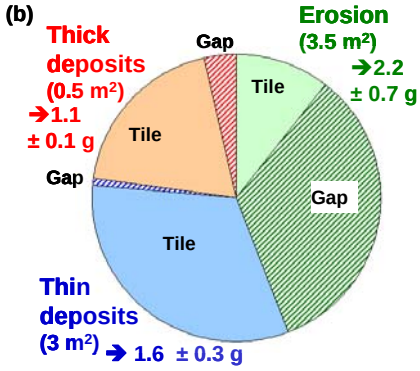

Figure 4-18: contribution of the different zones (erosion, thin deposits, thick deposits) to the overall wall inventory (in g of D) deduced from post mortem analysis, with corresponding error bars. For each zone, the contribution of the gap deposits and the tile is indicated.

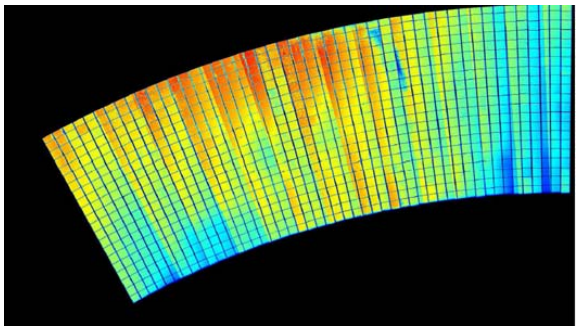

Figure 4-19: preliminary reconstruction of the surface mapping of a TPL sector by confocal microscopy. The leading edge is not included.

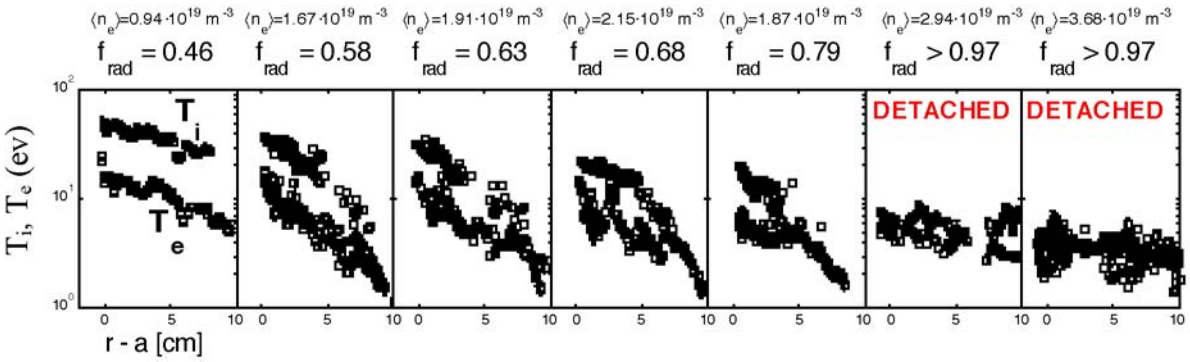

Figure 4-20: Radial profiles of SOL Ti and Te measured for a broad range of plasma central-line-averaged densities and radiated power fractions. Last two columns show Ti and Te measured in detached plasma.

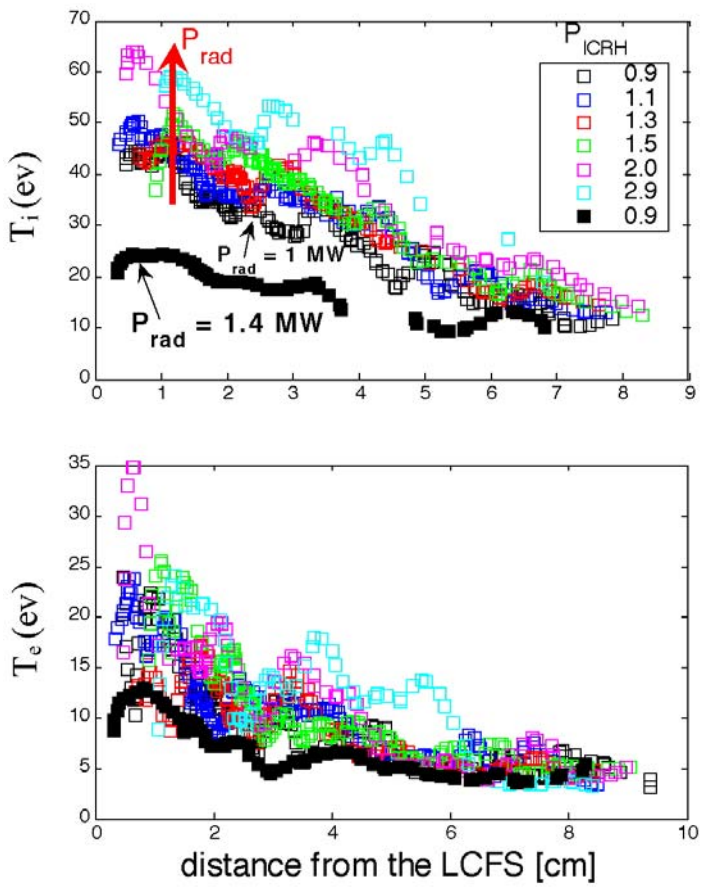

Figure 4-21: Radial profiles of SOL Ti and Te measured at gradually increasing ICRH power P_{ICRH}. Profiles are smoothed for clarity. Ip=1 MA, It = 1250A, <ne> = 5.3→5.6 10^{19} m-3. Open symbols: P_{ICRH} = 0.9→2.9 MW, P_{rad} = 1→1.5 MW. Full symbols: P_{ICRH} = 0.9 MW, P_{rad} = 1.4MW.

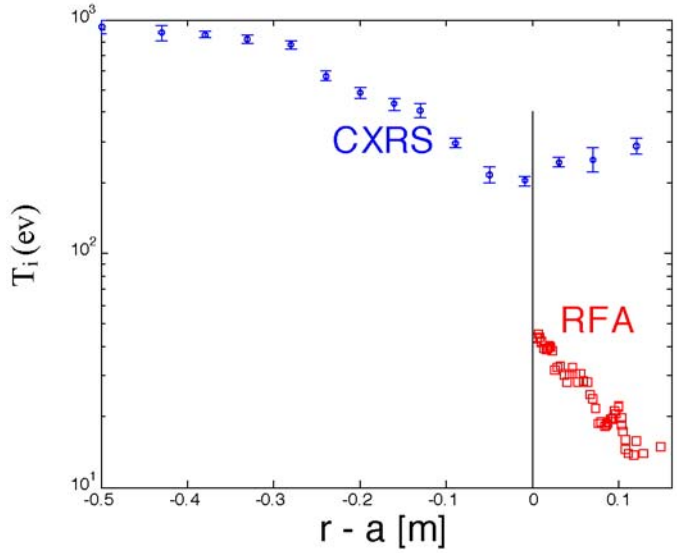

Figure 4-22: Radial profiles of T_i measured by RFA and CXRS in ohmic discharges with similar R and a (RFA: #42390, CXRS: #43533). LCFS is at r - a = 0.

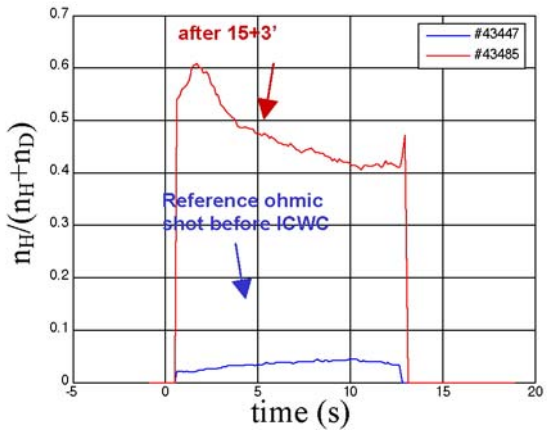

Figure 4-23: isotopic ratio during two reference ohmic shots before and after 15 min ICWC.

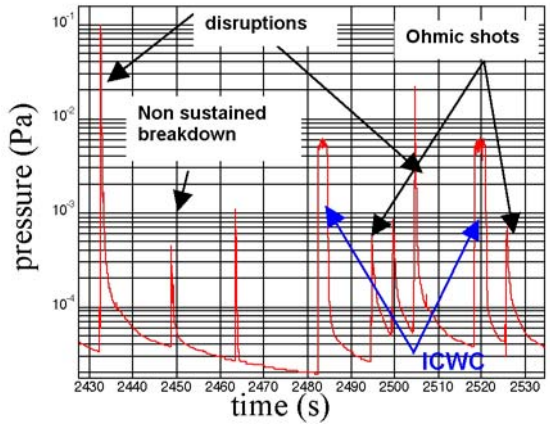

Figure 4-24: torus pressure during the assessment of ICWC efficiency for recovery after disruptions.

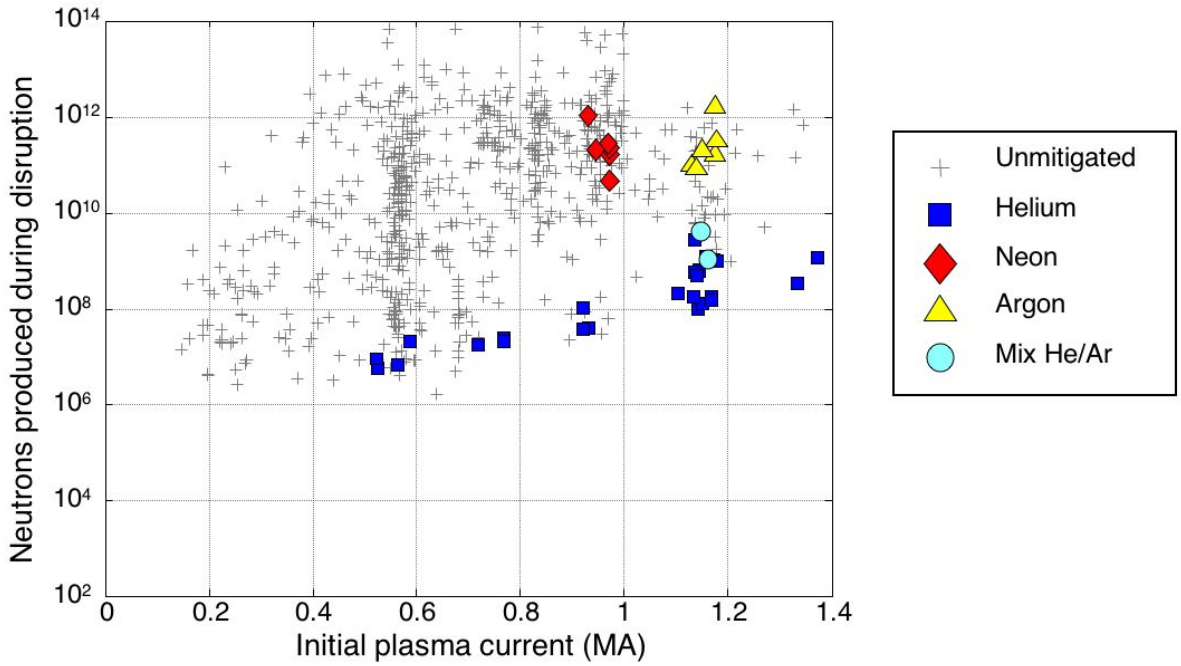

Figure 4-25: photo-neutrons (signature of runaway electrons) produced during mitigated and unmitigated disruptions on Tore Supra.

5 Technology developments

5.1 Plasma engineering

5.1.1 Diagnostics

5.1.1.1 Development of a fibre optic current sensor (FOCS) suitable for ITER

Steady-state operation of next step fusion devices, together with the high radiation level and temperature environment, will challenge the use of classical inductive magnetic sensors. Such equipments are crucial for tokamak operation and plasma current measurements. One possible complementary technique involves Fibre-Optic Current Sensor (FOCS), using pure silica single mode fibre (clad diameter 125µm), the sensing principle being based on polarization light rotation by Faraday effect in silica fibres. Such a fibre is intrinsically well insulated, insensitive to electromagnetic interference, and has a large dynamic range from DC up to about a few MHz. This makes FOCS attractive and compact sensors, although fibre protecting and guiding tubes are mandatory. In order to validate the technology of FOCS in a tokamak environment where the current to be measured is huge (I_p > 1MA), a FOCS has been installed around the Tore Supra vacuum vessel in 2008. Two modes of operation have been investigated: (i) reflection measurement scheme with a Faraday Rotator Mirror (FRM) put at one extremity of the fibre, and (ii) transmission measurement scheme. A linearly polarized laser beam (λ = 1550nm) is launched into the fibre, and the change in the polarization ellipse during the plasma discharge is analysed in terms of Stokes parameters (azimuth and elongation). The azimuth of the initial linear polarization at the fibre exit is clearly observed to rotate by a certain angle that depends on the plasma current (Figure 5-1). In the reflection measurement scheme, the use of FRM allows to drastically stabilize the polarization states, cancelling the effect of the random fluctuations of the light passing through the fibre. Nevertheless, the polarisation rotation is not univocal with respect to the plasma current. The transmission scheme gives linear results, but the system would require a day-by-day calibration.

5.1.1.2 Fast camera imaging

A fast camera aiming at studying pellet dynamics, disruptions and edge turbulence, has been installed in Tore Supra at the beginning of the 2008 experimental campaign [67]. The detector is a monochrome Phantom v7.1 with a maximum resolution of 800×600 pixels at a frame rate of 4800 frames/s, the maximum frame rate being of 150000 frames/s. The collected light is transmitted and coupled to the camera detector through an actively cooled optical fibre beam, which can be moved to select detector regions of interest. Moreover, optical filters can be added to the system (e.g. interference filters). Data are stored using a gigabit Ethernet connection, each acquisition producing up to 1.3Gbytes of data. Multi-trigger acquisitions can be performed by splitting the camera memory. The system has been used intensively in 2008, in particular for FCE-assisted breakdowns, disruption mitigation by massive gas injection (Figure 5-2), and edge turbulence studies.

5.1.1.3 Multi-sensors Image Analysis

Imaging diagnostics have become increasingly important for plasma physics and tokamak operation [68]. All these diagnostics generate a large amount of data and need interactive tools for a user-friendly quantitative image analysis, data synchronization, regions of interest (ROIs) definition and edition, calculation on ROIs, advanced graph plotting, quantitative modelling [69]. In the case of multi-sensor analysis, for example surface temperature correction on to visible spectroscopy imaging, a geometric efficient superimposition is needed between infrared and visible images before correction. In the same way, the superimposition between infrared images on to the real geometry of the monitored plasma facing components (PFCs) helps to an accurate identification of the overheating zones before a physicist interpretation. At Tore Supra, an object-oriented platform dedicated to these multi-sensor quantitative image analysis tasks has been developed. The "WOLFF" project (Warping TooL For Fusion) is open-source and written in the last version of C++ (Figure 5-3). Up-to-date object-oriented concepts like "design pattern", have been used to easily adapt and extend WOLFF features to plasmas physicists needs. This high level of abstraction has allowed WOLFF to be adapted to the MAST (UK) imaging systems in only one week.

5.1.1.4 Equatorial Visible Infrared Wide Angle Viewing System in ITER Port Plug Equatorial 1 – Optical Design

After having examined the different concepts proposed on the previous EFDA task (2006) for the preliminary optical design of the diagnostic, one concept has been focused on: the so called "separated lines" solution (Figure 5-4). This concept exhibits several advantages regarding optical performance, reliability, manufacturability and maintainability [70]. The optimisation of this concept, in terms of image quality, port integration and maintenance, is in progress. Moreover, in order to identify the best dioptric optical materials and cope with ITER severe environment (neutrons and gamma radiations), collaborations involving CEA and optical material supplier companies are being organised, as well as a radiation test program.

5.1.1.5 Reflectometry system developments

Fast sweep reflectometry upgrade

Present estimation of turbulence correlation time is around 5-10 µs, which is 2 to 4 times smaller than the current Tore Supra reflectometer sweeping time (20 µs). In the frame of the StudyFus ANR contract aiming at studying the fine temporal dynamic of the turbulence, significant effort has been done to decrease the sweep time by a factor of ten for the V- and W-band reflectometers, thus increasing the frequency signal range up to 500 MHz (Figure 5-5). The reflectometer designs have been deeply renewed onto three points: sweep reproducibility and stability, frequency bandwidth detection and fast acquisition at 2GHz sampling frequency. Such an improvement should allow for the study of fine temporal dynamics of the density fluctuations occurring on disparate scales, in their two dimensional development and across the different plasma regions (core to edge).

JET reflectometers

In order to provide accurate density profile measurements on JET plasmas, a set of 6 reflectometers is planed to be installed. In May 2008, a contract involving CEA, IST and UKAEA has then been elaborated. The CEA is in charge of building 3 reflectometers to cover the frequency range 50-150 GHz (V-, W- and D-bands). The W-band reflectometer should be delivered early 2009.

5.1.1.6 Fast electron diagnostic development (DENEPR)

In the frame of a collaboration with IPPLM/IPJ Warsaw, a set of 4 Cerenkov detectors has been mounted in Tore Supra on the new internal shaft specially equipped with optical fibres. First measurements have been performed at the end of Tore Supra 2008 experimental campaign. As illustrated in Figure 5-6, the signal surprisingly appeared to be quite localised in the vertical direction, while the lost electrons are supposed to cover a wide zone above the plasma. Moreover, electron fluxes corresponding to two energy ranges only were detected, and not at exactly the same vertical position. In fact, two detectors only (corresponding to energy ranges E1>84keV and E2>109keV) were able to collect electron fluxes, at a particular incidence angle. The Cerenkov detector set has been likely overprotected from high electron fluxes, and a modification of the detector protection is foreseen before the next experimental campaign.

5.1.1.7 Tests of FIR Interferometer improved electronics at JET

Following the successful improvement of the Tore Supra FIR interferometer electronics (allowing for fringe jump correction at a 100kHz frequency rate, thanks to a FPGA embedded calculator), collaboration has been settled between CEA and UKAEA aiming at carrying out tests on JET with CEA electronics. In order to match JET data acquisition system requirements, the electronic cards have been adapted and the FPGA calculator algorithms have been modified by CEA. After laboratory commissioning performed at JET, the electronics has been routinely connected to real interferometer signals during plasma operation, and cross-comparisons are made in parallel with JET hardware (Figure 5-7). Assessment of the reliability and the efficiency of the fringe jump correction will be performed during the JET 2009 campaign, to decide of a possible renewing of the JET FIR system electronics in 2010.

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Figures

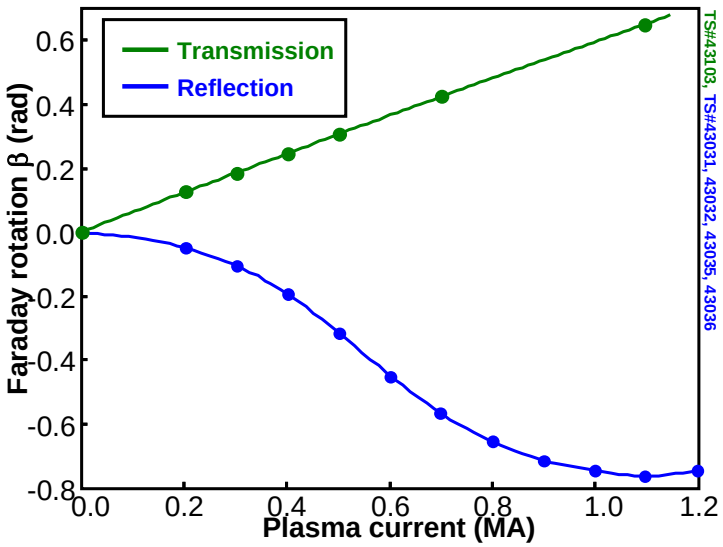

Figure 5-1: Faraday rotation versus plasma current for the transmission (green) and reflection (blue) FOCS operation modes

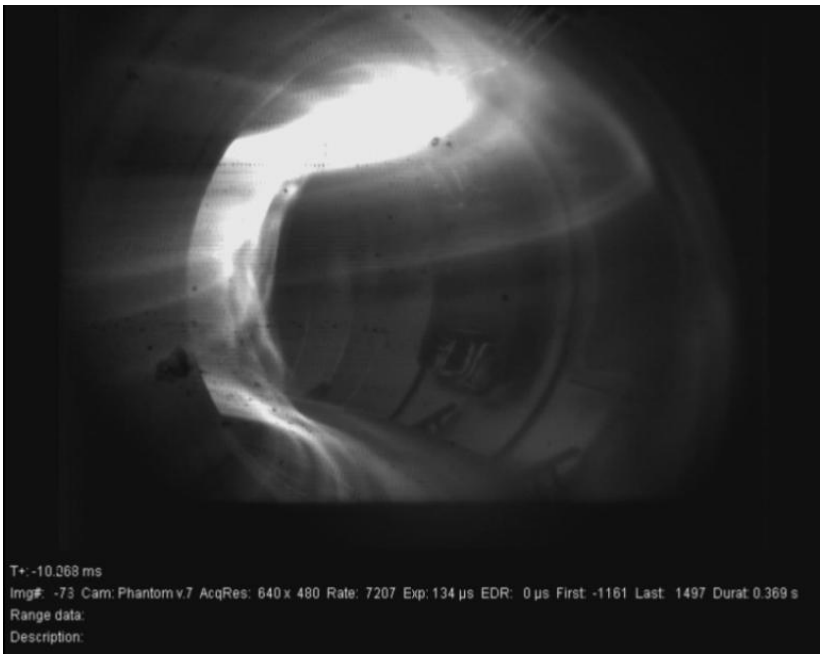

Figure 5-2: massive Argon injection during mitigation disruption experiment in Tore Supra

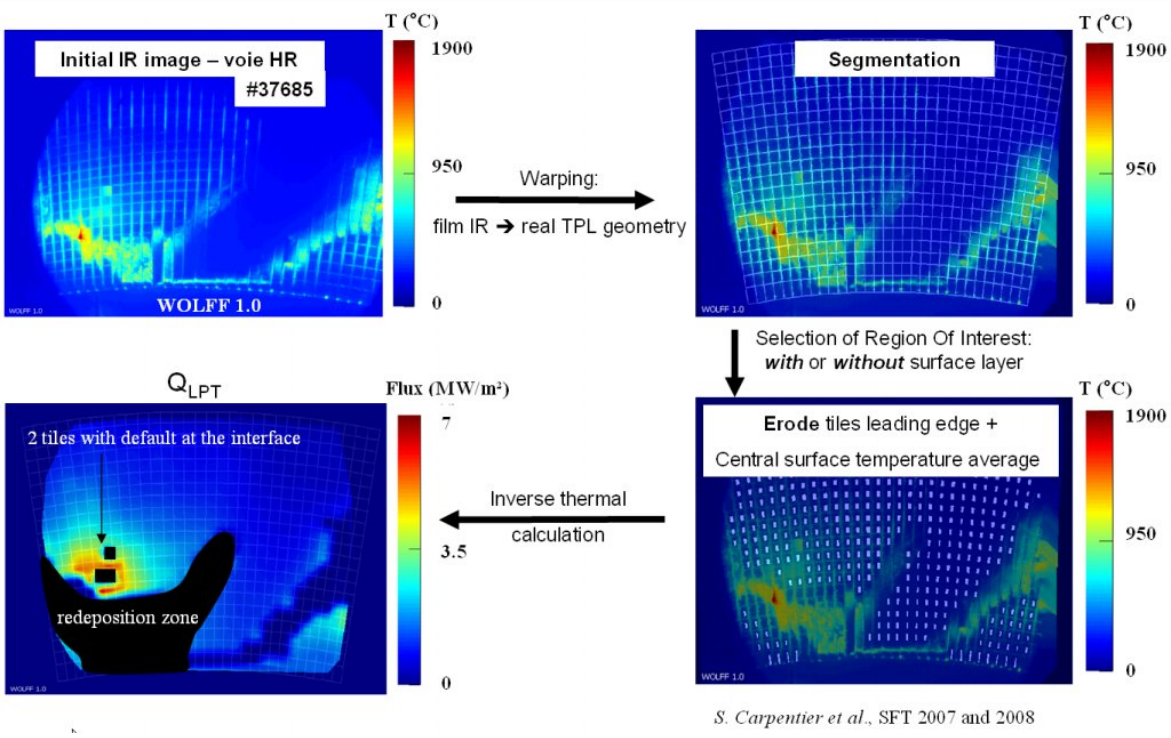

Figure 5-3: IR Toroidal Pumped Limiter analysis (TPL) with WOLFF. Upper left: IR TPL image. Upper right: IR TPL image superimposed on TPL real geometry. Lower right: detection and identification on TPL tiles with temperature extraction. Lower left: Power deposit calculation on TPL tiles using thermal inverse algorithm.

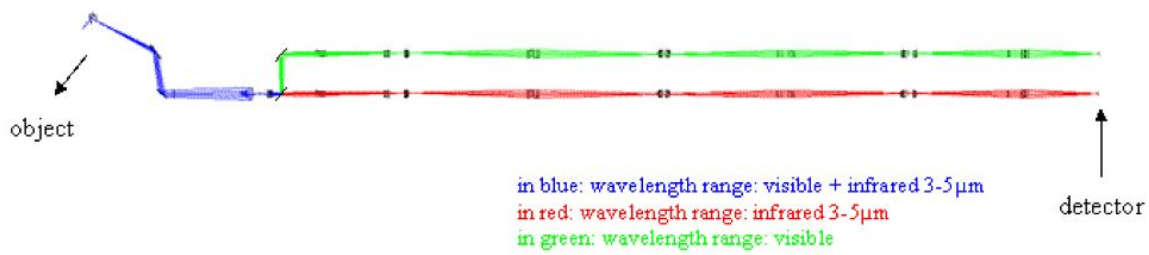

Figure 5-4: separated lines solution

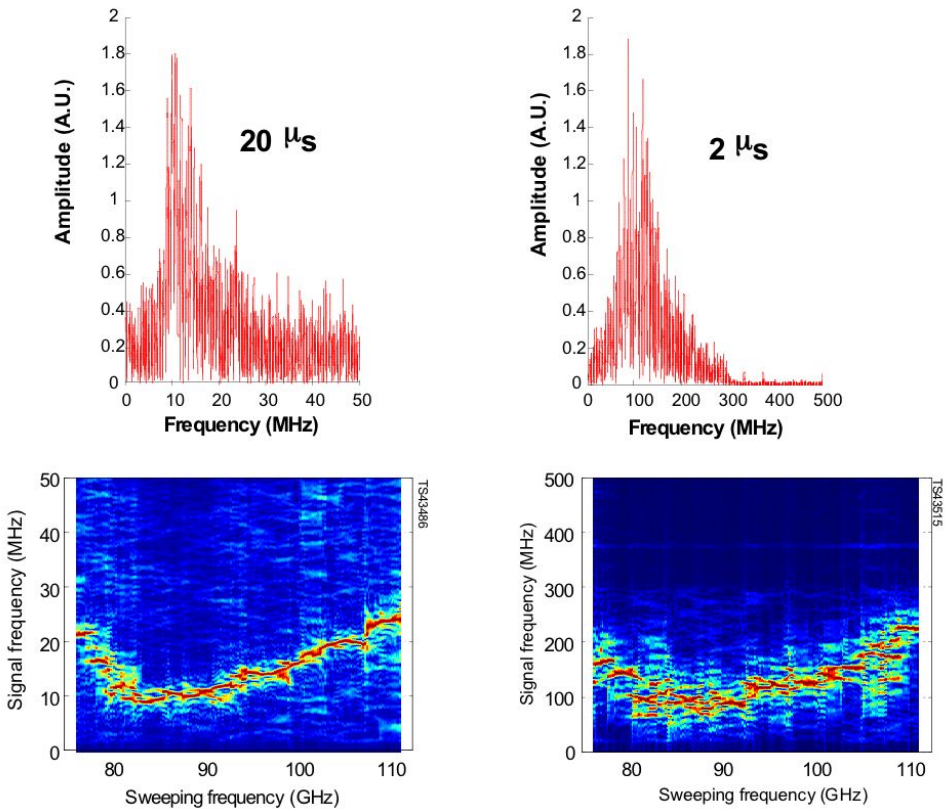

Figure 5-5: spectra of detected signals for 20 μs and 2 μs sweep times (top) and sliding FFT image colour of the frequency time trace evolutions (bottom). Plasma conditions were closely the same in both sweep time configurations.

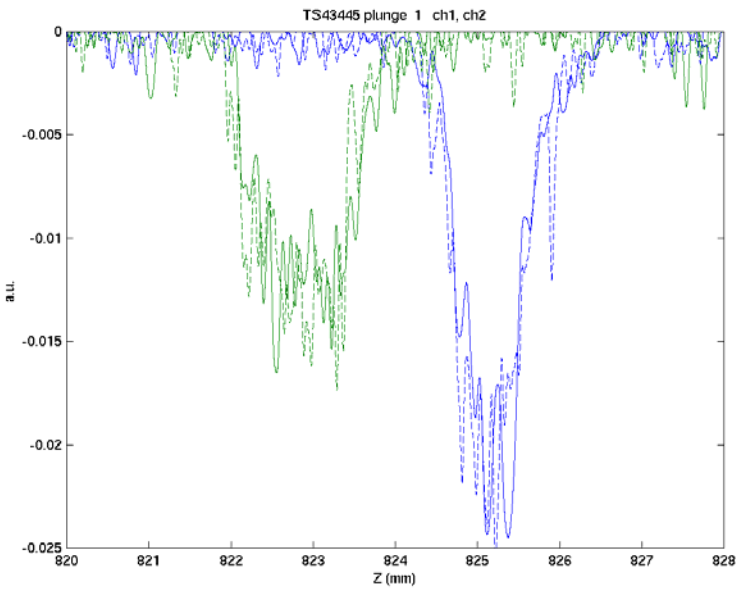

Figure 5-6: Cerenkov signal intensity as a function of the vertical position of two detectors: E1> 84keV (blue), E2>109keV (green)

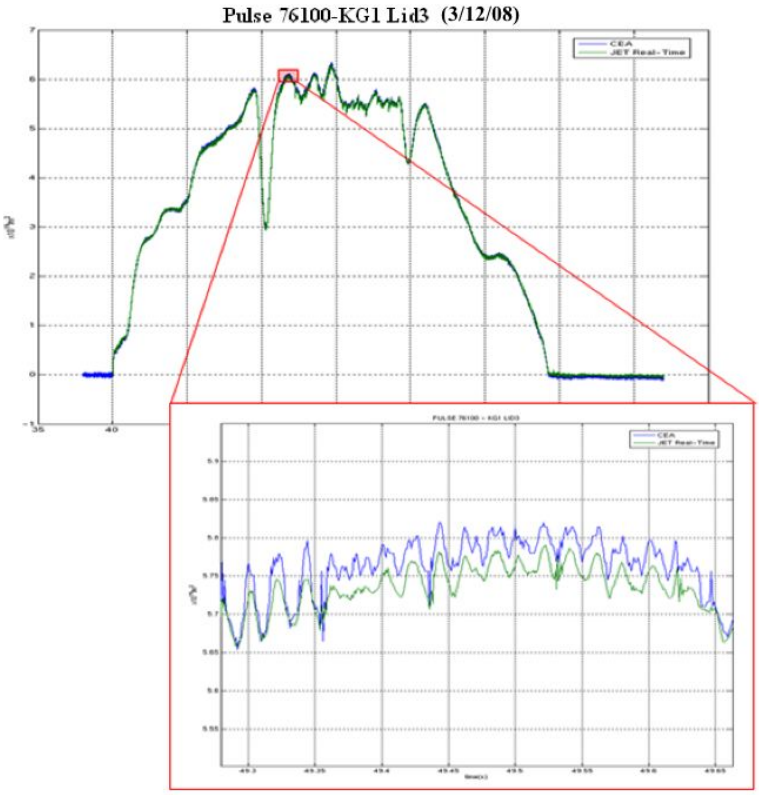

Figure 5-7: first density comparison test between JET real-time (green) and CEA electronics (blue)

5.1.2 Heating & current drive

5.1.2.1 Ion Cyclotron Resonance Heating

For the experimental campaign of 2008, the conditioning on plasma of the 3 ICRH antennas was made without difficulties, and the level of 8 MW total power achieved very rapidly, and 9 MW were obtained during the high power sessions. For the requirements of the programmatic actions AP 1 and 2, the power injection at different frequencies was made, for a total of 40 experimental sessions. In addition, some experiments using ICRH as a tool for the vacuum vessel conditioning were made. Mechanical studies have been made in preparation of the antennae maintenance of 2009 (for example: upgrade of the voltage probes). The first phase of test of the sliding RF contacts under vacuum has also been made at the end of the year (EFDA Task).

In the frame of the Cycle agreement, mechanical studies for the ICRH Antenna for ITER have been initiated. The “System Requirement Document” for the ICRH System for ITER has been written in the frame of an ITER contract.

Preliminary laboratory testing of ICRH antennas being a very useful step before their commissioning, we have tested the Tore Supra ITER-Like antenna with soda water and a maximum coupling resistance of 1.75 ohm/m has been obtained. Figure 5-8 shows the TS ITER like antenna connected to the water load. However, the antenna operational range is 2 to 6 ohm/m. To increase the coupling resistance up to 6 ohm/m, HFSS® and CST-MWS®

simulations have shown that loads need to exhibit much higher dielectric permittivity. To obtain such high values, two different approaches are possible [71]:

- Composite Materials: we have tested in laboratory several ceramic powders. The best result is for a Baryum Titanate (BaTiO3) powder with $\varepsilon_r \approx 500 + j\ 500$ at 48 MHz, leading to a calculated coupling resistance of 4 ohm/m. Antenna-load coupling tests are scheduled in 2009.
- Metamaterials: among a large number of potential solutions, we have chosen, in a first step, to evaluate an array of wires loaded with capacities able to reach $\varepsilon_r \approx 1\ 000 + j\ 100$, leading to a calculated coupling resistance of 5.5 ohm/m. If the 2009 feasibility study validates this approach, antenna-load coupling tests are scheduled in 2010.

We also participated to the study, design and commissioning of the ITER Like Antenna for JET, with the secondment of an RF engineer.

5.1.2.2 Lower Hybrid Current Drive

The experimental campaign started in May 2008 with the same configuration as the one of 2007: no major change has been made on the generator. The conditioning of the antennae required 3 days for a power of 300 kW max per tube. A water leak on a TS sub-system stops the operation at 2 MW for C3 and 0.7 MW for C2. In July, the system was operational again and had a very good availability: 2.5 MW on C3 and 1 MW on C2. During the campaign, 3 klystrons became unavailable on C2 due to water leak problem or high voltage withstand. At the end of the campaign, in November, the maximum injected power for C3 was 3 MW for 2 minutes duration. The generator was then stopped for the installation of 8 new klystrons (CIMES project).

We have conducted a “pre-Phase 1” evaluation of the physics motivations, technical design, cost and schedule for installing a Lower Hybrid Current Drive (LHCD) system on Asdex-Upgrade (AUG). The Figure 5-9 shows an artist view of the Antenna. The LH waves are a unique actuator for current profile control, being able to drive current far off-axis with the highest current drive efficiency. The maximum of the LH driven current density profile is predicted in the range $\rho = 0.3 - 0.7$ for the AUG reference scenarios, with a robust trend towards absorption around mid-radius. Combined with the other Heating and Current Drive (CD) sources available at AUG (NBCD, ECCD), it considerably broadens the flexibility of operation. Off-axis current drive capability is particularly useful in view of the Advanced Tokamak program planned for AUG after 2010. This study concludes to the overall feasibility of an LH system for AUG, using existing technologies and with significant margins to provide an efficient broadening of the current profile in the reference scenarios at $\bar{n} \approx 4.0 \times 10^{19}$ m^{-3}. This system would provide AUG with an enhanced current profile control capability for its Advanced Tokamak program and, if a PAM launcher is chosen, a unique opportunity to test this ITER relevant technology in ELMy H mode in Europe.

Following the assessment made in 2007 for a “day 1” LHCD system on ITER, the preparation for the design phase (EFDA task to be initiated in 2009) has been made [72].

In the frame of the collaboration between IRFM and SWIP (China), a young researcher visited us for 6 months, and worked on the pre-design of a new LHCD launcher to be mounted on HL-2A.

5.1.2.3 Electron Cyclotron Resonance Heating

The gyrotron prototype has been connected on a matched water load and a thorough campaign of research of optimized working parameters made with the following results:

- The performances of the prototype are permanently degraded compared to the original ones.
- This degradation is the consequence of the plastic deformation of the launcher of the gyrotron, due to abnormal interception and RF losses on this component.
- It is necessary to find a compromise between high RF output power and high interception for the routine operation of this gyrotron.

A prototype of a remote control system for the polarisation of the injected wave has been developed; this system will permit the control of the polarization while the injection angle will be modified toroidally or poloidally during plasma experiment. The design of a new system for the modification of the mirror position of the antenna has been made, aiming to a better accuracy of the injection.

The ECRH system has been largely used for the experimental campaign, for different studies:

- Start up of discharge in the ITER relevant conditions
- ECCD sawteeth control [73]
- ECRH modulation for heat pinch study
- Electronic fishbones at 2 T
- Grad (T)/T effect on impurities transport

In the frame of the collaboration between IRFM and SWIP (China), a young researcher visited us for 6 months, and worked on the pre-design of a new ECRH launcher to be mounted on HL-2A.

5.1.2.4 Neutral Beam Injection

Activities on the Mantis test bed:

During 2008, a number of diagnostics and experiments were developed in order to gain a better insight into the fundamental physics in the generation and extraction of negative hydrogen ions. A Cavity ring down spectroscopy diagnostic was successfully designed and commissioned on the KAMABOKO III ion source. Negative ion densities were measured as a function of pressure, power and plasma parameters in the filter filed region of the ion source in volume conditions. In order to investigate beam halos and beam aberrations, an emittance meter, designed by IPHC (Institut Polytechnique Hubert Curien, Strasbourg), was installed in the drift duct region of the MANTIS beam line. This diagnostic was commissioned and investigations of the effect of ion temperature and space charge on the beam emittance of a single negative ion beamlet were observed. An RF ion source has been designed for the MANTIS test bed. This ion source will allow fundamental physics studies into H^- formation in a “clean” tungsten free environment, in collaboration with the Laboratories involved in the ITER-NIS project (cf. below).

The 1MV test bed:

A successful collaboration with the JAEA laboratory in Japan took place in 2008 to test the SINGAP accelerator concept with 15 beamlets on the JAEA Megavolt Test Facility, the goal being to compare the Singap and MAMuG accelerator concepts proposed for ITER in the same experimental conditions. While beam optics was good (as expected) on Singap, the voltage holding appeared to be 200 kV less than that of the MAMuG. Also the electron power produced by the Singap accelerator was found to be two times larger than on the MAMuG. Data from this experiment enabled ITER to decide which accelerator to adopt: MAMuG.

Measurements of the beam halo have been done and we have established that the power fraction in the halo is <4 % when no caesium is used in the ion source and it increases to around 10 % when caesium is used to enhance the negative ion production. By removing the material downstream of the meniscus on the plasma grid, the halo fraction, while using caesium, could be reduced to around 5 %.

A new heated cathode structure has been designed and was installed in the Cadarache 1MV test bed during 2007. The results from the experiments showed that with the new configuration, there is a substantial improvement in the high voltage stand-off: we can hold the same HV (800kV) as before but at a reduced pressure which is the same as foreseen in the ITER neutral beam accelerator (0.03 Pa). We have also come to the conclusion that the relatively high dark currents that are found on the Cadarache test bed could be due for one part to the use of epoxy in the bushing.

A new test set-up for studying the electrical discharge between the acceleration grids in the ITER SINGAP has been designed and built. This test set-up will demonstrate the voltage withstand before and after a breakdown. The experimental campaign will start in May 2009.

Modelling activities: ITER-NIS

A research project (called ITER-NIS) dedicated to the development of physical models for the simulation of the plasma in the large size ITER negative Ion source has been accepted by the Research National Agency in France. ITER-NIS will involve for three years seven laboratories in different Universities (including Mantis test bed at IRFM) with expertise in cold plasma modelling, atomic physics, and ion source physics.

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Figures

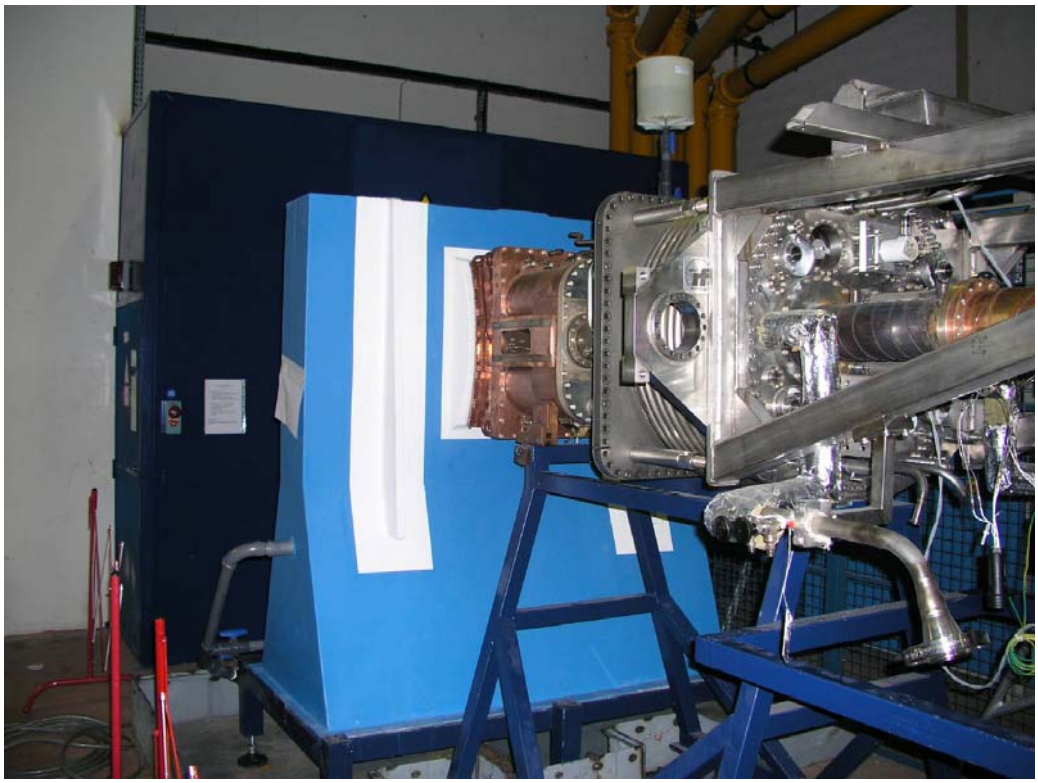

Figure 5-8: TS ITER like Antenna tested on baking soda water load

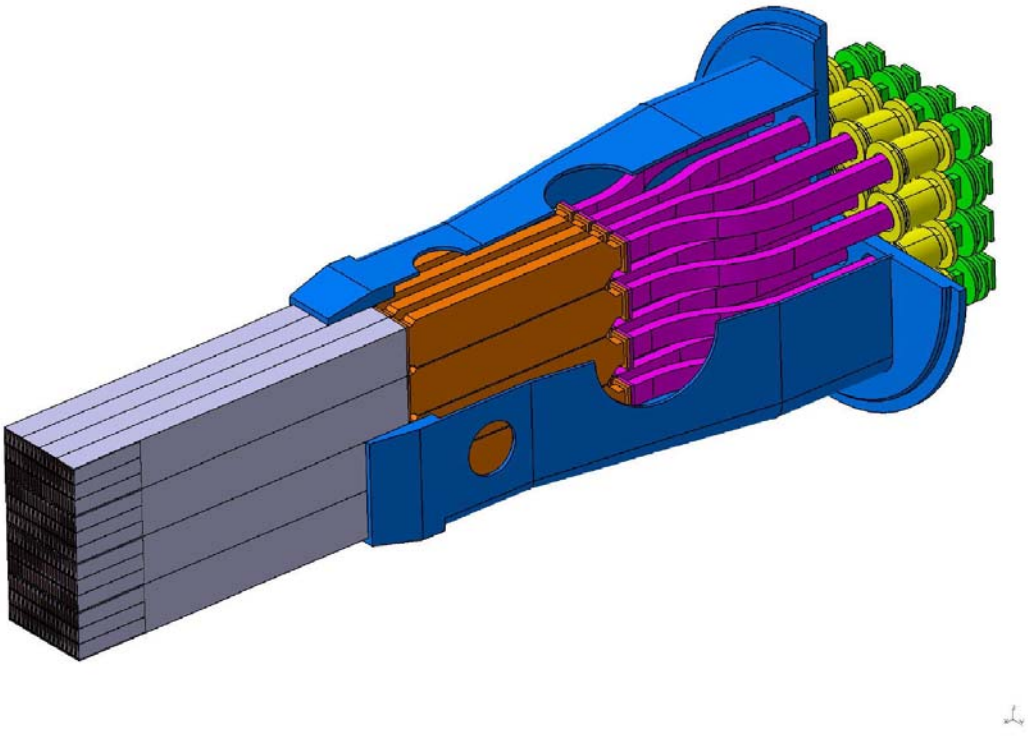

Figure 5-9: Artist view of the LHCD Antenna for AUG

5.1.3 Plasma facing components

5.1.3.1 Design of in-vessel components

Tile shaping of the beryllium tiles for the JET ITER-Like wall

An analysis was performed to predict the heat load on the JET ITER-Like wall beryllium (Be) tiles with a high level of details for several plasma configurations. The tiles are generally made as an assembly of Be blocks attached to an Inconel carrier. The Be plasma facing surfaces are properly shaped in order to optimise the power deposition (minimum exposure of edges). Castellation of the Be surface is made to reduce the thermal stresses in the material. When a complete shadowing is not performed, the lateral face of a block is wetted by the plasma. In that case, the penetration depth is calculated and the maximum power density at the face is determined. The different wetted areas are compared using a simple thermal computation, giving a temperature representative of the heat flux deposition on a single place [74]. The methodology is both accurate and efficient at computing power deposition and shadowing on crucial tiles (Figure 5-10). Synthesises results for all studied limiters showed that the most heated areas are front faces, and in case of penetration (tile to tile groove or castellation), the temperature increase is still dominated by the power deposition on the front (more than 80%). The supplementary temperature increase caused by the castellation is marginal. This means that operation will be primarily limited by front heat flux. Considering the heat exhaust capacity provided by the overall shaping, it confirms that the design is similar essence from the previous one with carbon tile. The tile surface heat flux will have to be controlled by passive and/or active monitoring to avoid overheating.

Design of Faraday shield for the ITER ICRF antenna

Based on the antenna with internal matching, a preliminary design of a faraday shield for ITER ICRH antenna was proposed and assessed. The conception of that design is modular in the sense that the antenna is an assembly of 6 individual modules, which are inserted in a port plug (2 horizontal x 3 vertical). The design of the 6 Faraday shield modules is based on a steel casing (Figure 5-11). A faraday shield module needs 18 plasma-facing components (bars: typical dimension are 340 mm long, 30 mm wide and 22 mm thick). The bar design (Figure 5-12) is based on the technology of the ITER first wall panels (sandwich of stainless steel, copper chromium zirconium and beryllium with a stainless steel water liner). The bar is assembled by hot isostatic pressing, and the water tightness is guarantied by TIG welds. The cooling channels are made from stainless steel tubes, which can be TIG/MIG, welded to the structure.

A first assessment of the peak beryllium temperature under heat loading gives 425°C (Figure 5-13), providing a good margin for resisting the ELMs. A specific feature of the bar is that it is clamped on both sides, to avoid the need for a sliding bond or an electrical connection close to the plasma. This feature is unique to this plasma facing components, as wall or divertor components allow for thermal expansion through sliding bonds or pinning. Thermo-mechanical analysis gives reasonable elongation amplitude during the heat loading (0.1 to 0.25%), which means that the fatigue would be acceptable. These favourable results indicate that the tile-structure bond is unlikely to be the limiting factor of the design.

The longitudinal stresses brought by the clamping of the extremities remain also reasonable with Von-Mises Stress across the connection assessed between 150 and 350 MPa (Figure 5-14). Optimization will allow increasing the margin with respect to 3 SM values (440 MPa @150°C). These promising results should however been further analysed and checked on actual mock-ups. A R&D program was proposed in that sense.

5.1.3.2 Qualification of high heat flux components for the ITER divertor

Definition of acceptance criteria for the ITER Divertor

A study was performed to assess the detection and the evolution during operation of calibrated defects artificially implemented on PFC samples, as an experimental basis for the definition of acceptance criteria for the bond armour/heat sink in the frame of industrial manufacturing conditions. This study consisted in particular in the manufacturing, SATIR inspection (non destructive infrared thermography test bed located at Cadarache – France) and fatigue tests at FE200 (high heat flux facility located at Le Creusot – France) of two sets of 54 samples (26 CFC monoblocks, 14 W monoblocks and 14 W flat tiles) with artificial defects, axially crossing over the monoblock geometry or transversally crossing from the edge of tile for flat tile geometry. For the 3 design options (Figure 5-15), an interlayer of pure Copper (Cu) was used as compliant layer between the armour material and the CuCrZr cooling tube (12/15 mm inner/outer diameter). In the past, the experience of the Fusion community showed that the most critical part of a PFC was the armour to heat sink bonding, therefore, maximum dimensions of an acceptable defect at the interfaces of bonding appeared to be relevant criteria with regards to thermo-mechanical fatigue. Consequently, crossing strip shape imperfections from 20° to 60° extension ($\Delta\theta$) at various angular positions (θ) were introduced on monoblock design and from 2 mm to 6 mm on flat tile design.

As a first step, the samples were inspected by transient infrared thermography SATIR technique. Detection of the imperfections on CFC monoblock geometry was found reliable with a limited uncertainty. In particular it was found that defect with an extension above 25° could be detected at the interface CFC/copper, with a SATIR inspection on the 3 sides of each monoblock. It was established that a defect at the Copper/CuCrZr interface can be detected with an extension higher or equal to 30°, whatever the position. Secondly, the samples were high heat flux tested following a program relevant with the lifetime of the Divertor PFC into the ITER machine. For each artificial defect, propagation or stability during the fatigue testing was assessed. It was observed that current CFC monoblock design option was compatible with the heat loads (up to 20 MW/m²) specified at the lower part of the vertical target, including the presence of armour/heat sink defects (up to 50° extension for a location at 0° or 45°) detectable with SATIR NDE technique developed at IRFM. The current W monoblock design appeared suitable for the upper part of the vertical target with defect extension up to 50°, but is not adapted for heat flux of 20 MW/m². The studied W flat tile design proved to be compatible with fluxes of 5 MW/m² but unable to sustain cycling fluxes of 10 MW/m². These results give confidence in the European capability to manufacture and characterize the ITER Divertor PFC with the required level of performance

On the basis of experimental results detailed in [75] the following acceptance criteria was proposed by IO: defects extension smaller than 50° at CFC/Cu interfaces (40° for Cu/ CuCrZr interfaces) could sustain cycling at 20 MW/m².

SATIR tests of vertical target qualification prototypes

Within the framework of the prequalification phase needed for the critical procurement packages shared by multi-party, including the Divertor plasma-facing components, each ITER Party should first demonstrate its technical capability to carry out the procurement with the required quality, and in an efficient and timely manner [76]. This is achieved via the successful manufacturing and testing of medium-size "Qualification Prototypes" (QP) made of three high heat flux (HHF) units. HHF units have a Composite Fibre Carbon (CFC) monoblock part and a tungsten part with either monoblock or flat tile geometry.

Presently, the technical specifications, for the procurement of ITER Divertor components, state that all copper interlayer on CFC armour must be subjected to 100% infrared thermographic examination. The SATIR inspection is considered a key non-destructive testing technique to assess the final quality of the joints between the CFC monoblock and the cooling tube [77]. Three batches of HHF units manufactured by European industry (PLANSEE Company and ANSALDO Ricerche) were received at CEA Cadarache. The CFC monoblock part of HHF unit is straight and consists of eleven CFC tiles, possibly separated by a 0.5 mm gap. Each of them has an axial length of 19.5 mm (or 20 mm if no gap is foreseen) and a width of 28 mm (Figure 5-16). Thermal responses of the three sides of a HHF unit are measured (Left, Top, and Right) (Figure 5-17) and a DTref_max parameter is defined to assess the quality of tested components. The physical relation between this DTref_max and the size of defect was established both by means of finite element modelling, taking into account physical properties variation of material and the background noise of the test bed, and by means of tests on samples with calibrated defects defined in the framework of the acceptance criteria study.

The position (θ), the extension ($\Delta\theta$) and an estimation of length (L) of a defect are given by the reversed geometric projection of the defect at the CFC/Copper interface (taking into account of the orthotropy ratio of the CFC thermal properties) combined with 3D finite element calculations (Figure 5-18). The error margin on the estimation of the defect position is evaluated at ± 5° angle. All the Cu/CuCrZr joints of HHF units were tested beforehand by ultrasonic technique. No significant indications of defect presence at the Cu/CuCrZr joint were reported in the ultrasonic inspection reports from manufacturers. Consequently, 100% of SATIR indications most probably stem from defect located at the interface CFC/Copper (or from the CFC itself). In the case of one monoblock, an unusual delay cooling during the SATIR testing was identified probably due to a misorientation of carbon fibres during the machining of the CFC monoblock. 80% of HHF units are declared acceptable by CEA taking into account uncertainties of measurement. In addition 98% of tested CFC monoblocks meet the acceptance criteria. The results highlight the need to set up a repairing procedure in order to accept a maximum of number of HHF units during Divertor procurement. The next step of this qualification work will concern the high heat flux testing of units, which will allow the consolidating of the acceptance criteria defined in the frame of the EU procurement for the Inner Vertical target of the ITER Divertor.

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Figures

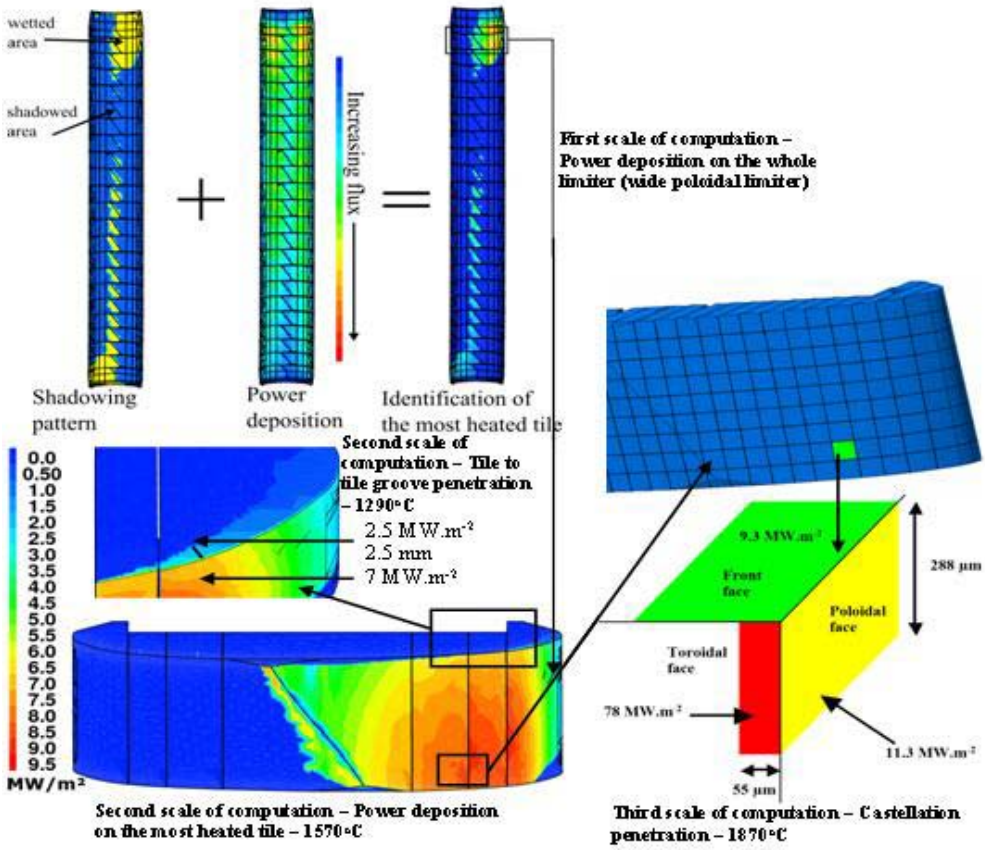

Figure 5-10: iterative process to define the most heated area – Wide Poloidal Limiter

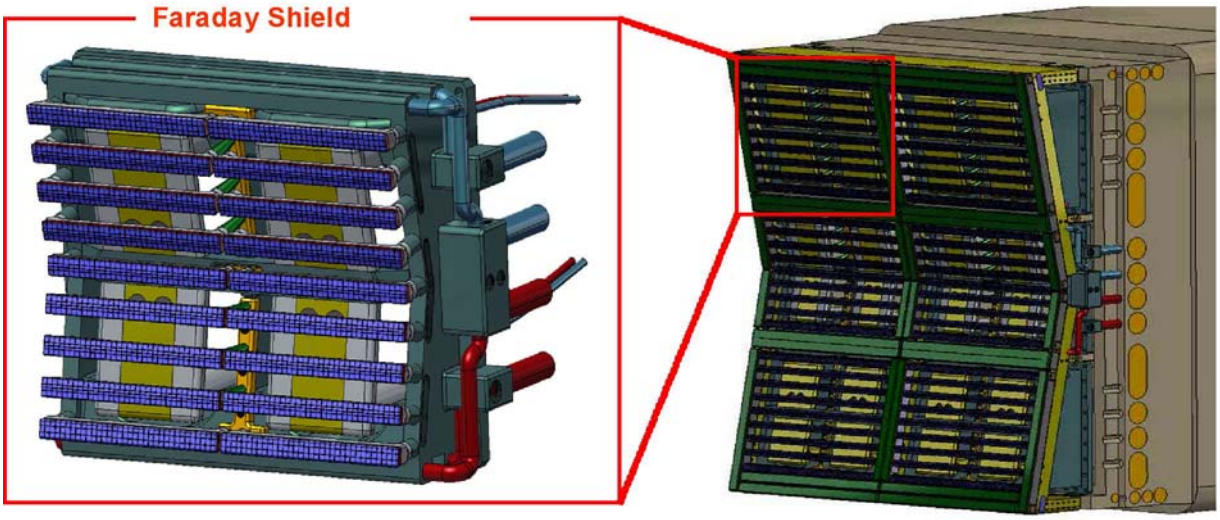

Figure 5-11: front view of the antenna with its faraday shield

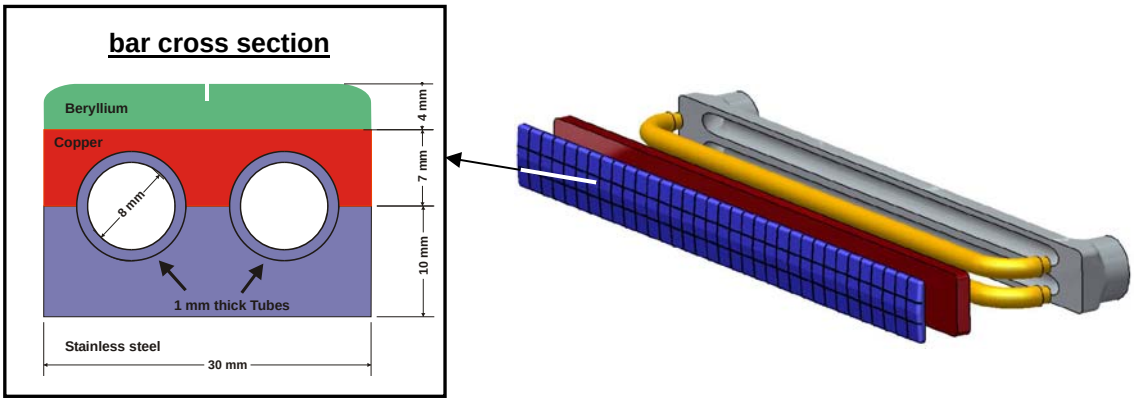

Figure 5-12: exploded view of a faraday shield bar

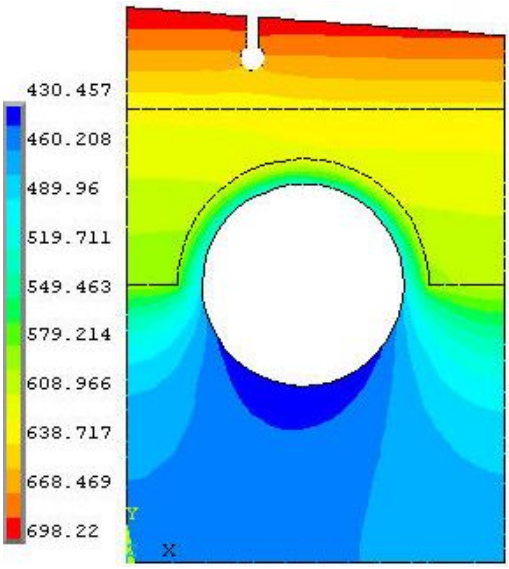

Figure 5-13: temperature distribution in the bar (Kelvin)

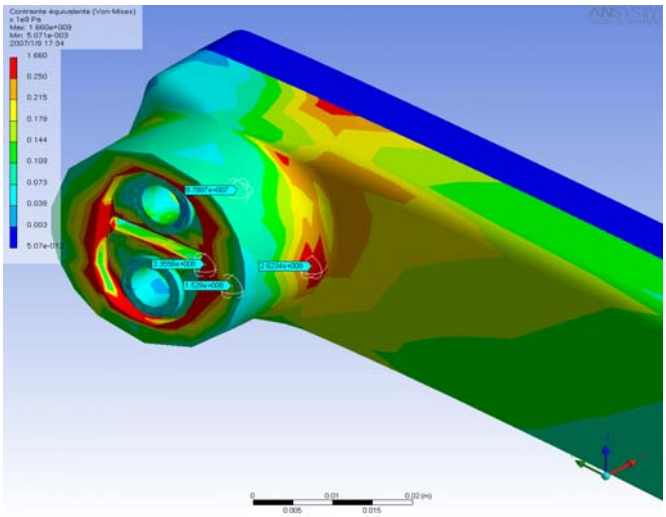

Figure 5-14: *Von Mises stress in the bar connection to the casing*

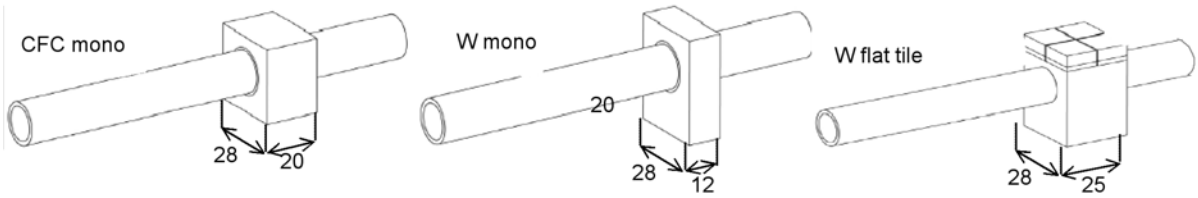

Figure 5-15: sketch of samples – Tube length 150 mm

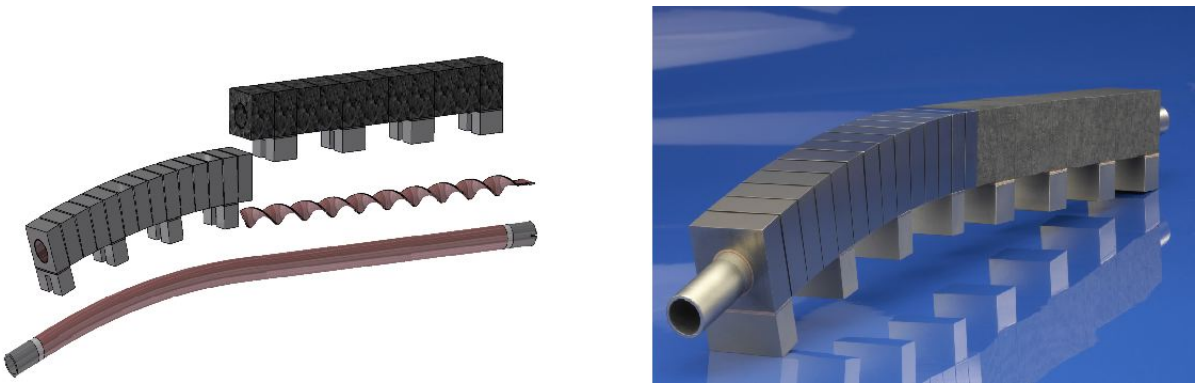

Figure 5-16: ITER prototype component manufactured by EU industry

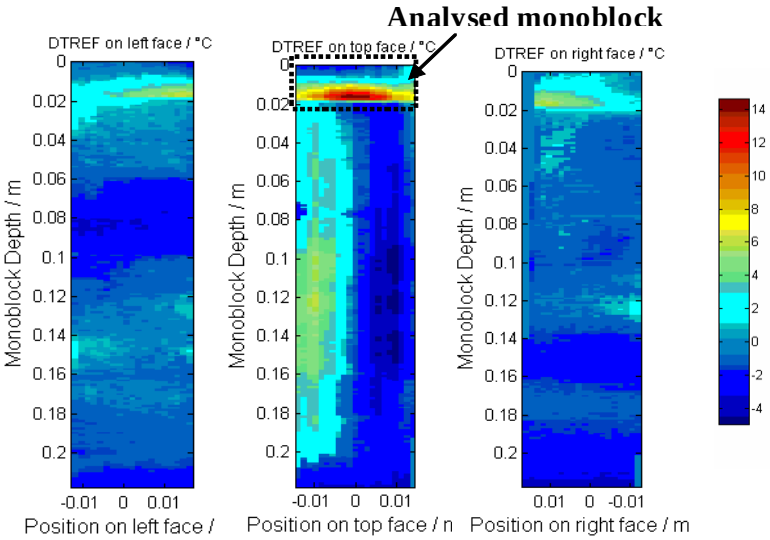

Figure 5-17: DTref Cartography on the three sides of one HHF unit (SATIR test)

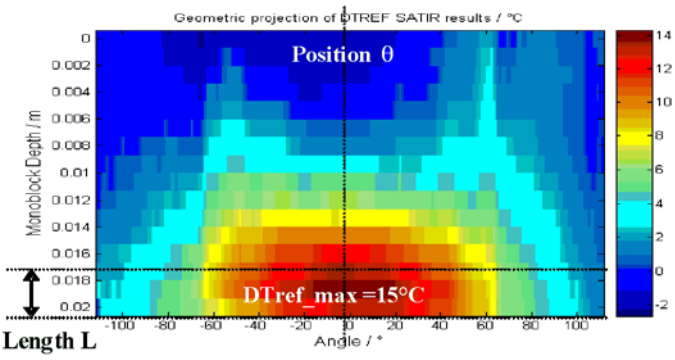

Figure 5-18: projection of the 3 DTref_max cartographies of the analysed monoblocks at the CFC/Cu interface to define the defect position θ and length L

5.1.4 Real time control

Real time control of the plasma discharges is a key issue due to its impact on both plasma performance and machine protection.

5.1.4.1 Plasma performance

Feedback control of sawteeth period, that was for the first time successfully achieved on Tore Supra in 2007 [78] [79], has been improved in 2008 using a new "search and maintain" control algorithm (see section 4.1.2).

The sawteeth period real-time computation algorithm was also used to start a new experimental program on the real time control of stationary states of the current (or safety factor) profile on Tore Supra. Open-loop experiments were first performed, using the Lower Hybrid Current Drive power as control variable, that showed that the safety profile can be varied at will from a sawthoothing monotonic one to a mildly reversed profile with $q_{min} \sim 3/2$ for a given plasma scenario (Ip=0.6 MA; nl=4 10^{19} m^{-2}; P_{IC}=3.5 MW). In this range of safety

factor profiles, five different states with different MHD activity have been defined. These states have been proved to be detectable in real time using the sawteeth period value and the knowledge of the state history from the beginning of the plasma discharge. Based on these results, and using the unique Tore Supra capability to sustain plasma discharges over several resistive diffusion times, it has been possible to succeed in controlling at will stationary states of the current profile (see section 4.1.1).

The work on the real time reconstruction of the plasma current profile has been continued in parallel. It is a very difficult subject both in terms of the numerical methods that have to be optimised to speed-up computation but also in terms of the physical reliability of the results which depend on the quality of the input data from diagnostics, that are often available in the form of integrals along chords. The main effort is put on the real time equilibrium code EQUINOX, developed in collaboration with the University of Nice [80] [81]. Several numerical developments were made in 2008: addition of new classes of parameterization of the two functions p' and ff' appearing in the Grad-Shafranov equation to be solved, improved choice of the regularisation parameter by the L-curve method, etc. A verification and validation process is under way on both Tore Supra and JET pulses (Cf. Figure 5-19) and the EQUINOX code will also soon be integrated under the Integrated Tokamak Modelling European Task Force framework.

Plasma current profile control design is a very difficult but very attractive issue where lots of work remains to be done to jump from semi-empirical PID type controller to model based control design. A new original approach using distributed non linear control oriented model development (PDE transport equations) was initiated in 2007 [82] [83], and some first closed loop simulation results were obtained in 2008 using a predictive control approach. A slightly different approach, based on the derivation of a linear state space model from a simplified set of transport equations with a two-time scale approximation has also been used to design closed control loops on JET [84] and JT60-U.

Experimental studies have also been performed on EC assisted plasma start-up, which is very important issue for ITER where the electric field for plasma breakdown will be relatively low in comparison to what is available on existing tokamaks. These studies have benefited to the development of a new control algorithm of the plasma current centre that has allowed a better early control of the plasma equilibrium [85] [86] [87].

5.1.4.2 Machine protection

Infrared imaging diagnostic is one of the key tools that may be used to protect plasma facing components from unforeseen overheating. Machine protection schemes based on real-time computation of specific areas maximum temperature from IR data were successfully tested on Tore Supra some years ago, but more advanced data processing algorithm are to be developed to improve the reliability in detecting specific events such as arcs on RF launchers, or sudden tiles overheating while rejecting perturbations due to light reflections and or presence of hardly attached carbon deposits [88]. Such developments are under way in the framework of a collaboration with the INRIA French Institute (Cf. Figure 5-20). A new real-time IR image acquisition and processing board has also been developed that will allow to embed advanced real-time algorithm, including to begin with the 2D calibration factor due to the optics [89].

Disruption mitigation experiments have been performed with the Tore Supra massive gas injection system, while varying both the type of gas (Helium, Argon and Neon), the quantity of particles injected and the plasma target [90]. First experiments aiming to a control of runaway electrons by the PF coils system have also been performed [91].

5.1.4.3 Development of a real-time control design and qualification platform

Real time control issues are now recognised as key issues for the safe operation of tokamak facilities and the optimisation of the plasma discharge performance. This will be of particular importance for ITER due to small design margins in many plasma and engineering parameters that may lead to a narrower operational space compared with present tokamaks [92]. These factors make it essential to develop real-time control design and qualification platforms as part of a future pulse validation methodology. The development of such a platform has begun within IRFM in 2008 in the framework of a new real-time control project. The idea is to develop a kind of "flight simulator" tool allowing not only to design control algorithms, but also to test them on the existing Tore Supra data acquisition and control platform prior to full experimental qualification. In order to favour a synergy with the European effort in the area of Integrated Tokamak Modelling, the ITM Task Force Kepler platform will be used as plasma simulation platform. The overall platform requires the use of many interfaces (Cf. Figure 5-21) that have been defined and will be developed in the following years.

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Figures

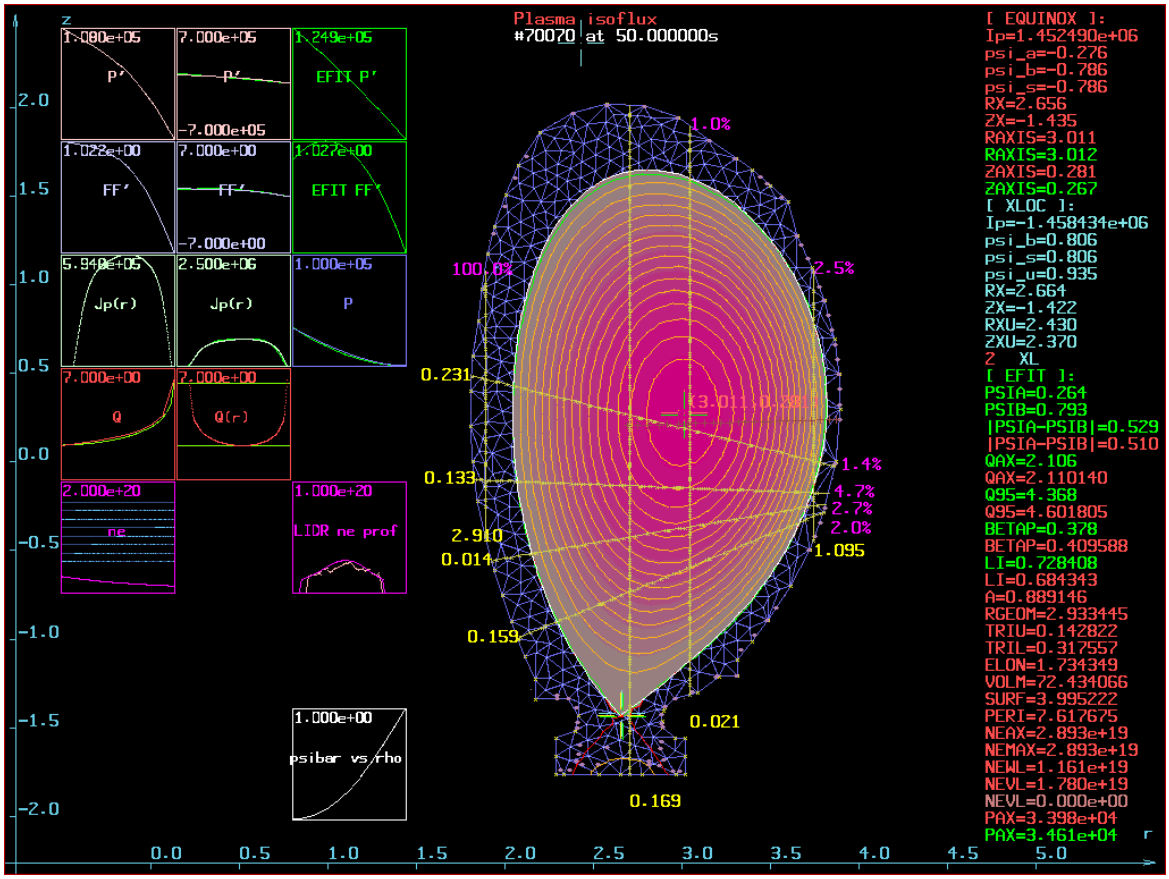

Figure 5-19: example of graphical output of the EQUINOX code

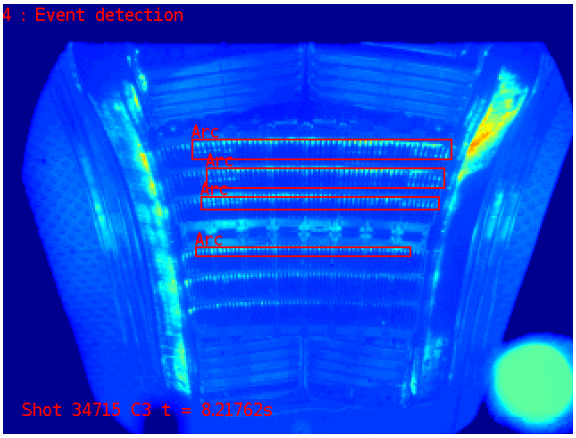

Figure 5-20: IR view of a LHCD Launcher with arc detected (red boxes) using a vision-based algorithm

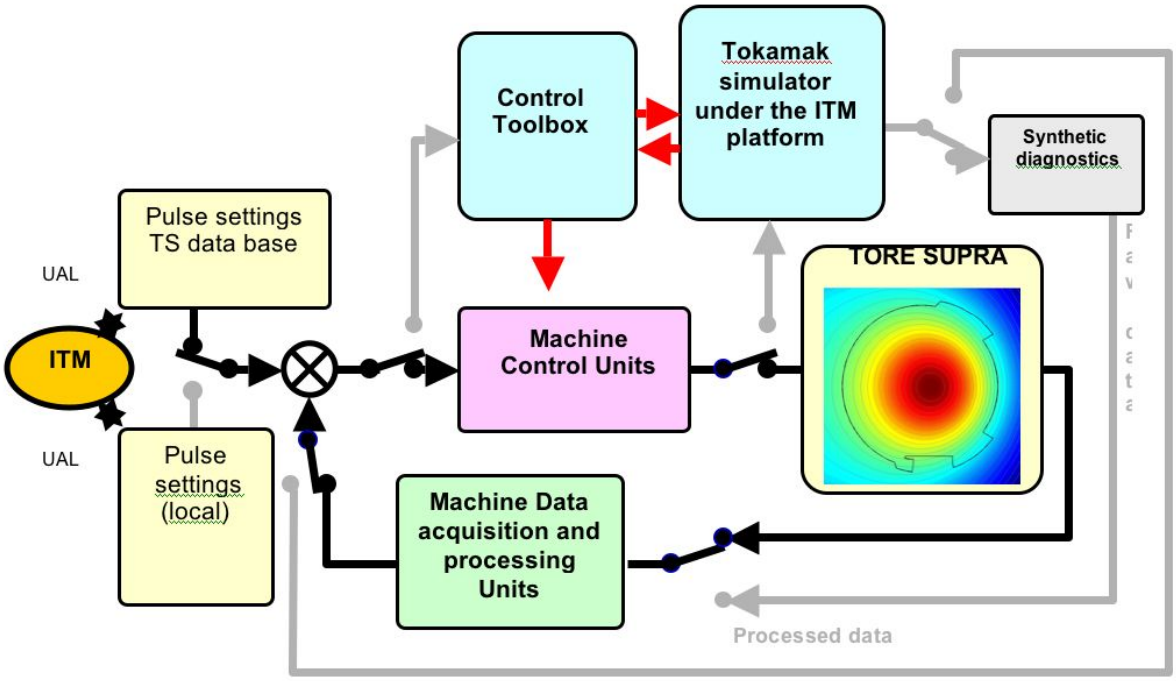

Figure 5-21: architecture of the foreseen real-time control design and qualification Tore Supra platform

5.2 Cryomagnetism

This section deals with development in cryomagnetism conducted in the frame of Euratom-CEA association. The following topics are reported: Magnets studies for ITER, Cryogenics studies for ITER, advances in test facilities.

5.2.1 Magnets studies for ITER

As part of EFDA contract 07-1700/1606, CEA provided its expertise to help to define the instrumentation for the ITER conductor samples tested in SULTAN at CRPP, Villigen (Switzerland), and also helped to analyse and interpret the associated measurements. CEA has a representative expert in the ITER Sultan Working Group (SWG). The contract covered the tests of 4 samples performed in 2007-2008. The conductor DC performances were estimated by both electric and calorimetric methods, the work included the analysis of the results in terms of the Nb_3Sn strain with respect to the single strand properties. AC losses measurements were also analyzed showing results consistent with expectations from strand properties, cable void fraction and cabling twist pitch lengths. The final report was delivered in October 2008. This work has continued since through a direct ITER contract.

Within the framework of the same EFDA contract, CEA performed a thermal stability analysis for the ITER PF#2, 3 and 4 conductors using the Gandalf™ code. This work showed that the copper content of the superconducting NbTi strand could be decreased down to usual values without a significant loss of stability thus allowing to decrease the cost of these conductors by decreasing the amount of NbTi strands. This work was submitted for publication in Cryogenics [93] and gave the opportunity to reassess the way to estimate the stability of an NbTi cable-in-conduit conductor [94]. The final report was delivered in January 2008.

As a contribution to the ITER Crash Programme, CEA performed destructive tests on both the TF Model Coil (TFMC) conductor sample and the TFMC 3-turn coil in order to estimate the residual strain of the conductor jacket in two different configurations. These tests showed differences in strain, which could be correlated at least qualitatively to the differences in performance of the conductor in the sample and in the TFMC itself. Results were presented at the ITER SWG in June 2008.

The HEAT task was dedicated to the characterization of Nb_3Sn industrial advanced strands and the influence of heat treatment on the critical properties. The second and last batch of nine industrial (ITER TF) strands was tested with the heat treatment specified for the ITER TF strands. All strands (batch#2) showed consistent J_C values, in the expected order of magnitude. Only one strand was found unstable and could not reach the critical criterion. Hysteresis losses were found consistent with the filament diameter values observed by micrography. All but one RRR values were found within the expected range. The micrographic views gave satisfactory results showing clear unreacted zones in strands (batch #2) and enabling to determine the effective filament diameter. The final report was delivered in December 2008.

CEA participated in the first round of the ITER strand benchmark using an unreacted strand provided by ITER IO. Critical currents (at 4.2 K) and RRR values were found well in agreement with other labs whereas filaments twist pitch and chromium thickness were found outside the acceptable windows and led to improve (or correct) the methods developed at CEA. Particularly, the chromium thickness that was measured by a weighting technique required to be estimated before (and not after) the heat treatment.

The Poloidal Field Conductor Insert (PFCI), which was initially planned to be tested in 2002, experienced a 5 year delay for its industrial fabrication but could finally be tested in 2008 in the CSMC facility at JAEA, Naka (Japan). Within the framework of the PFCITE task,

CEA participated in the preliminary discussions and modelling for the test program, in the test at JAEA, as well as in the analysis of the test results [95]. Regarding the NbTi conductor, CEA concentrated its work on the analysis of the DC performance with respect to the strand properties, the measurements of the AC losses by calorimetry, the analysis of the few runs related to the pulse mode operation, and the thermal stability. Regarding the intermediate joint, CEA worked on the DC performance (resistance and quench temperature), the AC losses measurement by calorimetry and the analysis of the pick-up coil magnetization signals. Whereas the results of the DC tests have been found very promising in terms of quench temperature (T_q) of the conductor, with a performance close to the one of the strand at maximum magnetic field in the cable, the few pulse current runs available have shown some significant ramp rate limitation of T_q at 5 kA/s, compared to the DC performance. The conductor showed AC losses at the expected level, increasing with cycling but reinitiated after each quench test. The joint DC resistance was found much lower than expected (2.2 instead of 5 nΩ) but the quench temperature was found about 1 K below expectation. The joint exhibited rather (unexpectedly) high AC losses under pulsed magnetic field, particularly for field variations along the coil axial direction. Clear "flux jumps" (FJ), likely due to saturation of strands in current leading to local quenches, could also be observed along both (radial and axial) field directions thanks to the pick-up coils located in the winding (see Figure 5-22).

5.2.2 Cryogenics studies for ITER

During 2008, R&D activity in cryogenics for Fusion at CEA was focused on studies for ITER in the framework of EFDA Contract 06-1515. This task was carried out in close coordination between two CEA laboratories: INAC/SBT (Grenoble) and IRFM/STEP (Cadarache). Three topics related to ITER high priority issues were covered: the analysis of the cryoplant operation modes (see Figure 5-23), the design of a development plan for the ITER Cryogenic System, and the design of the Cold Valve Boxes (CVB) of the cryopump system.

The analysis of the cryoplant operation modes addressed abnormal modes in order to assess the present design (production and distribution) of the cryogenic system. Abnormal modes coming from the ITER cryogenic components and those generated by the cryoplant itself were considered. Based on this analysis, the Process Flow Diagrams (PFDs) of the cryoplant and the cryodistribution systems were updated as Piping & Instrumentation Diagram (P&IDs) for the main cryodistribution boxes (ACBs/CVB) as well as cryolines going from cryoplant to tokamak building. The final report was delivered in March 2008.

The development plan addressed: an actual status report of the ITER cryogenic system project, a project analysis based on technical and managerial issues, a risk analysis for the overall project from design to operation (including a risk management proposal), and a schedule and resources loaded plan to fix any remaining issues and complete the preliminary design. This work has been completed and the final report was delivered in October 2008.

The design of the CVB of the cryopump system detailed the updated Process Flow Diagram (PFD) of the Cryopump and Pellet Injection System (PIS) distribution, and the Process and Instrumentation Diagram (P&ID) of one Cold Valve Box. The analysis of the operational modes of the cryopump system was also developed. This study also included a first version of the general design of the CVB and the interfaces with the components installed in the port cell. This task should be achieved beginning of 2009 with the diffusion of the related report.

5.2.3 Studies for DEMO

A paper has been jointly presented on DEMO at the 25th SOFT Conference by researchers issued from different CEA sectors (DEN, LIST, IRFM) [96]. Following the work that has been performed in the framework of DEMO studies, the paper insists on the consistent approach that is necessary to review the existing preliminary design of DEMO. The main results of the activities are presented including both in vessel and ex vessel components. The high interdependency between the several technological components is highlighted.

The main option for DEMO TF magnet system is the use of Nb_3Sn as superconducting material, however a superconducting material at high critical temperature has been studied (DSM/IRFU) that potentially could enable an operation of the TF system at 20 K and a reduction of the recycled power in the range of 10 MW. Two papers have been presented at the 2008 Applied Superconductivity Conference dealing with the critical properties of pre industrial round Bi2212 wires and their losses in pulsed field.

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Figures

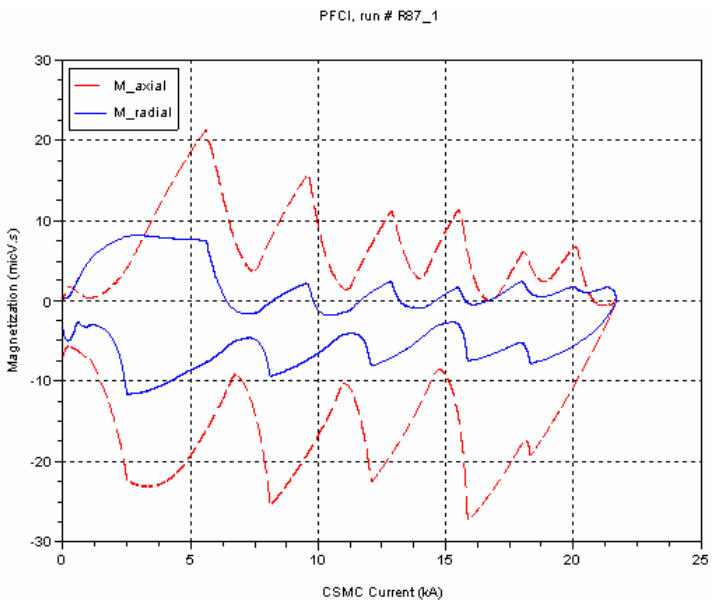

Figure 5-22: PFCI joint magnetization loops under a slow trapezoidal field pulse showing repetitive "flux jumps".

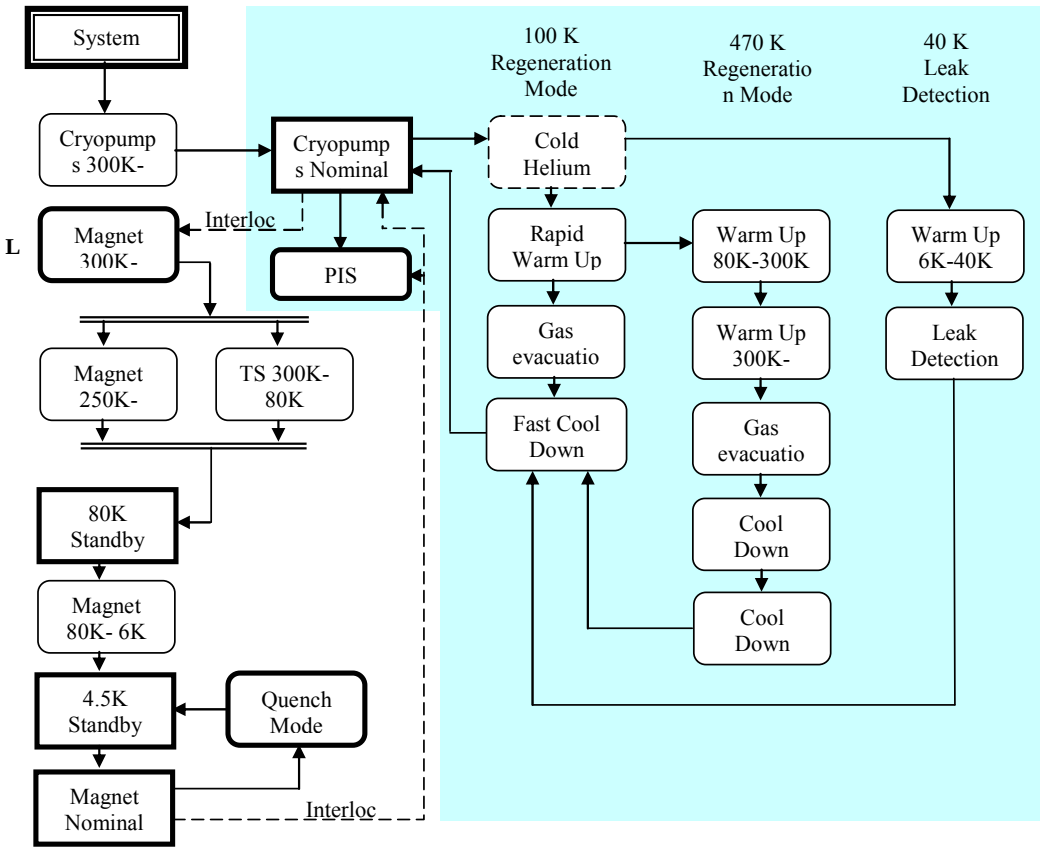

Figure 5-23: mode chart of the cool down for the ITER cryogenic system.

5.3 Electrical engineering

5.3.1 Evolution of the Tore Supra electrical network

A new electrical distribution of Tore Supra facilities was designed, using the new 15 kV electrical station built for the building site of ITER. This new distribution scheme allows having more flexibility, reliability and maintainability on the overall network and eases the exploitation. This new substation is supplied by 2×15 kV lines from the main electrical substation of the Cadarache Centre. During operational periods, the electricity consumption reaches 4 MW.

This year was also punctuated by several technical incidents on the general electrical network, which pointed out the lack of redundancy on our installation and the necessity to install junction box for the Electrical Motor Generator to ensure the safety of the magnet. Some electrical modifications have been carried out on the Uninterruptible Power Supply to maintain the capability for sustaining the desired consumption in case of electrical distribution loss.

5.3.2 Heating power supply

During the year 2008, the activity was mainly devoted to the replacement of the PCB transformers and capacitors, the development of the fast high voltage static switches for the CIMES project (Figure 5-24) and the exploitation of Tore Supra.

- Management of PCB transformers and capacitors evacuation:
    - upgrading and commissioning of old power supplies, to allow removal of PCB power supplies on the Mantis test bed,
    - installation of a new filter (capacitors) on the Singap test bed power supply,
    - evacuation of the PCB transformers and capacitors, procurement of new oil transformers for the Ion Cyclotron Resonance Heating driver power supplies.
- Work on the reliability of the fast high voltage static switches for the CIMES project and the improvement of the electronic circuit, including complete recoding of the software.
- Concerning the power supplies availability, we can note:
    - a few problems with the heating high voltage power supplies during the operation (2 days of unavailability),
    - ten days delay on the start-up of the diagnostic neutral beam, fault on circuit breaker,
    - three days delay on the test bed Singap, fault on a pneumatic contactor.

5.3.3 Toroidal and poloidal power supply

5.3.3.1 Toroidal power supply:

The electro-technical power safety circuit of the toroidal coils uses 6 Solenarc power breakers (Figure 5-25). These power breakers are designed to evacuate the toroidal current from the coils to power resistors, in case of quench or other default. After 20 years of successful operation, several mechanical parts and electrical contacts had to be replaced in order to keep the device operational. These equipments being obsolete and not any more maintained by the manufacturer, Schneider Electric, it was difficult to procure new parts for our equipments. The rejuvenation was carried out at the end of 2008 and the Solenarc power breakers are now back in operation.

5.3.3.2 Poloidal power supply:

For the operation of the poloidal system of Tore Supra, 2008 was a problematic year with the failure of the Spark Gap system (Figure 5-26). This is a key equipment of the fast mechanical switch used in the breakdown phase in order to initiate the plasma under good conditions. The procurement of new Spark-Gap from Efremov Institute was initiated, but failed due to the fact that these devices are considered as strategic by the Russian authorities, inducing major delays in their delivery. The decision was thus taken to develop new equipment, able to operate with the desired technical characteristics.

Finally, the 2008 experimental program was conducted with a plasma initiation phase adapted to the use of power supply converters only. This configuration is much more demanding for the electro-technical systems and not reliable enough, causing several defects on the poloidal power supplies during the year.

5.3.4 External Support

In parallel of our work, we were involved in:

- the pre-project studies of LHCD transmitter for ASDEX upgrade (power supply and fast protection by crowbar),

- the design review for the diagnostic neutral beam power supply of ITER,
- the studies of the 400 kV electrical network behaviour for ITER.

Figures

Figure 5-24: high voltage static switch for the CIMES project

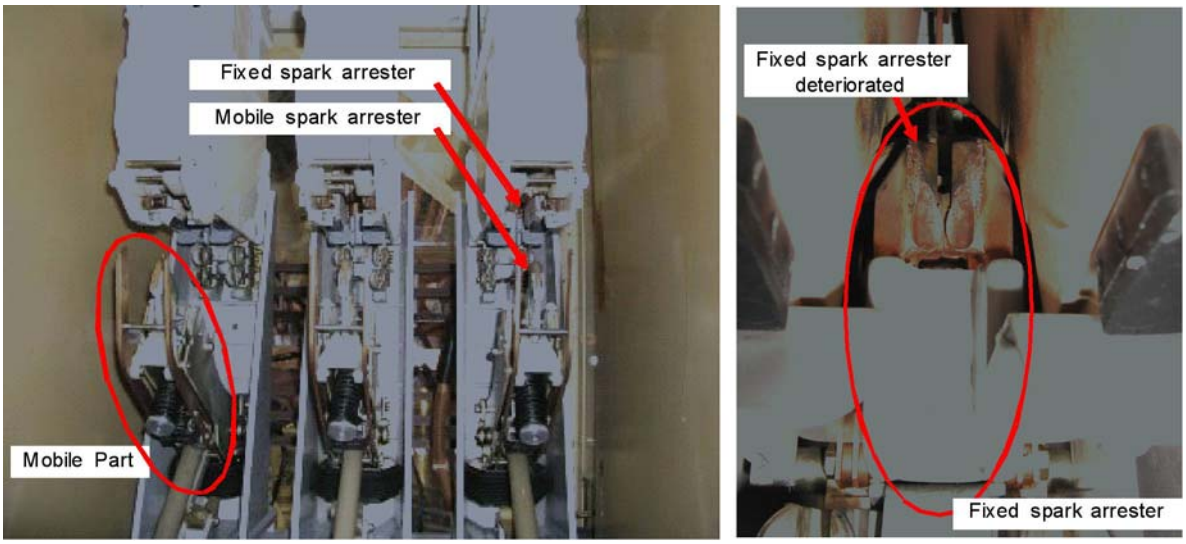

Figure 5-25: deteriorated Spark protection of the Solenarc

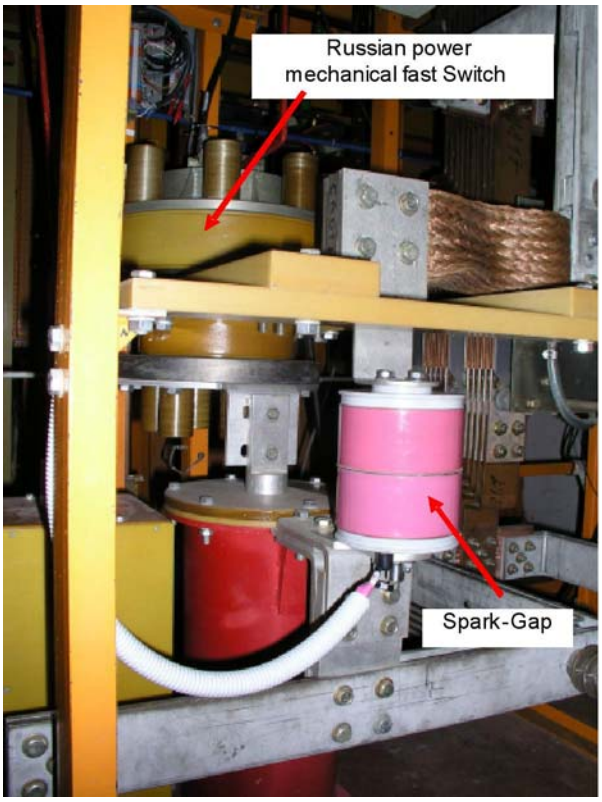

Figure 5-26: fast mechanical switch CR01

5.4 Remote handling

On a fusion experimental reactor, in-vessel interventions between plasma sessions are required for safety requirements, diagnostics calibrations or components checks. In every instance, these operations should have limited impact on machine availability and personnel radiation exposures. For that reason, it is mandatory that remote-handling systems operate in-situ without loss of the machine conditioning.

For that purpose, a multipurpose carrier prototype called Articulated Inspection Arm (AIA) was jointly developed by DRT laboratories (LIST/DTSI/SRI) and IRFM to demonstrate inspection abilities into Tore Supra vacuum vessel without breaking its conditioning. The AIA robot is an 8 meter long multi link cantilever arm composed with 5 modules of 160 mm diameter made in titanium. With a payload of 10 kg, its total weight is about 150 kg and it can be introduced through a small port of 250 mm diameter.

After a 1st promising complete deployment in Tore Supra under atmospheric conditions (25-04-2008), assembly and command control adjustment of the AIA carrier was completed in 2008. It included the qualification of the embedded vision diagnostic and the robot storage cask commissioning. The 3rd of September, deployment was conducted in the plasma vessel under Ultra High Vacuum (1.4 10^{-5} Pa) and at high temperature (120°C) conditions, a world first of its kind (see Figure 5-27). During the test, the complete deployment was carried out using automatic controls. Close up inspections of in-vessel components were performed under operators control with the camera positioned at 10 cm of the inner wall (Figure 5-28). Subsequently, no degradation of the vacuum vessel conditioning was detected and plasmas restarted without needing any additional cleaning of the wall. Prior

to this test the robot and the vision diagnostic was baked out up to 160°C for one week inside the storage cask.

In view of ITER applications, the design and configuration of the AIA prototype have been chosen to allow full surface inspection of one sixth of ITER plasma vessel. CEA is already engaged in development of interchangeable diagnostics or tools to be plugged on the front head of the arm. This concerns vision, water leak localization, Plasma Facing Components characterization and treatment by laser impact.

Because of the high potential use of the system, other applications are already identified today. For example, embedded tools could be used for:

- calibration of tokamak diagnostics like gamma-rays,
- particle detectors calibration using neutron sources,
- infrared endoscopes check using black body,
- alignment of diagnostics using laser beam or test pattern.

Experience gained with such a multipurpose robotic device will give very helpful information for future tokamak in-vessel remote handling integration.

References

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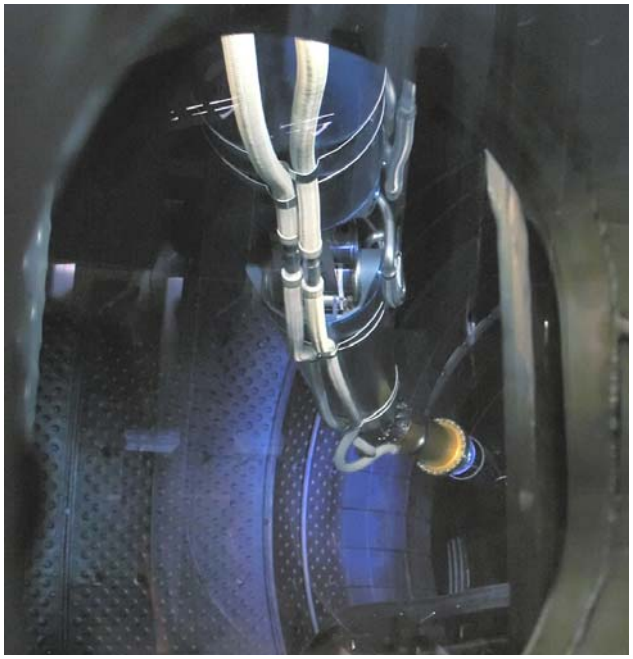

Figure 5-27: the robot and the vision diagnostic operation inside Tore Supra vessel at 120°C and 1.4 10-5 Pa

Figure 5-28: video captures of Plasma Facing Components during inspection

5.5 Control & data acquisition systems

Over the last year, CODAC activities have focused on projects linked to major improvements on Tore Supra, projects dedicated to the maintenance of the systems under our responsibility, and projects identified in our information technologies (IT) evolution road map. CODAC covers various activities, ranging from control-command to data acquisition and electronics, via the IT.

5.5.1 Control-Command

In addition to the support and expertise activities, three major control-command projects have been conducted:

- The deep modification of the control system for the LH project to allow the installation of the new Klystron generation.
- The AIA project in which we were involved for the realisation of the control of conditioning aspects (vacuum, baking mode, security)
- The external baking system project in which we have validated, for a complete operating campaign, all the regulation loops (27 in total) based on a new series of PLC linked by a fieldbus with a multichannel thyristor power unit.

5.5.2 Data acquisition

The migration of all the data acquisition units using a Linux operating system is in its second phase consisting in the development and the validation on plasma of a Linux VME based diagnostic. The migration of the data acquisition software has been started with the substitution of the message oriented middleware (MOM) RTworks with Notify, and the upgrade of the graphical interface for the supervision.

In 2008, two diagnostics using a new acquisition chain have been validated on plasma experiment:

- The new fast sweeping reflectometry (DREFRAP) with a 1GHz data sampling.
- The slow data acquisition of the fluctuation measurements (DTURB) diagnostic.

5.5.3 Electronics

The most significant projects in 2008 were:

- The power electronics DPSU (Drive Power Supply Unit), an important part of the poloidal magnets power supply command chain, has been refurbished.
- Development of a new device with low consumption capability allowing more than one year of battery lifetime for the Singap telemetry system that involves a temperature measurement system embedded on the 1 MV target
- The interferometer electronics that corrects most of the fringe jumps, built using the recent FPGA (Field Programmable Gate Array) technology that works routinely on Tore Supra and has been successfully tested on JET.

5.5.4 Information Technologies (IT)

The deployment of a local High Performance Clusters (HPC) infrastructure was the major project realized these last two years in the IT domain at the IRFM. HPC deployed in our institute, are essential tools to model and simulate complex physical processes, using very large scientific codes. Debug and test of these codes have to be done in a very interactive mode that is generally not allowed on supercomputers. So the IRFM has purchased and maintains a set of local clusters; the last one was installed in 2008, it is a 128 cores machine with InfiniBand communication and a 2To mass storage capacity.

5.5.5 Support to EFDA Remote Participation

The task (TW5-TPR-RPSUP contract n° 06-1376) has been running over 18 months since June 2006. A four months delay was granted in order to complete the final report. It constitutes a practical guide that may particularly help laboratories which would want to develop Remote Experiment Participation (REP) but also those which would like to use services offered to access remotely to an experiment. To conclude this work a poster was presented at the 25th Symposium on Fusion Technology (SOFT) held in Rostock from 15 to 19 September 2008.

5.5.6 ITM

In the framework of the ITM project the sub-project ISIP covers the hardware infrastructure (gateway), the framework and its associated tools (code platform), the data communication system (UAL or Universal Access Layer), the Web portal, several applications (ITM tools) and the data management (data structure and handling).

- The ITM applications and tools have been installed on the ITM Gateway in Portici.
- Kepler for ITM has been improved. Graphical tools have been added to facilitate the Fortran/C/Java code integration. A graphical interface has been developed to define and run the simulation. The 1st version was released on mid 2008.

- A trainee has worked 5 months on the access to the UAL from Scilab. The work is to be completed.
- EXP2ITM is a program developed to read specific tokamak data (Tore Supra, etc.) in the ITM structure (CPO).
- A Web portal has been designed in 2008 to provide a single sign-on point, to simplify the applications access and to use the GRID/HPC infrastructure. It has been released at the end of 2008.
- ISE is the graphical portal for the simulations. It has been subcontracted to the company STILOG.

5.5.7 EUFORIA

After the kick-off meeting in Goteborg the Euforia project officially started. The institute is in charge of the integration of the tools allowing the Kepler workflow to run on GRID and HPC infrastructures. Due to a lack of manpower we didn't succeed to reach all our goals but very relevant studies have been completed. The outcome was a significant improvement of the architecture, which simplifies the connection between the workflow engine based on KEPLER and the various GRID/HPC infrastructures.

6 Participation to W7X construction

The Wendelstein 7-X stellarator is presently under construction at the Max-Planck-Institute for Plasma Physics at Greifswald (Germany). IRFM contributes to the construction of this W7-X device with 6 persons working over 3 complementary fields:

6.1 Design and configuration control

Participation to the Configuration Control [99] of the machine for magnet system (coils, central ring, bus system, etc.) and thermal insulation (ports, outer vessel isolation, etc.).

Participation to the definition of space reservations (design space and maintenance space) for components in the torus hall and its adjacent buildings. Heating systems (NBI, ECRH), cryostat (various extensions), the coils power supplies and some diagnostics (diagnostic injector, Thomson scattering) were fully reviewed in 2008. Space reservation and layout of remaining diagnostics; routing of cooling and vacuum systems are planned in 2009.

Within the Design Office, contribution to the design of the thermal insulation of various components inside the cryostat. The close packing of components inside the cryostat and the complex shape of the ports require a careful design of the thermal insulation. The thermal shielding (see Figure 6-1) concerns all the 250 ports of the W7-X device and the 7 current leads domes. 7 domes and the 50 ports of the 1st module have been fully designed. In 2009, similar activities will take place for the 2 following modules.

6.2 Manufacturing and procurement

Procurement and manufacturing follow-up of a central ring structure [100] (see Figure 6-2), which is the main element of global coil, support structure (CSS). This task demands strong interactions between IPP internal departments and the subcontractors located Spain, Italy, Switzerland and Germany. Up to now, three modules were delivered in time and the last 2 ones are expected to be delivered in March and September 2009.

Procurement of the Support Structure of the Current Leads. This component, inside of the cryostat, must be able to accommodate the different thermal movements of the neighbouring pieces, some of them being cold (4 K) while others are at room temperature. There are 7 fixing boxes. The detailed design review was performed in 2008; manufacturing is ongoing for a delivery during 2009. Assembly testing will follow.

6.3 System and safety analysis

Within the Device Safety department, participation and coordination of safety studies for each main components and sub components. During 2008 the system analysis has been carried out and a few major hazard analyses have been performed. By next year the hazard analysis of the main systems (vacuum systems, the cryoplant, etc.) will be performed with the objectives to propose and implemented the associated protection systems. The full safety analysis of this complex machine is expected to be completed in 2011 in order to produce by the end of 2012 the W7-X safety cases.

References

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Figures

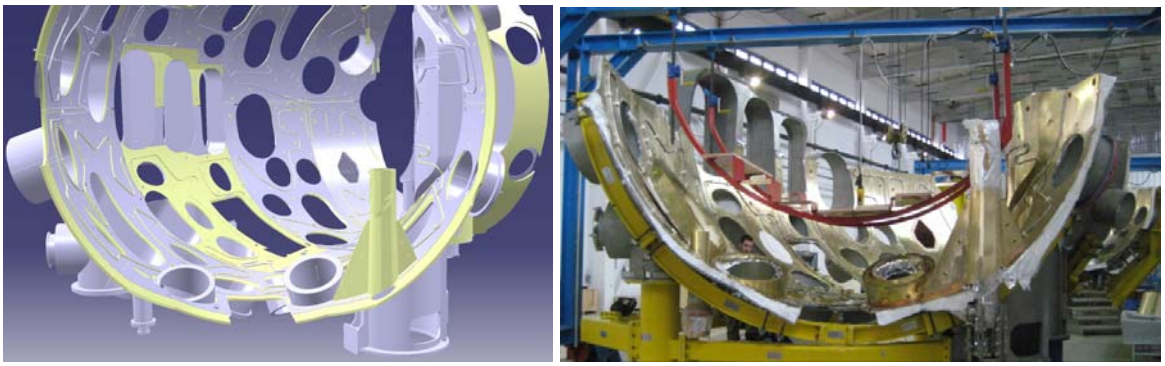

Figure 6-1: thermal insulation of the cryostat: design and mounting operations

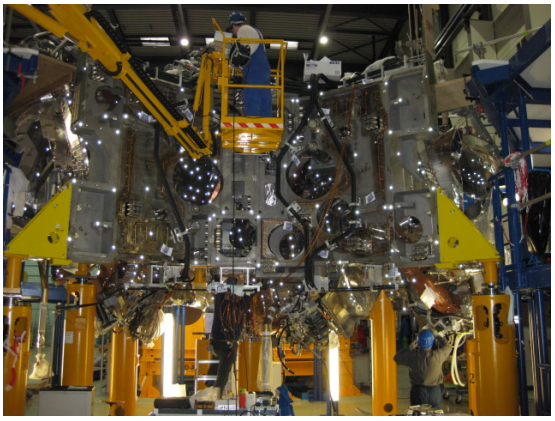

Figure 6-2: coils support structure on assembly stand

7 Broader approach activities

This chapter, being outside the scope of the Euratom-CEA Association, is for information.

7.1 JT-60SA

During the 2008 year, the main progress achieved on JT-60SA project was a strong consolidation of the overall tokamak design aiming at significant cost decrease. In this framework, the EU packages partially (9 out of 18 TF magnets, Power supplies) or fully (cryoplant) dedicated to CEA were considerably optimised. Besides, a new package was added in CEA responsibility perimeter, the TF cold tests facility, while the 140 GHz gyrotron procurement package was cancelled due to the project decision to postpone the ECRH installation.

The dedicated contribution of IRFM working groups in each of those packages is described in the following sections.

7.1.1 Toroidal field magnets

During this year, CEA actively participated to the consolidation of the machine design with increasing activity in manufacturing aspects, more elaborated simulation studies, and important experimental steps achieved.

Two powerful simulation codes TACOS [101] and TEXTO [102] were finalized and validated. Their role is globally for coil-oriented design, respectively devoted to comparative studies between options and absolute validation of particular configurations. The first TACOS applications allowed to validate two CEA design simplification proposals: removal of an internal super-insulation and consideration of an adapted casing cooling channels routing scheme. Mechanical studies were also conducted in 2D and 3D (see Figure 7-1), either showing delicate local zones in structures to be investigated (inter-coil attachment zone, insulation shear stress capacities) or demonstrating zones in which more design margins are allowed (winglet shear keys).

The first conductor pre-prototype TFCS1 (see Figure 7-2) was tested in SULTAN during two weeks (may 2008) and the data were analysed using appropriate CEA models. Globally, a satisfactory T_{CS} of 6.29K was found in operation-relevant conditions, and transient tests (AC losses, stability) turned out to be promising for investigating plasma disruption effects on the conductor. This sample is expected to significantly help for both validation of conductor design and qualification procedure.

An industrial cost study was prepared (specifications, call for tender) in order to investigate on a solid basis the manufacturing aspects of toroidal field magnets. Also, a documented proposal (from the manufacturability point of view) was provided to EU for a cooling channels configuration together with a proposal for conductor joints layout (CEA twin box concept).

7.1.2 Cryogenics studies

During 2008, preliminary study activities (R&D) in cryogenics for JT60-SA were carried out in close coordination between two CEA institutes: INAC/SBT (Grenoble) and IRFM/STEP (Cadarache). It includes many topics related to high priority issues:

- Cryogenic Reference documents: because of the updated design of the JT60-SA project, many modifications of the Project Integration Document were made in the field of cryogenics. A Conceptual Design Report draft, based on the Project Integration Document and miscellaneous others sources will gather all the specifications of the cryogenic components and the cryodistribution.
- Cryogenic industrial study: an industrial cost study contract was launched in support to technical choices, cost breakdown, size of components, manufacturing schedule, etc… A regular follow-up of this study is performed through meetings and periodic reports.
- Thermohydraulic calculations [103],[104]: they have been performed with the VINCENTA code in order to estimate the variable heat loads on the cryoplant during a reference scenario. Models of the supercritical helium circulating loops have been simulated for magnet system flows (EF, CS, TF and structures). Calculation will also be used to estimate buffer tanks size and to propose smoothing methods.
- Data acquisition, control system and electrical engineering: concerning data acquisition and instrumentation, the work has consisted in recommendations about choices of sensors and standards for the transmissions as the control and acquisition architecture of the cryogenic system must comply with signals and sensors that are part of the Japanese procurement. In the electrical engineering field, a study of various architectures of electrical supplies of the compression skids has been carried out, including cost estimations for investments and the operation phase.
- Experimental loop: activities have been conducted at CEA-Grenoble for the design and engineering of an experimental loop for pulsed heat load smoothing. This experimental loop will reproduce pulsed heat loads in the Central Solenoid circuits with a ratio of 20. Vessel of the cryostat (top flange and container) and the cryogenic lines to connect experiment to the 400W refrigerator were purchased. Call for tenders for heat exchanger and cold circulating pump have been written.
- Mode analysis and Process Flow diagrams [105]: this activity consists in the description of all the different tokamak states, the operation modes of the cryogenic system associated to these states and the determination of the heat loads occurring in each mode. In this frame, an optimization of the baking mode has been implemented. The description of the operation modes of the cryogenic system are illustrated by fluidic schemes (Process Flow Diagrams) performed using the Microsoft VISIO software (see Figure 7-3).

7.1.3 Power supplies

The major changes concerning the power supplies were due to the removal of the EF7 coil and the split of CS2 and CS3 coils: this results from the machine rebaselining which led to a new sharing between the European Partners. Major studies were dedicated to a review of design aiming at reducing the cost while maintaining the system performance. This year was also devoted to perform new simulations with the development of an electrical model for the whole electrical installation. With this new tool, technical studies and proposals have been made, and operating strategies have been validated.

Validations have also been conducted on the possibilities to re-use the existing transformers (Toroidal transformer of JT-60), the electrical diagram proposal of the Poloidal System and the scenario that is the essential point for the specification of our supplying.

An agreement was reached on code and standard, namely to use the IEC standard for JT-60SA EU procurement package.

7.1.4 The cryogenic acceptance tests of the JT-60SA toroidal field system and cold test facility

It has been decided by the end of 2008, that the cold test facility needed for the cryogenic acceptance tests of the JT-60SA toroidal field coils will be installed in France and that this test facility will be operated by CEA with participation of ENEA. The main motivation for the cryogenic acceptance tests of the JT-60SA TF system, before shipping it to Japan, is to demonstrate the capacity of the coils to operate at nominal conditions in the JT-60SA Tokamak in complement to the tests already performed during the fabrication phase. Especially, it has to be checked that the expected margins are present.

A first meeting took place on the 9 December 2008 at IRFM, to discuss about the tests contents and about the redaction of the contractual documents, consisting in the specifications for the tests and the specifications for the test facility. CEA and ENEA are presently participating in the documents redaction under the coordination of F4E. SCK-CEN and CRPP will contribute to the test facility by in kind procurements (respectively the cryostat and the cryoplant).

It has been decided that the coils will be tested one by one in a single cryostat. The testing phase will take approximately 2 years and the delivery of the first coil at the test facility is presently scheduled in April 2012. During the first phase, the preliminary specifications of the main test facility components will be written to prepare the call for tenders.

References

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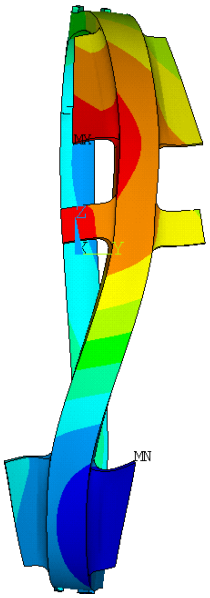

Figure 7-1: 3D mechanical deformation of a toroidal field coil.

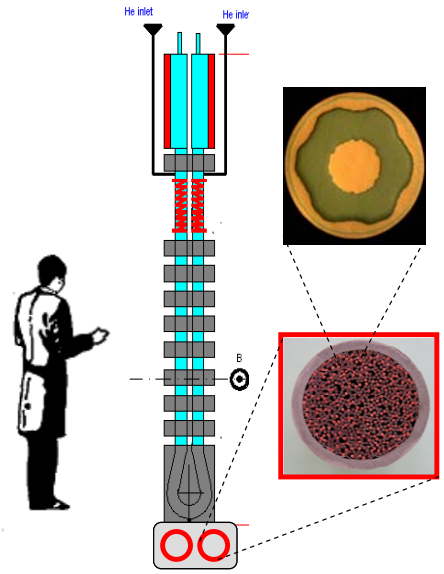

Figure 7-2: Schematic view of TFCS1 conductor pre-prototype.

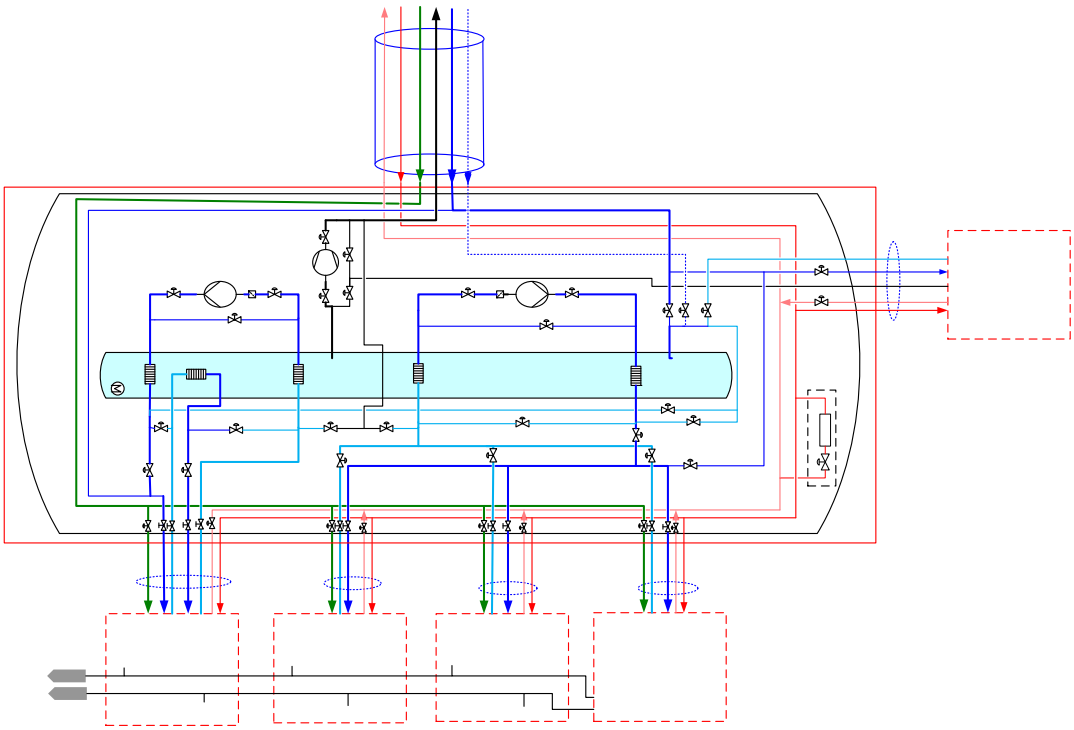

Figure 7-3: Process Flow diagram of magnet Auxiliary Cold Box.

7.2 IFMIF/EVEDA

ACB - Auxiliary Cold Box

7.2.1 Reminder: scope of the project

The Engineering Validation and Engineering Design Activities of the International Fusion Materials Irradiation Facility (IFMIF-EVEDA), one of the three projects of the Broader Approach (BA) agreement, cover mainly two aspects:

- the validation though the design, manufacturing, commissioning and test of the most challenging facilities of IFMIF:
  - The low energy part (up to about 9 MeV) of one of the two accelerators,
  - The lithium loop and target (at a scale about 1/3) with all its purification systems,
  - The elements of the High Flux Test Module at a scale 1:1, associated to an irradiation program of the capsules;
- the Engineering Design Activities of IFMIF, with the delivery of an Engineering Design Report, detailed cost evaluation and technical specifications for the urgent systems to build, enabling the Party(ies) to start the construction of IFMIF, if it(they) so wish, in a framework still to define.

The project is organized as follows:

- A Project Team, located in Rokkasho, Japan, coordinates the two Parties of the BA agreement; French members of the Project Team are attached to the Irfm;
- The Implementing Agencies, created in the frame of the ITER treaty (Fusion for Energy in Europe and JAEA in Japan) are responsible of the implementation of the project;
- Many Institutes contribute to the project; in France, CEA (Irfu, in Saclay) is involved in the accelerator by ensuring the European coordination and the delivery of major systems for the accelerator prototype, such as the injector, the drift tube linac, part of the RF system and the control command, and a set of diagnostics.

7.2.2 Main outcomes in 2008

In 2008, the main activities have been focused on the prototypes and the engineering design of the Test Cell. The rapid description below summarizes the main outcomes:

- **Accelerator Facility**: the main goal of the Accelerator Prototype is to deliver a D^+ beam of 125 mA in continuous wave (CW) at 9 MeV. The main systems of the accelerator are: an injector (140 mA, 100 keV), a Radiofrequency Quadrupole (RFQ) bunching and pre-accelerating the beam up to 5 MeV, a Matching Section, a Drift Tube Linac (from 5 to 9 MeV for the prototype and up to 40 MeV for IFMIF) and a High Energy Beam Transport (HEBT) line; diagnostics, the RF system, a Beam Dump and a Building complete the facility.
    - On a proposal of CEA and Ciemat, the initial conventional room temperature Alvarez drift tube linac (DTL) has been replaced by a more modern superconducting half wave resonator DTL, offering many advantages, such as reduced length and RF power (and thus operating cost), wider space between beam and drift tubes (reducing the beam interception, and thus activation), etc.
    - The preliminary design review of the challenging Radiofrequency Quadrupole (RFQ), task of INFN Legnaro, Italy, has been passed successfully. Brazing technology is one of the most difficult issues of this delicate system.
    - The Injector is an update of an existing source developed in CEA, called SILHI. Its Critical Design Review has also been passed successfully.
    - The contract for the construction of the building in Rokkasho has been awarded and the construction has started.
- **Lithium Target Facility**: the so-called EVEDA Lithium Test Loop (ELITE) has the following main characteristics: vertical lithium flow of 25±1 mm thick, 100 mm wide (instead of 250 mm for IFMIF) and at a nominal speed of 15 m/s (required for IFMIF to avoid lithium boiling under the intense flux of deuterons); the lithium must be particularly pure (a few wppm of O, N and corrosion products) to avoid the erosion of the nozzle, which propels the liquid lithium on the backplate.
    - An optimization of the shape of the backplate has been calculated, to decrease the risk of hydraulic instability (the backplate is curved to increase the pressure in the lithium flow, and thus its boiling temperature).
    - The bayonet concept (Italian proposal) has been chosen for the backplate of the EVEDA lithium test loop, to be built and tested in Oarai, Ibaraki, Japan.
    - Thermo-mechanical and neutronics calculations have shown that austenitic steel was inappropriate for IFMIF; the choice of reduced activation ferritic martensitic steel (F82H) for the prototype has thus been done, in spite of welding difficulties of this steel. Its better behaviour with respect to erosion and corrosion has also been demonstrated in a test loop in Brasimone, Italy.
    - Nitrogen and hydrogen/tritium trapping demonstrations at a laboratory scale have been done in Japanese Universities by respectively Fe/Ti (Tokyo) and Y (Kyushu) hot traps; experimental setups have been prepared to check under more realistic conditions (flowing, and not static lithium) their performance.
- **Test Facilities**: the Test Facilities are the set of systems and equipments directly connected with the nuclear part of IFMIF Plant: the Test Cell, where the modules containing the samples or devices to be tested are located, the access cell above it and the post irradiation examination (PIE) facilities where the samples will be characterized. Remote handling tools are another important part of the Test Facilities.
    - The Test Cell is in complete redesign. The initial design, proposed in the Comprehensive Design Report, 2004, was based on an arrangement where the

delicate and fragile modules were hanging below large blocks of concrete, ensuring the neutron and gamma shielding. In order to improve the flexibility of IFMIF, decouple the shielding with large blocks of concrete from the mechanical positioning of the modules, improve the reliability of the facility, a complete redesign has been launched with a more realistic layout.

- The High Flux Test Module (HFTM) has been also deeply revisited to improve its cooling and the temperature control of the samples under irradiation.
- First proposals of experiments on medium flux test modules have been made, in particular for breeding blankets.
- The neutronic code used for IFMIF design has been considerably improved; it now integrates the primary and secondary gammas and is compatible with MCNP v5 code. Its application to IFMIF modules led to a general increase of temperature by about 10 % due to gamma heating.
- Thanks to the arrival of Belgium as a European voluntary contributor, the irradiation programme of the HFTM is now more ambitious, by the use of the BR2 fission reactor in Mol; a first experimental setup has been proposed.

The Project Team in Rokkasho, made of 8 professionals at the end of 2008, is in charge of the overall management and coordination of the project. A Management and QA plan, describing the organization of the project and its main tools, has been written. The work breakdown structure is now stabilized and a preliminary programme for IFMIF's conventional facilities has been drafted. A first IFMIF-EVEDA workshop, gathering more than 70 people from Europe and Japan, was held in Karlsruhe in September; the main topic discussed was the interface between the D^+ beam, the lithium target and the high flux test module.

Figures

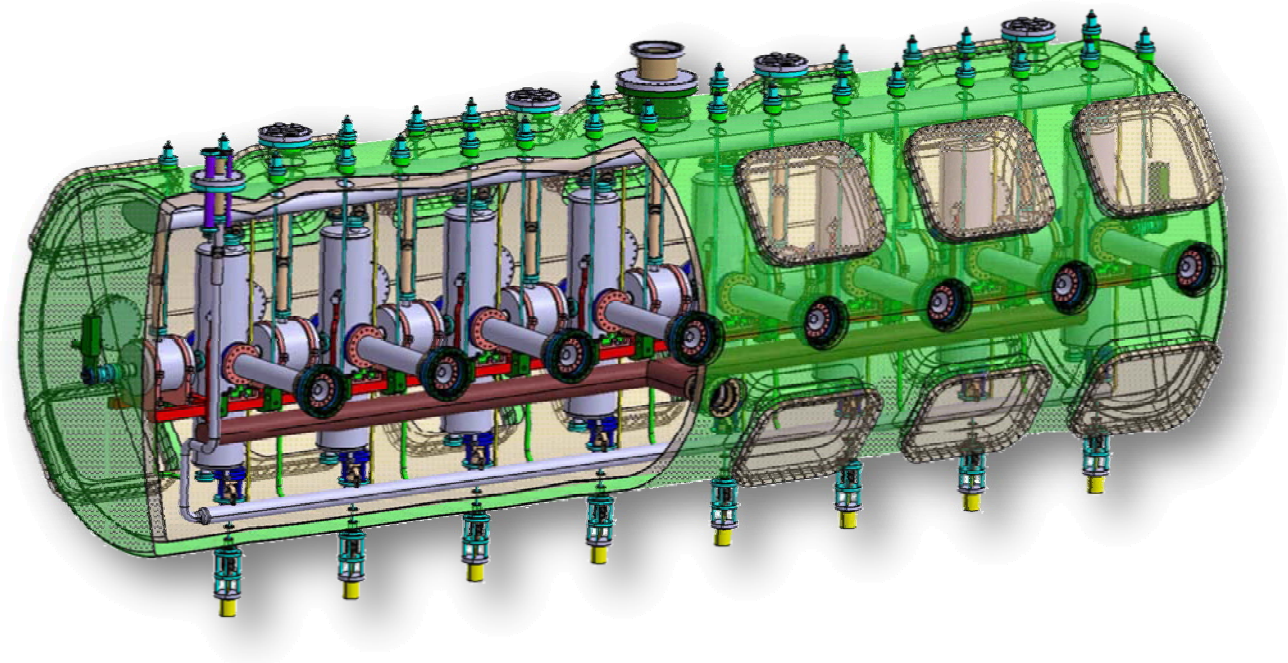

Figure 7-4: cryomodule of the DTL prototype. 8 half wave resonators are visible, each of them coupled to the RF system. Focusing is made by a set of superconducting solenoids.

7.3 IFERC

In the frame of the broader approach, it is planned to build in Rokkasho (Japan) a highly parallelised computer (HPC), called IFERC-CSC, whose computing power should be around a Teraflops. IRFM participate into this project via the IFERC Special Working Group number 1 (IFERC-SWG1), chaired by Pr K. Lackner and co-chaired by Pr T. Hirayama. One of its duties is to prepare the benchmarking process for the procurement phase. The IFERC-SWG1 met on July and October 2008 to select the high-level benchmark codes, which are to be sent to the bidders for answering the call for procurement.

The criteria applied to select the codes proposed by the both parties (Europe and Japan) were the following:

- scalability - codes should include both well tested, proven codes that already show performing features, and codes that are more difficult to parallelize,
- good coverage of numerical techniques for testing HPC architecture features,
- coverage of the various fundamental numerical schemes,
- coverage of a representative set of physical themes,
- codes should be able to address problem sizes to be run on the IFERC-CSC machine.

Based on these arguments, the high level benchmark suite was decided to consist of the following seven codes:

- GENE, GT5D and Gpic-MHD codes from gyro-kinetic codes,
- GEMR, MEGA and JOREK codes from fluid or hybrid fluid-kinetic codes, and
- MDCASK code from material science codes.

Three of these codes originate in Japan, and 3 in the EU, whereas one was originally developed in the US, but is now in use both in Japan and the EU. Low-level benchmark codes were also discussed during this meeting.

In 2009, the SWG1 will follow the project plan for implementing the CSC (market research, assessment of the benchmarking, user requirements, etc…). The computer should start operating in 2012.

8 Other activities

8.1 Public information

The Association Euratom-CEA continued in 2008 an important level of activities towards public information, along four main axes: dedicated conferences on fusion, participation to exhibitions and national "Science en Fête" operation, actions for students and pupils, organisation of Tore Supra visits.

For the general public, 25 large public conferences have been made during different manifestations all over France, from Paris to Marseille and from Poitiers to Nancy (2500 listeners). One presentation has also been made during the climate change meeting in Bangkok. We made also 3 participations (Nice, Digne, Domont) in round tables and "café des sciences".

The actions for students and pupils:

- 33 presentations in secondary schools, high schools, universities, mainly in PACA region but also in Paris, Lyon …(3000 listeners)
- 10 short presentations with magnet property discoveries (3h) for primary schools, total of participants: 300 pupils.
- 5 special days with presentation and Tore Supra visit for a total of 80 students (TIPE, Travail personnel)
- 15 young student welcomed for one week (secondary school training course)
- in the frame of the "Journées CEA Jeunes" organised within the whole Cadarache Centre, 15 small experiments on different subjects have been proposed to students during their 2-day stay

Concerning exhibitions, fusion was included in the CEA stand at the low carbon manifestation in Paris, and IRFM contributed to the stand organised by ITER at "Foire de Marseille".

For the French "Science en fête" operation:

- Participation at the European science city in Paris
- A stand on fusion at Marseille, in collaboration with ITER
- Practical works on light and magnets at Vinon college
- Fusion presentation: a total of 12 classrooms have requested a such presentation mainly in PACA region but also in Brittany.

5189 people have visited Tore Supra in 2008. It represents 90% of the number of Cadarache visitors. One quarter of the visitors are students (from college to university), a third is “large public”; teachers represent 3% of the visitor number and VIP 10%. A group of 16 persons, called “chercheur communicant” is in charge of this task. 20 journalists have also been received mainly for the press-review associated to 20 years of Tore Supra.

8.2 Teaching activities

Many agents of the Institute are involved in teaching activities, essentially at the level of the second year of the Master degree. Our activity is mainly concentrated on the recently created "sciences de la fusion" master degree. This master proposes 3 different formations on fusion science, either devoted to the physics of inertial or magnetic fusion, or to the associated technology. While the first two formations usually lead to PhD thesis, the latter one is more engineering oriented. This master is proposed in 4 sites in France (Aix-Marseille, Bordeaux, Nancy and Paris), and is habilitated by 10 high level educational establishments (including 7 Universities) clustered within a so-called Federation. We give lectures in all the sites but Bordeaux. In addition, we have also been present for a long time in 3 other masters, namely the "onde, matière, plasma" master from Paris-XI university, the "modélisations et simulations numériques" master from INSTN (Institut National des Sciences et Techniques Nucléaires) and the "physique théorique et mathématique" master of Aix-Marseille II university.

In 2008, our teaching activity corresponds to 200 hours of lectures (165h for the sole SF master) plus 60 hours of practical works. 19 people from IRFM get involved. The number of hours is significantly less than for the previous year (310h in 2007) since the formation devoted to the physics of magnetic fusion did not open in Aix-Marseille this year due to the lack of students.

In addition to the regular lectures, our Institute and INSTN-Cadarache also host all the students of the "sciences de la fusion" master a couple of days in September and a whole month in February, during which visits, practical works, seminars, lectures and examinations are organised. Finally, the lectures gave birth to a large number of original courses, written in English, which the students can freely access via the master web-site (www.sciences-fusion.fr).

8.3 Training programmes EFTS & GOTP

The Association Euratom-CEA is involved in the two training programmes EFTS and GOTP. EFTS activities started in 2007, with 5 trainees hired by IRFM for 4 training programmes:

- MATEFU (MAgnet TEchnology for Fusion), leaded by CEA, involves 8 trainees, and a progress meeting was organized in June 2008 at Cadarache. The activity of the CEA trainee in 2008 was related to the mechanical analysis of the different solutions envisaged for the JT-60SA structures using Ansys. He was also involved in the fabrication of the JT-60SA pre-prototype conductor, which was successfully tested at CRPP. He participated in the test of the NbTi strand, which is used in this conductor. He attended the 25th SOFT Conference, where he presented a paper on the JT-60SA superconducting strand, and took directly part to a European JT-60SA meeting at Garching in November.

- ETN_PFM (European Training Network for Plasma Facing Materials), leaded by CEA: the CEA trainee divided his 2008 activities between attending pedagogic courses and taking part into on-going projects in the PFC team of the CEA/Cadarache. For instance, he developed two innovative image processing techniques applied on IR thermography data from SATIR thermo-signal, allowing the identification of interface defects within the PFCs without the use of Reference element. These two techniques, based on data processing techniques and resolution of inverse problems in heat conduction, were presented during the conference SOFT2008 in Rostock (Germany) and published in Fusion Engineering and Design. During a short secondment in IPP/Garching, the trainee also set-up and took part into a High Heat Flux experimental campaign in the GLADIS ion beam test bed.

- EODI (Engineering of Optical Diagnostics): the main subject of work of the trainee in 2008 has been the advancement of the optical design of the Visible/IR diagnostic for the equatorial ports of ITER. The work was conducted in cooperation with CIEMAT with several mutual exchange visits with the CIEMAT trainee and they presented together the results of their optical study at the closing meeting of the TW6-TPDS-DIADES task to F4E. Tore Supra relevant optical studies were also conducted on the subjects of: calibration of IR endoscopes, design of the visible spectroscopy imaging diagnostic, design of the thermal He beam diagnostic, and preliminary optical studies for the Laser Ablation System Kit. For JET an optical design was proposed for the application of the photothermal method. The trainee participated in several training courses (advanced ZEMAX, basis of CATIA V5, radioprotection) and conferences (OCS'08 and HTPD'08) and is first author of a scientific paper (Rev. Sci. Instr.) and co-author of several other articles.

- ENTICE (European Network for Training Ion Cyclotron Engineers): the CEA trainee started to work in Cadarache on the development of an innovative concept of low power RF loads for IC antenna. He then went to Italy (Politecnico, Turino University) and Germany (IPP Garching, Spinner Gmb) where he developed arc detection system. In the meantime, IRFM hosted 2 others trainees (I. Zammuto and C. Hamlyn-Harris) who were involved in RF mechanical engineering studies for the ITER IC launcher and experimental activities on the Tore UPRA ICRF test stand.

The GOTP programme has been launched by EFDA in 2007 and CEA is involved in 5 of the selected programmes and leads two of them: ITER-PPE and LITE. The notifications have been received in July 2008, and autumn has been devoted to kick-off meetings, and selection then hiring of the candidates, which start their tasks in 2009.

9 Appendices

9.1 International collaborations

In addition to the strong involvement in ITER and JET, the Euratom-CEA Association maintains a high level of collaborative activities with other Euratom Associations as well as other international partners in many domains.

The following table presents only active collaboration for which a topic is precisely defined, with an identified responsible officer from Euratom-CEA as well as from the partner laboratory, presented by country, laboratory and scientific topic.

Among Euratom Associations, in the continuation of previous years, there is an important activity with the 3 German Associations (IPP, FZJ, FZK) totalizing 9 active topics, and with the Czech Association IPP, with scientific work on 4 topics, most of them concerning edge plasma studies. Collaboration with other associations is not less active, and 26 collaborations are reported from 11 different countries.

Outside the European Union, we have collaboration with almost all ITER partners (except Korea), on fusion related topics different from ITER. The strong increase of collaborative activities with China in 2006-2007 continues to be intensified and reaches now a level of 9 topics, with SWIP (Southwestern Institute of Physics, Chengdu) and ASIPP (Academia Sinica, Institute of Plasma Physics, Hefei). CEA scientists totalised 5 months in China in 8 missions for collaborative work, while Chinese scientist totalised almost 3 men-year in Cadarache in 6 missions. In addition a Chinese student completed his first PhD year in Cadarache. With other ITER partners, collaborations exist with the main fusion laboratories: JAEA for Japan, GA, UCSD and PPPL for USA, Mephi for the Russian Federation, and IPR for India.

Finally, it has to be noted that this document is not an extensive representation of all the collaborative activities, as it exists also less formalized scientific exchanges with other teams, in particular in the field of theory, which nevertheless play an important role for the scientific production.

country	Institute	topic	other lab(s) involved	# publis 2008	highlights
Austria	ÖAW Vienna	Validation of pellet codes (ablation + deposition), predictions for ITER		2	Improvement of the pellet code, will lead to publications in 2009
Belgium	ERM Brusssels	Ion Cyclotron Wall Conditioning	FZJ	1	Experimental studies on ICWC on Tore Supra and TEXTOR
	SKC/CEN Mol	Irradiation of IR fibres by gamma radiation and neutrons			The EFDA report was finished together with B. Brichard beginning 2008 (CEA technical note: DIAG/NTT-2007.026)
Czech Rep.	IPP.CR Prague	ELM control by ergodic fields		6	Optimisation of spectrum and scenarios of future experiments of ELMs suppression by existing RMP coils in COMPASS

country	Institute	topic	other lab(s) involved	# publis 2008	highlights
		Modelling power deposition on plasma facing components		1	Contribution to 18th PSI Conference Toledo. [R. Dejarnac, M. Komm, J.P. Gunn, R. Panek, "Power flux in the ITER divertor tile gaps during ELMs", J. Nucl. Mater (2009) in print.]
		SOL measurements with Langmuir probes			The probe was installed on schedule and successfully exploited during the 2008 campaign. Data are being analyzed in 2009.
		Kinetic code for SOL		1	Extensive work was done to interpret Tore Supra RFA probe measurements of lower hybrid hot spots using the code. A contribution was made to the EPS conference concerning ELM heat fluxes.
Germany	FZJ Jülich	He Beam Spectroscopy	PIIM Marseille		The optical design and mechanical design have been performed. A camera detector have been purchased to be implemented on the spectrometer (loan from TEXTOR team)
		chemical erosion	CP2M Marseille	3	Assessment of chemical and total erosion has been done by visible spectroscopy on the toroidal pumped limiter. Proposal experiment has been done to simulate chemical erosion with CH4 injection in TS. A bottle of CH4 has been purchased in order to perform the experiment that has not been planned yet.
		Plasma edge modelling with EIRENE and ERO codes	PIIM-U. Provence	1	First 3D calculations of recycling on the Tore Supra limiter have been performed. Comparison with spectroscopic data is underway. Results have been presented at the PSI 2008 conference.
		TPL dismantling: sample analysis	IPP, JSI, VTT Tekes, VR, CIEMAT, MEPHI		Participation to the DITS progress report
	FZK Karlsrühe	W plasma facing components for DEMO			Preliminary assessment of the possibility to examine DEMO divertor element with SATIR
	IPP Garching	turbulence studies of Asdex discharges		1	Publication in Physics of Plasmas
		TF ITM on core turbulence IMP4	CRPP/Lausanne	1	Good agreement in the first comparisons between gyrokinetic codes ORB5 and GYSELA

country	Institute	topic	other lab(s) involved	# publis 2008	highlights
		Test of Two colour Pyroreflectometer method at Asdex Upgrade	CNRS-PROMES , Odeillo	3	In December the installation of the two colour pyroreflectometer at ASDEX-Upgrade was prepared by bringing the measurement heads to ASDEX-Upgrade and defining the way the equipment should be connected.
		TPL dismantling: sample analysis	FZJ, JSI, VTT Tekes, VR, CIEMAT, MEPHI	2	NRA performed in 2008. Results shown to be consistent with TDS (IAEA 2008). Comparison with SIMS ongoing. Analysis planned for the second cutting campaign in 2009.
Ireland	DCU Dublin	Development of a negative ion source for ITER	IPHC		1. Successfully installed and commissioned CRDS diagnostic on KAMABOKO. 2.An emittance meter designed by IPHC was commissioned on the MANTIS test bed. Investigation of the effect of ion temperature and space charge on the emittance. 3. The sourcette was modified to adapt to the mantis test stand. Negative ion beams were extracted by operating the sourcette with and without the low work function grid. No enhancement in H- production was observed. It is proposed that the source geometry inhibited a large enough flux of neutrals to the low work function surface. 4. Development of a RF ion source. A number of design modifications were carried out on the MANTIS RF ion source, to allow this source to be adaptable to many different topologies and plasma generation mechanisms.
Italy	ENEA Frascati	Development of methods to use the soft X-ray emissivity of the plasma in Tore Supra for potential real-time applications		3	The Tore Supra SXR data were significantly improved as a result of a comprehensive calibration campaign of all the SXR camera detectors. Preliminary work was also performed on the qualification of a new SXR tomography algorithm (minimum Fischer) and for the preparation of a test bed for the energy resolved GEM SXR detectors in view of possible implantation of a PIXCS camera on Tore Supra
Poland	IPPLM Varsaw	Cerenkov detectors for ripple lost electrons energy measurements			First Cerenkov measurements obtained on 2 channels but no plasma parameters scan.

country	Institute	topic	other lab(s) involved	# publis 2008	highlights
Portugal	IPFN Lisboa	Software tools of the ISTTOK			We succeeded in performing a succession of pulses on the tokamak ISTTOK from the office of Joao Santos using FusionTalk to communicate with the technician working on the tokamak. FusionTalk is a multifunctional, multi-user Web based platform, that comes out to be a complete remote participation tool, with intrinsic mechanisms of data sharing and presentations display.
Sweden	VR Stockholm	Modelling of total radiative power			1D modelling of the radiated power in Tore Supra - First application to experimental data.
Switzerland	EPFL Lausanne	Gyrokinetics	IPP Garching	2	Good agreement in the first comparisons between gyrokinetic codes ORB5 and GYSELA
UK	UKAEA Culham	Adaptation of WOLFF software to MAST	IUSTI		Implementation of WOLF software for MAST
		Modelling of the deposition profile associated to pellet injection. Implementation of the corresponding code in JETTO and ASTRA.	Universit é de Vienne (Autriche)	2	Improvement of the pellet code, will lead to publications in 2009
		ITER like wall project: Tile shaping checking		2	Different plasma configurations have been studied to calculate the heat flux deposition on JET inner wall components.
		Two Colour Pyroreflectometry for JET	CNRS PROMES , Odeillo		The final report was submitted (DIAG.NTT-2008.033), with the main conclusion: the port envisaged (60 mm diameter) is too small or the laser is not powerful enough to make meaningful measurements in the present state at JET.
China	ASIPP Hefei	Collaboration on Plasma wall Interaction and Edge Plasma diagnostics			High interest for collaboration on IR thermography diagnostic
		Codes CRONOS and TPROF			Installation of CRONOS at Hefei (ASIPP); training of B. Ding; scenario simulations for EAST
	SWIP Chengdu	Supersonic Molecular Beam Injectors		1	R&D tests were performed to optimise the performance of the present facility in injecting impurities (nitrogen, neon, argon) for impurities transport studies

country	Institute	topic	other lab(s) involved	# publis 2008	highlights
		Collaboration on Plasma wall Interaction and Edge Plasma diagnostics			High interest for collaboration on PFCs, heat load deposition, IR thermography diagnostic and speckle interferometry
		ECRH experiments on Tore Supra			LH+EC and EC modulation experiments on HL-2A realised
		Development of analysis method for CXRS diagnostic		1	The assessment of genetical algorithm potential use for spectral analysis of CXRS data has been performed. Improvement of the method is foreseen.
		Development of RF heating antennas			Pre-design of LHCD and ECRH antennae for HL-2A made by Chinese visitors with the help of GCHF team,
		Fusion plasma diagnostic and heating			ECRH modulation experiments have been performed in HL-2A. Similar results have been observed than in Tore Supra, in which a strong inward heat transport has been evidenced in off-axis ECRH experiments.
		Particle Transport			Density pump-out has been observed on HL-2A during the ECRH experiments. During the ECRH phase, various phenomenologies have been observed for the particle transport behaviour: simple pump-out, double pump-out, pump-out/pinch and pump-out/pinch/pump-out. Connection to the TEM/ITG system is then analysed.
India	IPR Gandhinagar	Quench protection in forced flow superconducting magnets			In the framework of a PhD under the joint supervision of M. Duchateau (CEA) and M. Pradhan (IPR), M. Sharma (IPR) is investigating the subject of quench protection and stability in forced flow superconducting magnets. This activity is on going but it has been slowly progressing in 2008 because M. Sharma main work, during the recent period, has been the participation to the preparation and the tests of a model TF coil for SST1.

country	Institute	topic	other lab(s) involved	# publis 2008	highlights
		Review of the magnet system of SST-1 Tokamak			During the first commissioning phase of SST1 in 2006, many leaks have been detected in the joints region. To repair the joints a solution has been proposed. Applying this solution the joints are cooled only by conduction and no longer by forced flow helium. This solution has been reviewed using Ansys by CEA. A note including technical comments was sent in June 2008 to IPR. During the test of the model coil in August 2008, using this new solution, no leak was detected demonstrating the validity of the solution.
		Lower hybrid current drive simulation studies for SST1 machine using CRONOS code			1 publication in preparation
Japan	JAEA Naka	SINGAP test at the Megavolt Test Facility			test of the Singap accelerator on the MV test bed at Naka (Japan), to compare both concepts (Singap and Mamug), under the same operating conditions. The Mamug concept has reached higher performances than the Singap (less stray electrons); as a consequence, it is the Mamug concept that has been chosen for the ITER-NBI system.
Russia	Mephi Moscow	Wall conditioning and T removal techniques			A contract was signed with the company LIDASA (linked to the MEPHI laboratory) on the basis of a detailed technical specification. The main results are expected in 2009
		Deuterium retention and carbon migration		1	Last reports concerning the contract with MEPHI were sent in early 2008. Future topics of collaboration were identified, but are not implemented yet.
USA	GA San Diego	ELM control by ergodic fields in DIII-D and ITER	PPPL, Wisconsin Univ., IPP.CR, ITER Org	7	Finalisation of RMP coils for ITER based on vacuum modelling and MHD plasma response with toroidal rotation and NTV models developed essentially during ITER design Review
		Implantation of CRONOS on DIII-D & participation to experiments			Benchmark of new NBI code on DIII-D shots

country	Institute	topic	other lab(s) involved	# publis 2008	highlights
		Comparison of gyrokinetic NL et QL models		4	1) Validation of the quasi-linear model QuaLiKiz versus nonlinear gyrokinetic simulations with GYRO 2) Comparison of the turbulence measurements on Tore Supra with expectations by GYRO and GYSELA
	PPPL Princeton	Assessment of the implantation of a dust electrostatic detector in TS			Implantation of a dust electrostatic detector in TS
	UCSD San Diego	Developments of the Kepler framework			Improved the code platform Kepler according to the ITM needs

9.2 National collaborations

From 2007, the fusion research activity in France is scientifically coordinated in the frame of the FR-FCM (Fédération de Recherches sur la Fusion par Confinement Magnétique). The main goals of the FR-FCM are, in addition to its coordinating role, to establish a French Fusion research work programme integrated as much as possible in the European one. In the frame of the FR-FCM, a close support unit based in Cadarache next to the Tore Supra Tokamak but also to the IO site has been created. This organisational structure offers offices and work facilities to all the scientists involved in the French Fusion WP and it is dedicated to strengthen the contacts with all the actors involved in this WP. And last, the FR-FCM has also as duty to increase the visibility of the French fusion research program in the national and international landscape.

Based upon pre-existing active structures, FR-FCM is a formal agreement between CEA, CNRS, INRIA and 6 French Universities (Universities of Aix-Marseille I, II, III, of Nice, of Nancy and l'Ecole Polytechnique).

In 2008, about 50 projects have been supported by the FR-FCM representing more than 60 PPY for 4300k€ of budget. The research fields covered by the FR-FCM WP are MHD (5 projects), Turbulence and Transport (12 projects), Heating and Current Drive (11 projects), Plasma Wall Interaction and associated Radiating processes (9 projects). Two new fields have been introduced in 2008 concerning Materials for fusion (5 projects) and Diagnostics and Information Processing (5 projects).

All the final reports corresponding to 2008 FR-FCM projects can be found in the document EUR-CEA / CG 86 – 5A.

9.3 Technofusion

Technofusion activities are reported in the document EUR-CEA-FC-1740 (available on this CD).

9.4 Keep in touch on Inertial Fusion Energy research activities

Activities conducted in the frame of the keep in touch of the Member State's civil research activities on Inertial Fusion Energy are reported in the document EUR-CEA / CG86 – 5.1a (available on this CD).

9.5 Publication List

9.5.1 Papers in peer review journals (82)

Analysis of thermal-hydraulic gravity/buoyancy effects in the testing of the ITER poloidal field full size joint sample (PF-FSJS)
R. Zanino, P. Bruzzone, D. Ciazynski, M. Ciotti, P. Gilson, S. Nicollet, L. Savoldi
Advances in Cryogenic Engineering, vol.710, p.544-551(2004)

Observation of Hamiltonian Chaos in Wave-Particle Interaction
A. Macor, F. Doveil
Celestial Mechanics and Dynamical Astronomy, 49 (2007) B125-B135

Computing ITG turbulence with a full-f semi-lagragian code
V. Grandgirard, Y. Sarazin, X. Garbet, G. Dif-pradalier, Ph. Ghendrih, N. Crouseilles, G. Latu, E. Sonnendrücker, N. Besse, P. Bertrand
Communication in Nonlinear Science and Numerical Simulation 13 (2008) 81-87

Defining an Equilibrium State in Global Full-f Gyrokinetic Models
G. Dif-pradalier, V. Grandgirard, Y. Sarazin, X. Garbet, Ph. Ghendrih
Communication in Nonlinear Science and Numerical Simulation 13 (2008) 65-71

The way Towards Thermonuclear Fusion Simulators
A. Becoulet, P. Strand, H. Wilson, M. romanelli, L.G. Eriksson
Computer Physics Communications 177 (2007) 55-59

Simultaneous Measurements of Ion Temperature by Segmented Tunnel and Katsumata Probe
J. Adamek, M. Kocan, R. Panek, J. Gunn, E. Martines, J. Stöckel, C. Ionita
Contribution to Plasma Physics 48, No. 5-7,1-5 (2008)

Influence of a Fast Electron Component on the Plasma Sheath Structure at the Inside of the Tunnel Probe
J. Gunn, J. Gunn, G. Van oost
Contribution to Plasma Physics,48, No.5-7,497-502 (2008)

Evaluation of Thermal Gradients and Thermosiphon in Dual Channel Cable-In-Conduit Conductors
B. Renard, J.L. Duchateau, B. Rousset, L. Tadrist
Cryogenics 46 (2006) 629-642

Gas flow and related beam losses in the ITER neutral beam injector
A. Krylov, R. Hemsworth
Fusion Engineering and Design 81 (2006) 2239-2248

Design Resonant Magnetic Perturbation ELM Suppression System for JET
G. Agarici, M. Becoulet, E. Nardon, A. Saille, P. Thomas, J.M. Verger
Fusion Engineering and Design 82 (2007) 974-981

Calculation of heat flux and evolution of equivalent thermal contact resistance of carbon deposits on Tore Supra neutralizer
J. Gardarein, R. Reichle, F. Rigollet, C. Le Niliot, C. Pocheau
Fusion Engineering and Design, 83 (2008) 759-765

Mission and highlights of the JET Joint Undertaking: 1978-1999
J. Jacquinot
Fusion Science & Technology Vol. 53 mai 2008 page 866

Physics studies with the additional heating systems in JET
L.G. Eriksson, L.G. Eriksson, M. Mantsinen, M. Mayoral, D. van Eester, J. Mailloux, C. Gormezano, T. Jones
Fusion Science & Technology Vol.53, mai 2008, page 1103

Core transport studies in JET
P. Mantica, G. Corrigan, X. Garbet, F. Imbeaux, J. Lönnroth, V.V. Parail, T. Tala, A. Taroni, M. Valisa, H. Weisen
Fusion Science & Technology Vol.53, mai 2008, page 1152

Advanced tokamaks scenarios at JET
C. Gormezano, C. Challis, E. Joffrin, X. Litaudon, A. Sips
Fusion Science & Technology Vol.53, mai 2008, page 958

Burning plasma studies at JET
S. Sharapov, L.G. Eriksson, A. Fasoli, G. Gorini, J. Källne, V. Kiptily, A. Korotkov, A. Murari, S. Pinches, D. Testa, P. Thomas
Fusion Science & Technology vol.53, p.989-1022 mai 2008

Integrated plasma shape and boundary flux control on JET Tokamak
M. Ariola, G. De tommasi, D. Mazon, D. Moreau, F. Piccolo, F. Sartori, L. Zabeo
Fusion Science & Technology, vol 53, p (789-805), (2008)

Computational Images of Internal-Transport-Barrier Oscillations in Tokamak Plasmas
J. Bizarro, X. Litaudon, T. Tala
IEEE Trans. Plasma Sciences, vol. 36, N°2

Thermal quadrupoles approach for two-dimensional heat flux estimation using infrared and thermocouples measurements on the JET tokamak
J. Gardarein, Y. Corre, F. Rigollet, C. Le Niliot, R. Reichle, P. Andrew
International Journal of Thermal Sciences 48 (2009), p. 1-13

A drift-kinetic Semi-Lagrangian 4D code for ion turbulence simulation
V. Grandgirard
Journal of Computational Physics 217 (2006) 395-423

Bezier Surfaces and finite elements for MHD simulations
G. Huysmans, O. Czarny
Journal of Computational Physics, vol 227, p (7423-7445), (2008)

Characterization of carbon dust produced in sputtering discharges and in the Tore Supra tokamak
C. Arnas, C. Dominique, P. Roubin, C. Martin, C. Brosset, B. Pegourie
Journal of Nuclear Materials 353 (2006) 80-88

Aqueous corrosion in static capsule tests representing multi-metal assemblies in the hydraulic circuit of TORE SUPRA
M. Lipa, J. Blanchet, F. Cellier
Journal of Nuclear Materials, Vol. 4547, p.1 a 5, (2008)

DC glow discharge conditioning of remote areas in fusion devices
D. Douai, D. Garnier, S. Bremond, C. Grisolia, J. Bucalossi, P. Shigin, L. Begrambekov
Journal of Physics: Conference Series 100 (2008) 062034

Influence of cable layout on the performance of ITER-type Nb3Sn conductors
D. Ciazynski, L. Zani, P. Bruzzone, B. Stepanov, R. Wesche, L. Savoldi, R. Zanino, A. Nijhuis, Y. Ilyin, S. Turtù, V. Corato, G. De Marzi
Journal of Physics: Conference Series 97 (2008) 012027

Predictive study of the Poloidal Field Coil Insert behaviour under pulsed current tests
B. Lacroix, D. Ciazynski, J.L. Duchateau, S. Nicollet, N. Pauty
Journal of Physics: Conference Series, vol.97,p 1-6 (2008)

Density fluctuation measurements using X-mode fast sweep reflectometry on Tore Supra
L. Vermare, S. Heuraux, F. Clairet, G. Leclert, F. Da Silva
Nuclear Fusion 46 (2006) S743-S759

Beyond scale separation in gyrokinetic turbulence
X. Garbet, Y. Sarazin, V. Grandgirard, G. Dif-pradalier, G. Darmet, Ph. Ghendrih, P. Bertrand, N. Besse, E. Gravier, P. Morel, E. Sonnendrücker, N. Crouseilles, J. Dischler, G. Latu, E. Violard, M. Brunetti, S. Brunner, X. Lapillone, T. Tran, L. Villard
Nuclear Fusion 47 (2007) 1206-1212

Inter-machine comparison of intrinsic toroidal rotation in tokamaks
L.G. Eriksson, A. Ince-Cushman, J. Degrassie, L.G. Eriksson, Y. Sakamoto, A. Scarabosio, A. Bortolon, K.H. Burrell, B. Duval, C. Fenzi, M. Greenwald, R.J. Groebner, G.T. Hoang, Y. Koide, E. Marmar, A. Pochelon, Y. Podpaly
Nuclear Fusion 47 (2007) 1618-1624

Pellet ablation and mass deposition in FTU: analysis of vertical and low field side injection experiments
B. Pegourie, L. Garzotti, B. Pegourie, H. Nehme, D. Frigione, S. Martini, E. Giovannozzi, O. Tudisco
Nuclear Fusion 47 (2007) 288-296

First experiments of plasma start-up assisted by ECRH on Tore Supra
J. Bucalossi, E. Trier, P. Hertout, M. Lennholm, F. Saint-Laurent
Nuclear Fusion 48 (2008) 054005

Analysis of DEMO Scenarios with the CRONOS Suite of Codes
J. Garcia, G. Giruzzi, J.F. Artaud, V. Basiuk, F. Imbeaux, M. Schneider
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TOPLHA and ALOHA:comparison between Lower Hybrid wave coupling codes
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Simultaneous Measurements of Ion Temperature by Segmented Tunnel and Katsumata Probe
J. Adamek, M. Kocan, R. Panek, J. Gunn, E. Martines, J. Stöckel, C. Ionita
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L. Gargiulo, P. Bayetti, J.J. Cordier, J. Friconneau, C. Grisolia, J.C. Hatchressian, D. Keller, Y. Perrot
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A. Durocher
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X. Courtois, A. Durocher
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Estimation de la résistance de contact des dépôts carbonés recouvrant le LPT de Tore Supra à partir des températures de surface IR mesurées lors de décharges pl
S. Carpentier, Y. Corre, G. Dunand, J. Gardarein, C. Le Niliot, V. Moncada, R. Reichle, F. Rigollet, J.M. Travere
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C. Gil, D. Elbeze
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Projets optiques de l'IRFM
R. Reichle
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Détermination des propriétés thermiques de couches de carbone par thermographie
R. Daviot, E. Gauthier, S. Carpentier, Y. Corre, J. Gardarein
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Urgent R & D
R. Reichle
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Frascati (Italie), 08/07/2008 - 08/07/2008

High photon flux and spectral resolution spectrometer development for ion temperature and rotation velocity measurements in fusion devices
C. Fenzi, G. Colledani, S. Hacquin, J. MIGOZZI
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M. Davi, Y. Corre, D. Guilhem, A. Manzanares, J. MIGOZZI, R. Reichle, S. Salasca, J.M. Travere
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Multiscale turbulent transport with GYSELA
G. Dif-pradalier, V. Grandgirard, Y. Sarazin, X. Garbet, Ph. Ghendrih, P. Angelino
Seminaire UCI Irvine (Californie USA), 02/04/2008 - 02/04/2008

Multiscale turbulent transport with GYSELA
G. Dif-pradalier, V. Grandgirard, Y. Sarazin, X. Garbet, Ph. Ghendrih, P. Angelino
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A minimal collision operator for implementing neoclassical transport in gyrokinetic simulations
X. Garbet, G. Dif-pradalier, C. Nguyen, P. Angelino, Y. Sarazin, V. Grandgirard, Ph. Ghendrih, A. Samain
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MHD limits in non-inductive tokamak plasmas: simulations and comparison to experiments on Tore Supra
P. Maget, H. Lutjens, G. Huysmans, M. Ottaviani, X. Garbet, P. Moreau, J.L. Segui
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Injection supersonique d'impuretés dans le tokamak Tore Supra
D. Villegas, R. Guirlet, C. Chen, V. Marty, S. Bremond
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Prédire le transport turbulent dans les tokamaks: l'approche gyrocinétique
Y. Sarazin, V. Grandgirard, G. Dif-pradalier, X. Garbet, Ph. Ghendrih, P. Angelino, N. Crouseilles, E. Sonnendrücker, G. Latu, J. Dischler, E. Violard, P. Bertrand, N. Besse, E. Gravier, P. Morel, M. Boulet
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9.5.3 Euratom-CEA reports (2)

Fusion Technology Annual Report of the Association EURATOM-CEA 2006
T. Salmon, F. Le Vagueres
EUR-CEA-FC-1738 2007

C3PO, a ray-tracing code for arbitrary axisymmetric magnetic equilibrium
Y. Peysson, J. Decker
EUR-CEA-FC-1739 2008

9.5.4 Thesis (5)

Contrôle du profil de densité dans le plasma de Tore Supra. Comparaison de différentes méthodes d'alimentation en particules
N. Commaux
Thèse Paris 11, septembre 2007, Paris XI

Etude des Flux de Matière dans le Plasma de bord des Tokamaks: Alimentation, Transport et Turbulence.
P. Tamain
Thèse Université de Provence Aix-Marseille I, novembre 2007 Université de Proven

Etude des plasmas non-inductifs par injection d'ondes à la fréquence cyclotronique électronique dans le tokamak Tore Supra
F. Turco
Thèse Provence, 23 juin 2008 Université de Provence

First-principle description of collisional gyrokinetic turbulence in tokamak plasmas
G. Dif-pradalier
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Transfert cinétique dans un plasma de fusion magnétique à flux forcé
G. Darmet
Thèse LYON, novembre 2007 LYON

9.5.5 University grade for research leader (3)

Internal transport barriers: critical physics issues for steady-state tokamak operation
X. Litaudon
Thèse Aix Marseille I, 2008

Problèmes de reconnexion magnétique dans les plasmas de fusion
M. Ottaviani
Thèse Université de Provence I, février 2007 Université de Provence Aix-Marseill

Sources et transport des impuretés dans le tokamak Tore Supra
R. Guirlet
Thèse Aix-Marseille I, 9 juillet 2008 Universite de Provence Aix-Marseille I

9.5.6 Lectures (8)

Design, operation and manufactue of water cooled plasma faing components
R. Mitteau

Drift wave transport models and transport barriers
X. Garbet

Introduction to drift wave turbulence
X. Garbet

Introduction to the theory of confinement
X. Garbet

Master Fusion, Cours CFP (modules FCM8 et PTF8a)
R. Mitteau
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Plasma Wall Interactions
R. Reichle

Turbulence and transport in magnetised plasmas
Y. Sarazin, X. Garbet

Variational description of low frequency waves in magnetically confined plasma Ecole d'été CEA-EDF-INRIA, Nice, France
R. Dumont

9.5.7 Student's reports (12)

Analyse et traitement du signal du diagnostic polarimétrie infrarouge pour les plasmas de Tore Supra
L. Comandini

Etude du couplage des ondes à la fréquence hybride basse sur Tore Supra
B. Frincu

Fermeture fluide non-collisionnelle pour les plasmas chauds magnétisés
D. Zarzoso

Génération de courant par ondes rapides dans les plasmas de fusion magnétique
O. Izacard

Intégration et pilotage à distance de miroirs motorisés de MOU de Gyrotrons 118 Ghz du générateur FCE de Tore Supra
J. Saidi

Modélisation du dépôt d'énergie causé par les ELMs sur la paroi d'ITER
T. Derrien

Optimisation de la tomographie X-Mou sur Tore Supra en vue d'une application temps-réel
L. Deambrogio

Pertes électroniques dans Tore Supra lors du chaffage à la fréquence cyclotronique électronique
R. Goupillou

Propagation d'ondes de dérive anisotropes dans un plasma
S. Khosh aghdam

Rapport de stage de Master sur la generation de courant par l'onde hybride basse
F. Duthoit

Remise en état et amélioration de la supervision du réseau électrique
L. Galai

Simulations non-linéaires des modes MHD dans les tokamaks
F. De solminihac

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ASSOCIATION EURATOM-CEA CEA/DSM/IRFM
Centre de Cadarache, 13108 Saint-Paul-Lez-Durance, France
TEL: 33 4 42 25 46 59 - FAX: 33 4 42 25 64 21 - E-mail: dirirfm@cea.fr
http://www-fusion-magnetique.cea.fr

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