Annual Progress Report 2010
Cover Picture:

Left: Theory calculation of cylindrical ideal MHD resistive wall modes eigenvalue spectrum
Right: Empirical eigenvalue spectrum from the system identification experiment at the EXTRAP T2R fusion facility at the Royal Institute of Technology, Stockholm.

Compiled from contributions from the research groups of the Swedish Fusion Research Unit

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Head of the Swedish Research Unit 2010

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# Annual Progress Report 2010

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Preface

On behalf of the Swedish Fusion Research Unit, we are pleased to present the Annual Progress Report for 2010 covering research carried out under the Contract of Association between the Swedish Research Council (VR) and the European Atomic Energy Community, EURATOM.

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Head of the Swedish Fusion Research Unit 2010*

Edited by:
Jan Scheffel
Deputy Head of the Swedish Fusion Research Unit (from 2011)

Association EURATOM-VR

* New Head of the Swedish Fusion Research Unit (from 2011) is Professor Göran Ericsson, Uppsala University
1 EXECUTIVE SUMMARY

1.1 General introduction

Controlled thermonuclear fusion offers the prospect of an intrinsically safe, virtually inexhaustible energy source. It is seen as potentially having a key role in the long-term energy system, primarily for base load electricity production, provided it can be developed to become economically competitive. The international proof-of-principal experiment ITER plays a vital role in the development of fusion as an energy source. The mission of the ITER experiment is as follows:

- Demonstrate capability of steady state fusion power production.
- Optimise burning plasma confinement under reactor conditions.
- Have dimensions comparable to a power station and produce about 500 MW of fusion power (10 times more power than needed to run it).
- Demonstrate or develop new technologies and materials required for fusion power stations.

The Agreement on the Establishment of the ITER International Fusion Organisation for the Joint Implementation of the ITER Project is between the seven participating Domestic Agencies: the EU, Japan, USA, Russian Federation, China, South Korea and India. The European Atomic Energy Community (EURATOM) is the Domestic Agency representing the European Union in the ITER International Organisation. A Joint Undertaking for ITER and the Development of Fusion Energy (Fusion for Energy or “F4E”) was approved by the Council of Ministers in 2007. The objectives of F4E are the following:

- To provide the contribution of EURATOM to the ITER International Organisation.
- To provide the contribution of EURATOM to Broader Approach (BA) activities with Japan for the rapid realisation of fusion energy.
- To prepare and coordinate a programme of activities in preparation for the construction of a demonstration fusion reactor and related facilities including the International Fusion Materials Irradiation Facility (IFMIF).

Included in the first objective to provide the contribution of EURATOM to ITER are to procure and provide the components and equipment to the ITER Organisation and to prepare and coordinate EURATOM’s participation in the scientific and technical exploitation of the ITER project. The Fusion for Energy agreement covers 35 years from 2007 to 2041. The first plasma is scheduled for 2019 and first deuterium-tritium operation is scheduled for 2026.

The development of fusion power is a key action in the European Framework Programme and the research is co-ordinated and managed as a part of the EURATOM agreement. The delivery of the European contribution to ITER is the responsibility of F4E, however a substantial support effort is required in the accompanying European fusion programme, which is co-coordinated by the European Commission under EURATOM auspices. The work is performed by various groups in the member states under Contracts of Association (CoA). The Contract of Association establishing the Swedish Research Unit (RU) is between the Swedish Science Research Council (VR) and EURATOM. The CoA defines the roles of the
Association Steering Committee, the Research Unit and the leadership of the RU, the Head of Research Unit (HRU). The CoA also includes a Work Plan for the Association which normally spans several years.

There are additional agreements between the Associations. The European Fusion Development Agreement (EFDA) provides a framework for coordinating activities which are included in the European fusion research programme:

- Co-ordinated activities in physics and emerging technology.
- The collective use of the JET facilities, which is the largest fusion experiment now in operation and is located in the UK.
- Training and career development of researchers, promoting links to universities and carrying out support actions.
- European contributions to international collaborations that are outside the Joint Undertaking for ITER and the Development of Fusion Energy.

The long-term strategy of the European fusion programme is based on well-defined steps. Operation of present-generation experiments, in particular the JET experiment, has established a base for the design of ITER. These experiments will now be used to plan for the exploitation of ITER. The results of the ITER project together with efforts carried out in parallel should enable the next step, the construction of a demonstration reactor, DEMO. The focus is now on ITER, however progress to fusion power plants includes additional elements: (i) a test facility for materials, the International Fusion Materials Irradiation Facility, IFMIF, (ii) continued exploration of concept improvements that may, in the longer term, be attractive, and (iii) continued development of the technology required for DEMO and future power plants.

The Swedish fusion research unit encompasses a range of competencies that are important for the European fusion programme and for the ITER project. The Swedish Association has as its basic goal to make important contributions to the ITER project and to the long term goal of a prototype fusion reactor.

### 1.2 Research Unit

The formation of a Swedish Fusion Research Unit is enabled by the Contract of Association between EURATOM and the VR. Swedish fusion research activities are carried out at four universities and one industry, which together form the Swedish Research Unit. The following universities participate in the fusion research: the Royal Institute of Technology (KTH) in Stockholm, Chalmers University of Technology (CTH) in Göteborg, Uppsala University (UU) and Lund University (LU). A group at Studsvik Energy AB is also a part of the Research Unit.

The activity of the Association EURATOM-VR is directed by the Steering Committee, which during 2010 included the following members:

- **Members:**
  - Douglas Bartlett (EU RTD-J4), Ruggero Gianella (EU RTD-J4), Marc Cosyns (EU RTD-J5), Lars Börjesson (VR), Johan Holmberg (VR), Goran Bogdanovic (Ministry of Education).
- **HRU:**
  - James R Drake (KTH)
- **Secretary:**
  - Per Karlsson (VR).
The fusion plasma physics research is mainly carried out at universities and is concerned with transport, stability including active control of instabilities, plasma wall interaction, heating of the plasma, energetic particles, and diagnostic development and implementation, in particular neutron diagnostics. This research includes both experimental and theoretical work with a strong element of computer code simulation. Emerging Technology projects are in the area of tungsten and tungsten alloy development. Studsvik Energy AB has been involved in fusion technology and is now focusing on work for Fusion for Energy, the European agency responsible for fulfilling the EU responsibilities to the ITER International Organisation.

The research activities within the RU are organized in a number of research projects which are presented in the Work Plan of the RU. These activities are well integrated into the EURATOM fusion programme and the activity is a part of the accompanying programme which supports the ITER project. It includes substantial participation in the EFDA JET projects as well as collaboration with other Associations. A special feature of the Swedish fusion Research Unit is that it is university-based and involves doctoral student participation and education.

1.3 Overview of research activities

The Contracts of Association and the EFDA Agreement provide the framework for the co-ordinated European fusion research activity. All the member states with fusion research units participate in EFDA. The EFDA leader during 2010 was Mr. Francesco Romanelli. The leadership is aided by staff forming Close-Support Units. The EFDA Steering Committee, made up of representatives from the Associations that are members of EFDA, functions as a management board for EFDA. The instrument for co-ordination of the work is a Work Plan prepared by the EFDA leadership and approved by the EFDA Steering Committee. The work plan normally spans several years. An annual EFDA Work Programme, based on the Work Plan, is also prepared by the EFDA leadership and approved by the Steering Committee. The EFDA Work Programme together with the Work Plans for the Research Units is used as the basis for the annual Work Programmes prepared for each research unit. The Work Programme for the Swedish Research Unit is approved by the Association Steering Committee on an annual basis.

The Work Programme for the Swedish Research Unit for 2010 is summarised in Table 1.1. The total volume of activity in terms of Professional Person Years is about 43 PPY (a PPY is based on salaried activity). The number of professionals involved is about 61, of which 20 are PhD students. The two major areas of activity are Provision of support to the advancement of the ITER Physics Basis and Development of plasma auxiliary systems, the latter being primarily neutron diagnostics and spectroscopy–based diagnostics.

The university academic staffs are also involved in undergraduate teaching, which is not included in the PPY accounting. The training of PhD students is an important part of the activity of the academic professional staff and this activity is integrated in the research projects.
**Table 1.1. Overview of Swedish RU Work Programme activity for 2010.**

<table>
<thead>
<tr>
<th>EFDA Work Programme area</th>
<th>Approximate person power years (PPY)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Provision of support to the advancement of the ITER</td>
<td>27.3 PPY</td>
</tr>
<tr>
<td>Physics Basis</td>
<td></td>
</tr>
<tr>
<td>Energy and particle confinement / transport</td>
<td>CTH-4.7, KTH-0.3</td>
</tr>
<tr>
<td>MHD stability and plasma control</td>
<td>KTH-4.1, CTH-3.0, UU-0.9</td>
</tr>
<tr>
<td>Power and particle exhaust, and plasma wall interaction</td>
<td>KTH-4.5, UU-0.1</td>
</tr>
<tr>
<td>Physics of plasma heating and current drive</td>
<td>KTH-2.2</td>
</tr>
<tr>
<td>Energetic particle physics</td>
<td>CTH-3.0, KTH-1.1, UU-0.7</td>
</tr>
<tr>
<td>Theory and modelling for ITER</td>
<td>CTH-2.2, KTH-0.5</td>
</tr>
<tr>
<td>Development of plasma auxiliary systems</td>
<td>9.0 PPY</td>
</tr>
<tr>
<td>Neutron diagnostics</td>
<td>UU-7.5</td>
</tr>
<tr>
<td>Spectroscopy diagnostics</td>
<td>KTH-1.0, LU-0.5</td>
</tr>
<tr>
<td>Development of concept improvements and advances in fundamental understanding of fusion plasmas</td>
<td>1.1 PPY</td>
</tr>
<tr>
<td>Operational regimes and plasma characteristics for improved concepts; theory and modelling</td>
<td>KTH-1.1</td>
</tr>
<tr>
<td>Emerging technologies</td>
<td>2.7 PPY</td>
</tr>
<tr>
<td>Material science and advanced materials for DEMO (tungsten and tungsten alloys)</td>
<td>KTH-2.7</td>
</tr>
<tr>
<td>Secondments</td>
<td>2.6 PPY</td>
</tr>
<tr>
<td>CSU-JET</td>
<td>KTH-1.0, UU-1.0</td>
</tr>
<tr>
<td>JOC</td>
<td>UU-0.6</td>
</tr>
<tr>
<td>TOTAL 2009</td>
<td>42.7 PPY</td>
</tr>
<tr>
<td>Training and career development</td>
<td></td>
</tr>
<tr>
<td>All universities are involved and the personnel participation is included under the respective activity areas.</td>
<td></td>
</tr>
</tbody>
</table>

The activity of the Swedish RU is well integrated into the EFDA Work Programmes. The EFDA Work Programme is organised into Task Forces (TF) and Topical Groups (TG). There is a Task Agreement for each TF and TG where the activity undertaken by each RU is specified. In addition there is an EFDA JET Work Programme which specifies the exploitation of the jointly operated JET experiment. The JET activity is based on Scientific and Technical (S/T) tasks carried out by the Research Units under the supervision of the EFDA JET Associate Leader. A summary of Swedish RU involvement in the Work Programme for the EFDA Task Forces and Topical Groups and the EFDA JET Work Programme for 2010 is provided in Table 1.2. This table also indicates the financial provisions for additional support as specified in the CoA (Art 8.2) and the EFDA Agreement.
An indication of the level of integration of the Swedish RU into the EFDA work programmes is seen in the fact that about 57% of the Swedish RU activity is directly specified in EFDA and EFDA JET Task Agreements (40% receiving baseline support and 17% receiving additional support in some form). The principal areas where the Swedish Association focuses its activity that are included in Task Agreements of the EFDA and EFDA JET Work Programme are summarised in Table 1.2.

Table 1.2. Summary of Swedish RU involvement in EFDA Task Forces and Topical Groups and EFDA JET Work Programme for 2010.

<table>
<thead>
<tr>
<th>EFDA Work Programme area</th>
<th>Baseline support PPY</th>
<th>Additional support PPY</th>
</tr>
</thead>
<tbody>
<tr>
<td>EFDA Task Force Integrated Tokamak Modelling</td>
<td>Art 8.1</td>
<td>Art 8.2a</td>
</tr>
<tr>
<td>EFDA Task Force Plasma Wall Interaction</td>
<td>1.78</td>
<td>1.44</td>
</tr>
<tr>
<td>EFDA Topical Group Materials</td>
<td>2.0</td>
<td></td>
</tr>
<tr>
<td>EFDA Topical Group Diagnostics</td>
<td>1.95</td>
<td>0.7</td>
</tr>
<tr>
<td>EFDA Topical Group Transport</td>
<td>1.3</td>
<td>0.2</td>
</tr>
<tr>
<td>EFDA Topical Group Magnetohydrodynamics</td>
<td>3.5</td>
<td>0.9</td>
</tr>
<tr>
<td><strong>subtotal EFDA TF and TG</strong></td>
<td>12.58</td>
<td>4.64</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>EFDA JET Work Programme area for scientific and technical tasks</th>
<th>Art 6.3 Notif</th>
<th>Art 8.2c Art 6.3 Order</th>
</tr>
</thead>
<tbody>
<tr>
<td>Enhancement Project</td>
<td>0.1</td>
<td>0.2</td>
</tr>
<tr>
<td>JET Fusion Technology</td>
<td>0.7</td>
<td>(12kEur spec)</td>
</tr>
<tr>
<td>Campaigns</td>
<td>3.7</td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>EFDA Goal Oriented Training</th>
<th>Art 8.2e</th>
</tr>
</thead>
<tbody>
<tr>
<td>EFDA GOTiT</td>
<td>2.4</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>CSU Secondment</th>
<th>Art 8.2g</th>
</tr>
</thead>
<tbody>
<tr>
<td>EFDA JET CSU</td>
<td>2.0</td>
</tr>
</tbody>
</table>

| **TOTAL** | 17.08 |
| Percent of total Research Unit PPY | 40% | 17% |
1.4 Highlights of the research activity

Support to the advancement of the ITER physics base

Energy and particle confinement and transport:
- Calculated: corrections to the neoclassical plateau regime transport in transport barriers
- Analyzed: impurity transport driven by electrostatic turbulence
- Studied: radiative scenarios at JET tokamak using the 1D transport code RITM

MHD stability and plasma control:
- Assessed: automated determination of plasma response spectrum for MHD eigenmodes
- Toroidal rotation effects on the sawtooth instability clarified

Plasma wall interaction:
- Qualitative analysis of dust produced by laser induced ablation has shown the generation of fine (50 nm – 1µm) crystallites. Damage threshold values for ablative removal of several carbon materials have been determined

Physics of plasma heating and current drive:
- Code SELFO-light has been developed, allowing self-consistent modelling of RF and neutral beam/alpha heating multiple cyclotron resonances, and fast wave current drive
- Code library, RFOF, has been developed for handling RF interactions in gyro-centre following Monte Codes, allowing a large number of beam heating codes to be developed to model RF heating with very sophisticated model for the wave particle interaction

Energetic particle physics:
- Studied: plasma shutdown scenarios for ITER with injection of different impurities
- Determined: effect of Resonant Magnetic Perturbations (RMP) on high energy runaway electrons using 3D Monte Carlo code

Development of plasma auxiliary systems

Neutron diagnostics, instrumental developments:
- Proof-of-principle neutron camera installed at MAST
- Studied: Finite Larmor radius effects of energetic fuel ions on fast ion measurements

Neutron diagnostics, physics achievements:
- Studied: fast ions and their correlation to plasma instabilities
- Studied: impurity reactions and their implications for future JET experiments

Development of concept improvements
- Invitation from Nova Science Publishers, Inc to publish the new time-spectral method (GWRM) for solving initial-value problems as first chapter in book on theory of pde’s

Emerging technology
- Novel chemical routes have been developed to fabricate nanostructured tungsten powders and ODS-W composites

The major specialised equipment used by the Association includes:
The EXTRAP T2R reversed-field pinch is located at KTH
The UU group has delivered neutron spectrometers to JET (MPRu and TOFOR)
2 Support advancement of the ITER physics base

2.1 Energy and particle confinement and transport

2.1.1 Transport modelling


Introduction
This project is directed towards understanding the bulk transport in today’s fusion experiments and to find ways of improving the performance of a reactor. With “bulk transport” we here mean the transport of the thermal (or quasi thermal) particles. This is done by participating in several international working groups: (JET task forces, ITPA, the Integrated Modelling Task Force, (ITM), and the Transport Topical Group (TTG). There are also several other international collaborations going on.

Summary
The transport work during 2010 concerns effects of plasma flows, momentum transport, particle transport, fluid closure, edge plasma physics and detailed geometry effects. Effects of the reactive parallel heatflow were added to the 2008 version which has been included in the Multi Mode Model (MMM). As it turned out the parallel heatflow made very little difference. We have also continued to work on Predictive Transport Simulations, Transport Modelling and Transport Theory. We have participated in the ITPA (International Tokamak Physics Activities) Expert group meetings the EFDA-JET programme, the Transport Topical Group (TTG), including both simulations of transport barriers and impurity transport, and the Integrated Tokamak Modelling Task force (ITM-TF).

EFDA JET highlights
JET has during 2010 been shut down due to installations of “the ITER like wall” . However, simulations of Internal Transport Barriers (ITB) (mainly JET69454) have continued and during 2010 also simulations of the Edge Transport Barrier (ETB) associated with H-mode have been done. The toroidal momentum transport includes a pinch which is enhanced by finite beta effects. This effect improves confinement in most experiments (including ITER). However, it does not give an internal transport barrier by itself. This is instead due to poloidal momentum spinup which is triggered by a steep temperature gradient (due to e.g. the toroidal momentum pinch). This was now found to be the scenario in the self consistent simulation of the formation of an ITB in four channels reported already for 2009. However finite beta effects are required for these results and have been studied in more detail during 2010. We recall the good high beta results with our model from 2008 (Laborde et. al. Phys.Plasmas 15, 102507 (2008)). During 2010 also simulations of the ETB (H-mode transition) have started. For this purpose also electron ion collisions of the free electrons were included. The final trigger for the barrier formation is also here the poloidal momentum spinup. We reduced the ion and electron temperatures by a factor 5 in JET 69454 and, as usual, kept the edge
boundary fixed. This gave an ETB which restored the experimental pedestal (both height and slope) in the simulation. The slope of the ETB is mainly limited by the kinetic ballooning mode. Note that the barrier was resolved in the simulation using the same density of radial gridpoints everywhere. Also the model was the same over the whole radius so that there was not anything in the setup of the simulation that told the system that an ETB was expected. Both nonlocal and electromagnetic effects are important for both the ITB and the ETB. Later also the kink term (current gradient) was included, leading to peeling effects on the ETB. However, these results were not presented until in San Diego 2011.

During the end of 2010 an upgrade of our transport model was made so that the correlation length now depends on flowshear. This improved the results for stiffness with rotation strongly. The first results were reported by Paola Mantica at the EPFW workshop in Mayrhofen in December 2010.

Detailed comparisons between fluid and gyrokinetic model predictions and JET experimental results for turbulent impurity transport were performed. It was shown that the theoretical predictions were in good agreement with the experimental trends regarding the important scaling with impurity charge number. The impurity peaking factors for large Z-values due to ITG/TE turbulence were found to saturate at a level substantially below the neo-classical predictions. The results were presented at an invited talk at the EFDA TTG&EU-US TTF Meeting, Cordoba 7-10/9/2010.

**ITPA and TTG highlights**

We have participated in the ITPA Transport and Confinement topical group in Culham and the TTG workshop in Cordoba. In both workshops results on momentum transport and transport barrier formation were presented. In Cordoba the first simulations of the ETB were presented.

For the ITPA work the emphasis has been on the spinup of poloidal momentum in ITB. Good agreement was obtained in some cases where the ITG mode was stable in the transport barrier region while the trapped electron mode was governing the transport. We have now used also the off diagonal elements from the stress tensor which reproduce the Coriolis pinch from gyrofluid theory (included also in MMM 2008), thus we are now using only fluid theory. The toroidal effects increase both the diagonal and off diagonal transport leaving the steady state results almost unchanged but increasing the transient transport. Here also the fact that the poloidal spinup is nonlocal, due to convection of poloidal momentum from more unstable regions was reported for the first time. Also the fact that no poloidal spinup is obtained electrostatically was pointed out in Culham.

**Nonlinear Condensation Modes.**

An invited talk (JW) was given at ICTP Trieste on the trigger of condensation modes in the tokamak edge. Possibilities for nonlinear instability were found and one may consider such instabilities in connection with ELM`s. This was a completely analytical work.

**Basic nonlinear phenomena and nonlinear structures**

The spinup of poloidal momentum in our transport code is due to zonal flows. Also vortex dynamics has been studied by a nonlinear analytic model of reaction diffusion equations. The main application has been to the expansion of the universe including vortex motion. Our collaboration with the Bogoliubov Institute for Theoretical Physics on statistical problems has continued with studies of particle trapping, flowshear, partly coherent situations and fluid
closure. In particular the Waltz rule for stabilization of instabilities by flowshear was recovered. Zonal flows are also described with this model. The fluid closure was studied in more detail with sources in the Fokker-Planck equation using a Greensfunction technique. This lead to closure at the irreducible part of the fourth moment under rather general conditions. Further work on the fluid closure, comparing particle pinches (which agree with experiment in our reactive fluid model) in the reactive model and a model where Landaudamping was added showed much stronger pinch in the reactive model. Impurity transport is almost insensitive to the closure as we had seen previously. All these results on the fluid closure were partly obtained in 2011 and presented at the IFP-CNR – Chalmers workshop in Varenna in June 2011. Our previous work on particle pinches lead to an invitation from Nature Physics to write an article about the pinch in the Levitated Dipole experiment at MIT (JW, Nature Physics 6, 167 (2010)).

**Perpendicular conductivity and radial electric field control in a tokamak**

*M. Tendler*

A radial electric field in the edge tokamak plasma plays a crucial role in the formation of an edge transport barrier (ETB) and transition from a low confinement (L-mode) to a high confinement (H-mode) regime. Active control of the ETB is required to operate ITER and other tokamaks, in particular to suppress Edge Localized Modes (ELMs) which are responsible for the power bursts at the divertor plates and should be avoided during the reactor operation. The control of the ETB via active control of the radial electric field is discussed.

The models for the perpendicular conductivity of a tokamak are analyzed and compared with each other. The difference between the effective radial conductivity in a cylinder and tokamak is emphasized. It is demonstrated that the main mechanism for the perpendicular conductivity in a tokamak is the neoclassical mechanism, when the ion current is proportional to the deviation of the radial electric field from its neoclassical value. This conductivity via Einstein relation is connected to the neoclassical ion heat conductivity. The mechanism for the toroidal rotation spin up in the co-current direction is addressed.

Two groups of experiments are analyzed within the framework of the theoretical model: biasing experiments and modification of ETB by the Resonant Magnetic Perturbations (RMPs). In the first group of experiments initiated at UCLA a biased electrode is inserted inside the separatrix so that the radial current flows across the flux surfaces and further through the separatrix. The perpendicular conductivity and current-voltage characteristic are predicted by analytical models and simulations with 2D transport codes and are compared with experimental data from TEXTOR, Tuman3-M and other tokamaks. It is shown that the analytical models for the perpendicular conductivity are consistent with simulations and experiments.

Resonant coils for RMP are installed or planned on almost all large tokamaks: DIII-D, JET, MAST, ASDEX-Upgrade (AUG) and ITER. In the experiments with RMPs a radial current of electrons along a stochastic magnetic field line is generated and due to the ambipolarity constrain the same ion current across the flux surfaces is enforced to provide for the closure of currents. As a result the radial electric becomes more positive and radial convective flux of particles arises creating a pump-out effect so that the density gradient in the ETB is reduced. Models for the effective electron conductivity in a stochastic magnetic field are reviewed as
well as the results of the numerical simulations. The theories for the modification of the radial electric field and the toroidal rotation during RMPs are analyzed and compared with the results of various simulations with fluid and kinetic codes.

An important issue observed in the experiments and predicted theoretically is the self-screening of the RMPs by parallel currents. It is demonstrated that this effect can yield multiple solutions for the radial electric field and hence for the ETB structure. The predictions for ITER ETB control are addressed.

### 2.1.2 Particle and impurity transport

*T. Fülöp, I. Pusztai, F. Tamas, G. Papp*

The focus of this project is to study turbulent and neoclassical transport in the presence of impurities and large density- and temperature gradients. During 2010, we have developed a model for impurity transport driven by electrostatic turbulence and used it to determine quasilinear particle fluxes. Furthermore, we calculated corrections to the neoclassical plateau regime transport in transport barriers; it is found that the ion temperature gradient drive of the bootstrap current can be enhanced significantly in the presence of strong radial electric fields.

**Impurity particle transport in tokamak plasmas**

Impurity transport driven by electrostatic turbulence has been analyzed in weakly collisional tokamak plasmas using a semianalytical model based on a boundary layer solution of the gyrokinetic equation. Analytical expressions for the perturbed density responses were derived and used to determine the stability boundaries and the quasilinear particle fluxes. Scalings of the mode frequencies and quasilinear fluxes with charge number, effective charge, impurity density scale length, and collisionality were determined and compared to quasilinear gyrokinetic simulations with GYRO. An analytical approximate expression of the zero-flux impurity density gradient was derived and used to discuss its parametric dependencies.

**Neoclassical plateau regime transport in transport barriers**

In tokamak pedestals with subsonic flows the radial scale of plasma profiles can be comparable to the ion poloidal Larmor radius, thereby making the radial electrostatic field so strong that the $E \times B$ drift has to be retained in the ion kinetic equation in the same order as the parallel streaming. The modifications of neoclassical plateau regime transport - such as the ion heat flux and the poloidal ion and impurity flows - are evaluated in the presence of a strong radial electric field. The altered poloidal ion flow can lead to a significant increase in the bootstrap current in the pedestal where the spatial profile variation is strong because of the enhanced coefficient of the ion temperature gradient term near the electric field minimum. Unlike the banana regime, orbit squeezing does not affect the plateau regime results.

**Publications section 2.1.2**

**Peer reviewed journals**


**Presentations at conferences**


### 2.1.3 Integrated modelling

**P. Strand, H. Nordman, Y. Liu, D. Yadykin**

Integrated Modelling activities has been promoted within a few different projects, mainly EFDA Integrated Tokamak Modelling Task Force, in support of the JET Transport code project and in terms of infrastructure and framework development support under the EUFORIA – EU Fusion for ITER Applications which is not funded under Euratom but was created to support EFDA fusion applications on European grid and HPC infrastructures.

During 2010 the involvement in Integrated Tokamak Modelling Task Force changed from providing the Scientific and administrative lead as Pär Strand, Chalmers stepped down as Task Force leader in October to more direct active involvement in the Integrating Modelling Projects (IMP’s) where Chalmers are actively participating in IMP 12 (Equilibrium, MHD, and Disruptions), IMP3 (Transport Code and Discharge Evolution) and IMP4 (Turbulence and microinstabilities). A leadership role in the ITM is maintained through KTH in IMP 5 (Heating and current drive) where Thomas Johnson is a deputy project leader in IMP5 in charge of integration of H&CD codes into the ITM-TF infrastructure.

Activities related to ITM were in IMP12 devoted to the integration of the MARS-F and CarMa codes into the ITM structure. Kepler actors for both codes were made available and successfully tested. A modular component to describe the vacuum region was developed for ITM. This may be used by a range of linear MHD stability codes in a portable fashion and would generalize vacuum calculations on ITM. The integration of the CHEASE equilibrium code with MARS-F is in progress to provide a RWM stability chain (similar to existing HELENA-ILSA chain). For IMP4 work on the integration of the generic interface to anomalous transport codes, TCI, was started. TCI provides standardized interfaces to GLF23, RITM, Weiland and the EDWMmodel (a version of the transport model developed at Chalmers extended to an arbitrary number of impurity species). The work continues in 2011 with full integration into the European Transport Solver (ETS) under IMP3. Further development work is ongoing in all areas.
Support work on JET for the TCI code(s) supporting the integration of the code into JETTO continued during 201. This was also used in the Integrated Scenario Modelling Group activities which in turn also featured direct participation from Chalmers.

Activities related to the EUFORIA project were focused on the RWM stability studies in the multi-parametric space using Grid infrastructure. These activities were performed in collaboration with the colleagues from the Poznan Supercomputing and Networking Center, Poznan, Poland. Using MARS-F code as a basis, Kepler interface was created between the code and the Grid. This provides an effective parallelization of the code that made possible time effective RWM stability studies along several parametric axes simultaneously. The studies of the RWM stability were performed for the ITER geometry including the models for the uniform resistive wall and active control coils. On the infrastructure side Chalmers provided access to grid computing facilities for the EUFORIA and FUSION virtual organisations (VO) The EUFORIA project was coordinated by Chalmers (Pär Strand) and formally ended end 2010.

Publications section 2.1.3

Peer reviewed journals


5. V.F. Pais, S. Balme, H.S.Akpangny, F. Iannone, P. Strand Enabling remote access to projects in a large collaborative environment, FUSION ENGINEERING AND DESIGN 85 (3-4):633-636 2010

Presentations at conferences

2.1.4 Pedestal properties and confinement in JET

L. Frassinetti

External magnetic perturbations are an essential tool in fusion plasma physics to address severe problems related to Edge Localized Modes (ELMs), magnetic islands and Neoclassical Tearing Modes (NTMs). Resonant Magnetic Perturbations (RMPs), by interacting with the corresponding plasma Tearing Modes (TM), can (i) produce stochastic magnetic field in the pedestal region mitigating ELMs, (ii) induce rotation for NTM-stabilization or (iii) move the magnetic island to an optimal position for Electron Cyclotron Current Drive (ECCD) stabilization. While theoretical studies on the interaction between RMPs and TMs are relatively developed, recent experimental works are focused mainly on the RMP effect on global plasma parameters. From an experimental point of view, the effect of an RMP on the TM dynamics and on the plasma flow is not yet studied in a controlled fashion.

The work is performed along two main lines of research.

1) The advanced feedback algorithms developed on EXTRAP T2R are used to produce RMPs in order to study the underlying physics that regulates the interaction of the RMP itself with the plasma. The capabilities of the EXTRAP T2R feedback system allow the generation of “clean” RMPs, i.e. without secondary harmonics originated by aliasing and side-band effects. This greatly simplifies the study of the RMP effect if compared to similar studies on tokamaks, where several side-bands harmonics are present.

It is found that a static RMP affects the rotating TM by amplifying and suppressing the corresponding magnetic island (depending on their relative phase) and by producing acceleration and deceleration of the island velocity. The global effect is a plasma velocity reduction mainly localized at the resonant radius that spreads to the core and to the edge. From the velocity reduction profile and from the torque balance equation, it was possible to estimate the plasma dynamic viscosity: \( \nu = 10^{-7} \text{kg/(m·s)} \).

2) The RMP studies are extend to JET, where their effect on the pedestal properties has been studied, with particular emphasis on the density pump-out phenomenon. It was found that pellet and gas injection techniques can compensate the density pump-out effect.

The pedestal studies have then been extended to a multi-machine comparison of the pedestal properties. The pedestal behaviour in JET, DIII-D and ASDEX has been compared, showing that in all three devices the pedestal width in normalized poloidal flux scales with \( \sqrt{\beta_p} \).

Figure 2.1.1. Example temperature pedestal profiles and their fits for JET with an \( \text{mtanh} \) fit to all data points. The \( \text{mtanh} \) fits deconvolved are shown in red. Without deconvolution, the fit indicates a width of \( w = 4.28 \pm 0.25 \text{cm} \), after deconvolution, the actual pedestal width is \( w = 2.26 \pm 0.21 \text{cm} \).
The measurements of the electron temperature and pedestal properties in JET are strongly dependent on the experimental settings of the JET High Resolution Thomson Scattering (HRTS). Due to the shape of the HRTS instrument function, the measured pedestal width can be significantly overestimated [Frassinetti et al., submitted to Rev. Sci. Instrum.]. The recent implementation of a deconvolution technique to compensate the instrument function effect opened up the possibility of detailed studies on the temperature and density pedestal behavior in the JET H-mode plasmas and the possibility of pedestal multi-device comparison.

Figure 2.1.1 shows the effect of the deconvolution (continuous line) in JET. The deconvolved profile (red line) has a pedestal width 2cm smaller than the experimentally measured profile (empty dots). The instrument function effect on the pedestal width can be significant and depends on the HRTS configuration, on the plasma equilibrium and on the actual pedestal width.

**ELM suppression via RMP and density pump-out effect in JET**

One of the most promising techniques to suppress edge localized modes (ELMs) is the application of external resonant magnetic perturbation (RMP) that reduces the pedestal pressure gradient by creating a stochastic region near the pedestal itself. A drawback of this technique is the density pump-out effect, i.e. a reduction in the pedestal electron density.

Figure 2.1.2 shows the time evolution within the ELM cycle of the pedestal height (first row) and pedestal width (second row) for electron temperature, density and pressure. The density pump-out effect is evident when the RMP is applied, see the blue dots in figure 2.1.2c.

To compensate this effect, techniques such as pellet and gas injection have been applied during the RMP experiments in JET. The aim of this work was to study the behavior of the pedestal properties during these experiments.

On the basis of previous experience, various particle fuelling rates have been tested on two different divertor configurations, finding those for which the compensation takes place. In terms of plasma
confinement, while core pressure is found to be almost unvaried with the particle fuelling, the edge pressure pedestal improves towards the value obtained during the H-mode without the application of the magnetic perturbation.

This is shown in figure 2.1.3(a) in which the pressure profiles selected in the 80%-99% of the ELM cycle are plotted for a standard H-mode plasma, (blue dots), a plasma with ELM mitigated by the RMP (red dots) and a plasma with ELM mitigated by RMP plus gas injection compensation (green dots). The continuous lines in figure 2.1.3a show the deconvolved profile. Figure 2.1.3b show the pressure gradient at the pedestal for deconvolved and not-deconvolved fits (continuous and dashed lines respectively). The pressure gradient with gas injection is comparable to the pressure gradient of a standard H-mode without RMPs (green and blue lines).

**Multidevice pedestal scaling experiments in the DIII-D, AUG, and JET**

The work related with the numerical tool development for the characterization of the pedestal properties in JET has been extended in order to perform a multi-machine comparison. The numerical tools developed for JET have been adapted to DIII-D and Alcator C-mod. The work on Alcator C-mod is still in progress, but results of the pedestal property comparison in JET and DIII-D are already available. The work has been extended to ASDEX-Upgrade (AUG), even if the VR has not yet been directly involved.

At this stage the work was focused on multidevice pedestal scaling experiments in the DIII-D, ASDEX Upgrade, and JET tokamaks in order to test two pedestal width models. All three devices show a scaling of the pedestal width in normalized poloidal flux as \( \sqrt{\beta_p} \), see figure 2.1.5. The underlying width dependence in EPED1 model, \( \Delta_p = \sqrt{\beta_p} \), has been tested against pedestal width measurements from three experiments. On AUG and JET, it has been observed that the scaling can be explained in terms of a constant pedestal width measurement in radial outer midplane coordinates and a flux surface compression at increasing global beta poloidal.

**Stiffness mitigation and improved ion confinement**

The study of the electron temperature \( T_e \) and density \( N_e \) pedestal properties have then been extended to the study of the entire \( T_e \) and \( N_e \) profiles in JET H-mode baseline and hybrids plasmas. This gave a contribution to the understanding of the ion heat transport and stiffness in JET.

Ion Temperature Gradient (ITGs) modes feature a threshold in the inverse ion temperature gradient length (\( R/L_{T_i} = R|\nabla T_i|/T_i \), with R the tokamak major radius) above which the ion heat flux (\( q_i \)) increases strongly with \( R/L_{T_i} \). This property leads to stiffness of \( T_i \) profiles with respect to changes in heating profiles.
In the core of hybrids plasmas the rotation is high but smooth, and the flow shearing rate is ~3-4 $10^4$ s$^{-1}$.

The profiles then lie well above ion threshold even at small $q_i^{GB}$, indicating low stiffness. In the outer region the stiffness level is high, with normalized gradient length $R/L_{Ti}$ in the range 4-6. The dependence of the normalized gradient length $R/L_{Ti}$ vs the magnetic shear $s$ at low and high rotation is plotted in figure 2.1.4 from a JET H-mode and Hybrid database. The scatter of points is due to the range in parameters in the database. The two clouds clearly separate at low s, with much larger $R/L_{Ti}$ at high rotation. This evidence lead to suggest that stiffness mitigation in the broad low $s$ region is at the origin of the improved core ion confinement [Mantica P. et al. Phys. Rev. Lett. 2011].

**Publications section 2.1.4**

**Peer reviewed journals**


**Other workshops and conferences**


5. Marc Beurskens, Jerry Hughes, Tom Osborne, Lorenzo Frassinetti, Rich Groebner, Philip Snyder, Philip Schneider, Elisabeth Wolfrum. *H-mode Pedestal Scaling in C-mod, DIII-D, ASDEX Upgrade and JET*, ITPA Pedestal and ELMs meeting, Boston, 30th March 1st April 2011.

2.2 MHD stability and plasma control

2.2.1 Active MHD mode control

P. Brunsell, E. Olofsson (PhD student), M.W.M. Khan (PhD student), L. Frassinetti, J. R. Drake
In collaboration with:
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G. Marchiori, A. Soppelsa, G. Manduchi, RFX Team, Consorzio RFX,
E. Witrant, UJF-INPG/GIPSA-Lab, Grenoble, France,
C. R. Rojas, H. Hjalmarsson, EES/Automatic Control, KTH

Project in brief
The project scope is the development of methods for active MHD mode control applicable for
tokamak and reversed field pinch devices. A specific aim is the controller design for resistive
wall mode (RWM) stabilization at the ASDEX Upgrade tokamak, using the set of control
coils recently installed on the device. Algorithms for RWM control are presently tested on the
EXTRAP T2R reversed field pinch, which provides valuable experience on practical MHD
control system design and implementation.

System identification signal processing methods is assessed for automated determination of
the plasma response spectrum for MHD eigenmodes. No assumptions on the geometry of the
eigenmodes are imposed. It is possible to directly map the most unstable autodetected
empirical system pole to the corresponding theoretical resistive shell MHD eigenmode. A
basic scheme of system identification is the characterization of system dynamics from
sampled input and output data. The overall geometry of the eigenmode set is comparable to
theory. The method needs refinements in terms of quality control and error bound
computations, but it seems likely that system identification for control could become an
important aid in the development of accurate MIMO model-based feedback systems.

The recent development and application of system identification methods have yielded plasma
response parameters for reversed field plasmas in EXTRAP T2R. These data sets are used for
synthesis of a Fourier mode decoupled control system. Tracking of total radial magnetic field
reference, or m=1 radial magnetic field amplitude reference are both possible. Both methods
are implemented in the same framework. Although Fourier mode controllers have been tested
before, they have not been tuned based on system identification data, which is a novelty of the
present controller.

The ASDEX Upgrade enhancement project for active MHD control is carried out in
collaboration between Max-Planck-Institut für Plasmaphysik, Forschungszentrum Jülich,
Consorzio RFX and Royal Institute of Technology (KTH). KTH involvement in the project is
mainly in the design of the local controller. The internal control coils are arranged as three
toroidal rings of eight coils each at different poloidal positions at the low field side inside the
ASDEX Upgrade vacuum vessel. Four coils above the midplane and four coils below the
midplane are currently operational, another eight coils are scheduled for installation in the
near future.
Introduction
The research program on active MHD control aims at the development of methods applicable for tokamak and reversed field pinch devices. A specific aim is the development of a controller design for active resistive wall mode (RWM) stabilization at the ASDEX Upgrade tokamak, using the set of control coils recently installed on the device. Algorithms for RWM control are presently tested on the EXTRAP T2R reversed field pinch, which provides valuable experience on practical MHD control system design and implementation.

A general template for process control systems has been adapted for the MHD mode control; system identification followed by controller setup based on identification results. A mode controller has been implemented that decouples the control of individual Fourier modes. The basic application is resistive wall mode control but the methodology could also be used for to produce and maintain well defined resonant magnetic perturbations (RMPs) for edge localized mode control in tokamaks.

EXTRAP T2R device
The EXTRAP T2R reversed field pinch device, shown in Fig. 2.2.1-1, is equipped with a thin shell, which enables studies of resistive wall mode stability.

![EXTRAP T2R device at Alfvén Laboratory](image)

The main parameters of the device are shown in the table below.

<table>
<thead>
<tr>
<th>Table 1. EXTRAP T2R parameters</th>
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<tbody>
<tr>
<td>Parameter</td>
</tr>
<tr>
<td>Major radius</td>
</tr>
<tr>
<td>Minor radius</td>
</tr>
<tr>
<td>Wall diffusion time</td>
</tr>
<tr>
<td>Plasma pulse length</td>
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<tr>
<td>Plasma current</td>
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<tr>
<td>Plasma electron temperature</td>
</tr>
<tr>
<td>(typical)</td>
</tr>
<tr>
<td>Plasma electron density</td>
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</tbody>
</table>

Fig. 2.2.1-1. EXTRAP T2R device at Alfvén Laboratory
An extensive array of external control coils provides excellent capabilities for research on active magnetic feedback control. The active control system was developed and installed on EXTRAP T2R during 2004-2006 in collaboration with Consorzio RFX. The arrays of active coils and sensors distributed over the toroidal surface are shown in Fig. 2.2.1-2.

Fig. 2.2.1-2. Two-dimensional arrays of sensor flux loops and active saddle coils installed at EXTRAP T2R.

The main features of the active control system are:

- 128 magnetic flux loop sensors at 4 poloidal and 32 toroidal positions inside the thin shell.
- 128 active saddle coils at 4 poloidal and 32 toroidal positions outside the thin shell. Saddle coils and sensor flux loops are pair-connected at each toroidal position to form 64 independent m=1 coils and sensors.
- 32 power amplifiers units providing at total of 64 independent channels. Audio amplifiers are used with output power of 800-1200 Watt providing radial magnetic field at the coil centre of about 3 mT.
- An integrated digital controller unit, contained in one VME bus crate including CPU board, ADCs and DACs. Control algorithms are implemented in software.

**System identification**

The usage of computationally feasible system identification signal processing methods is assessed for automated determination of the plasma response spectrum for MHD eigenmodes. No assumptions on the geometry of the eigenmodes are imposed. It is possible to directly map the most unstable autodetected empirical system pole to the corresponding theoretical resistive shell MHD eigenmode. A basic scheme of system identification is the characterization of system dynamics from sampled input and output data. It is assumed here that the MHD mode dynamics can be approximated by a linear time-invariant discrete-time multi-input multi-output (MIMO) system. An experiment designed to provide information on the plasma response needs to excite a rich-enough range of frequencies in space and time. In this study a randomized perturbation is used, so called dither injection. Figure 2.2.1-3 shows the signal schematic for closed-loop dither injection experiments in EXTRAP T2R.
The experimental data obtained is used for predictor-based multivariate state-space system estimation. The result of system identification is shown in Figure 2.2.1-4. Three branches of poloidal mode numbers $m$ are present together with spatial aliasing of higher toroidal mode numbers $n$. The most unstable mode empirically observed ($n=11$) is very nearly monochromatic and corresponds directly to the most unstable theoretical mode for typical RFP equilibria in EXTRAP T2R. However, the other eigenvectors typically do not project with a convincing one-to-one mapping to theory. The overall geometry of the eigenmode set is nevertheless comparable to theory. The method needs refinements in terms of quality control and error bound computations. It seems likely that system identification for control could become an important aid in the development of accurate MIMO model-based feedback systems.

Fig. 2.2.1-4. Left: Theory calculation of cylindrical ideal MHD resistive wall modes eigenvalue spectrum (in the time-discrete system). Right: Empirical eigenvalue spectrum from the system identification experiment. The eigenvalue stability criterion is $|\lambda|<1$ as implied by the conversion between discrete- and continuous-time $\gamma$ growth rates: $\gamma = \tau_s^{-1} \ln|\lambda|$ where $\tau_s$ is the sampling interval.
**Mode controller based on system identification data**

The recent development and application of system identification methods have yielded plasma response parameters for reversed field plasmas in EXTRAP T2R. These data sets are used for synthesis of a Fourier mode decoupled control system. Robustness is assessed by calculation of the sensitivity transfer function. Tracking of total radial magnetic field reference, or m=1 radial magnetic field amplitude reference are both possible. Both methods are implemented in the same framework. Although Fourier mode controllers have been tested before, they have not been tuned based on system identification data, which is a novelty of the present controller.

Independently growing helical Fourier modes have been observed experimentally in the RFP, as predicted by linear MHD theory. A feedback system that identifies and reacts on predefined spatial patterns could generally be denoted a mode controller. Two complications arise while designing this system: First, experimental parameters for the plasma dynamics are needed, and second, the experimental spatial eigenmode structure is required. This implementation uses data obtained in system identification experiments. The Fast Fourier Transform (FFT) routine is used for decoupling of modes. Digital controllers are synthesized for each of the FFT components that achieve good closed loop performance.

The tracking capabilities are verified experimentally, as shown in Figure 2.2.1-5. The reference signals are simply square-pulse and cosine-pulse, for static and rotating modes respectively.

![Fig.2.2.1-5.](image)

Measurements from radial field sensor coils. Output control verification of m=1 modes stabilized and sustained at a non-zero reference amplitude.

a) Sequence of static-phase modes, n=1, 3, 5, 7.

b) Rotating-phase modes, n=1, 5.
**Development of MHD mode control at ASDEX Upgrade**

The ASDEX Upgrade enhancement project for active MHD control is carried out in collaboration between Max-Planck-Institut für Plasmaphysik, Forschungszentrum Jülich, Consorzio RFX and Royal Institute of Technology (KTH). ASDEX Upgrade is currently being enhanced with 24 in-vessel saddle coils for (i) Edge Localised Modes (ELM) mitigation by Resonant Magnetic Perturbation (RMP), (ii) locking of plasma modes to rotating magnetic error fields for disruption avoidance and (iii) feedback control of Resistive Wall Modes (RWM). KTH involvement in the project is mainly in the design of the local controller that produces the request waveforms for the coil power supplies based on supervising commands from the ASDEX Upgrade control system and measurements of a set of poloidal field sensors in the ASDEX Upgrade vacuum vessel.

The internal control coils are arranged as three toroidal rings of eight coils each at different poloidal positions at the low field side inside the ASDEX Upgrade vacuum vessel. The coils will be used to produce stationary or time-dependent perturbation fields with toroidal mode numbers up to n=4. Four coils above the midplane and four coils below the midplane are currently operational. These coils have five turns each and create a mainly radial field with toroidal mode numbers up to n=2. Another eight coils are scheduled for installation in the near future.

**Publications section 2.2.1**

**Peer reviewed journals**


**Presentations at conferences**

2.2.2 Resonant magnetic perturbation effects on plasma rotation

L. Frassinetti, E. Olofsson, P. Brunsell, J. R. Drake, W. Khan

Feedback algorithm for RMP generation

The EXTRAP T2R feedback system (active coils, sensor coils and controller) is used to study and develop new tools for an advanced control of the MHD instabilities in fusion plasmas. The new feedback algorithms developed in EXTRAP T2R reversed-field pinch allow a flexible and independent control of each magnetic harmonic. The methods developed in control theory and applied to EXTRAP T2R allow a closed-loop identification of the machine plant and of the resistive wall modes growth rates. The plant identification is the starting point for the development of output-tracking algorithms which enable the generation of external magnetic perturbations.

An example of the performance of the algorithms developed in EXTRAP T2R for the generation of an external perturbation is shown in figure 2.2.2-1. In the discharge of figure 2.2.2-1, the reference signals are zero for all the harmonics except for the $(m,n) = (1,-12)$ which is set to 0.4mT between 10 and 30 ms. The active coil current necessary to generate this spectrum is shown in figure 2.2.2-1 (a) and the measured result at the sensor coils is shown in figure 2.2.2-1 (b). The algorithm is indeed able to keep the perturbation at a constant level for the desired time length. All the other harmonics, including error fields and RWMs, are still suppressed to a very low value. Note that to obtain a ‘clean’ result, the work done by the active coils is not obvious. The feedback must simultaneously

1. suppress the error fields,
2. suppress the RWMs,
3. generate the external perturbation,
4. consider the plasma response to keep the perturbation to a constant amplitude.

This is clear in figure 2.2.2-1 (a), where the evolution of the active coil current spectrum is shown. The strong activity corresponding to the harmonics ±2 is needed to suppress the main error fields of EXTRAP T2R. Activities for the RWM suppression is also present. But most interesting is the work needed to generate and control the perturbation. This is shown in figures 2.2.2-1 (b) and 2.2.2-1 (c), where the time evolution of the amplitude and the phase of the active coil current harmonic $(1,-12)$ is plotted.
RMP effect on a rotating tearing mode island

An external magnetic perturbation with a harmonic resonant inside the plasma (RMP) can interact with the corresponding TM and affect its dynamics. The experimental behaviours can be reasonably explained using the model described by Fitzpatrick and Guo and adapted to EXTRAP T2R.

With a RMP, the electromagnetic torque can strongly modulate the velocity of the island that will spend more time around a stable equilibrium position. In figure 2.2.2-2(a) the experimental angular distribution of the island position is plotted for a plasma with no RMP (thin line) and with RMP (thick line). Indeed, for the case with no RMP the distribution is uniform. Interestingly, with the RMP the distribution has two peaks, separated of \( \approx \pi \).

From the experimental point of view, this behaviour can be understood by considering the time evolution of the TM phase. The TM is initially rotating at its natural velocity, but as soon as the RMP is switched on, the TM starts to decelerate. The TM is significantly slowed down after \( \approx 10 \) ms as shown in figure 2.2.2-2(b). In this time window, the TM is not rotating with a steady velocity but neither is completely locked, since every 10–20\( \mu\)s a \( \pi \) phase jump occurs. This behaviour is the reason of the double peak in figure 2.2.2-2(a). The corresponding behaviour of the TM amplitude is shown in figure 2.2.2-2(c). The amplitude is strongly modulated. Figures 2.2.2-2(d), 2.2.2-2(e) and 2.2.2-2(f) show a comparison with the theoretical model. A reasonable agreement between experimental and modeled results is found.

Plasma viscosity estimation and RMP effect on a plasma flow

RMP are used in tokamaks to suppress or mitigate ELMs. But the perturbations also have negative effects on the plasma performances, such as the flow braking. The braking is due to the interaction of a rotating tearing mode with the corresponding static external perturbation and/or to neoclassical toroidal viscosity effects generated by non-resonant external side-band harmonics.

The EXTRAP T2R feedback system is used to generate well controlled RMPs in order to study the underlying physics related to the flow braking. The radial profile of the velocity braking in EXTRAP T2R when RMPs are applied is shown in figure 2.2.2-3. The red lines correspond to the effect produced by the RMP with harmonic \( n = -12 \). The maximum velocity variation occurs

Figure 2.2.2-2
(a) Distribution of the angular position of the magnetic island \((1, -12)\) with no RMP (thin line) and with RMP with harmonic \((1, -12)\) and amplitude 0.5mT (thick line). (b) and (c) show the experimental time evolution of the TM phase and TM amplitude with the RMP applied. (d) shows the modelled distribution of the island position with RMP. (e) and (f) show the time evolution of the modelled TM phase and TM amplitude with RMP.

Figure 2.2.2-3
Velocity variation profiles for the RMP with harmonic \( n = -12 \) (red) and \( n = -15 \) (blue). The shaded areas corresponds to the regions were the harmonics \( n = -12 \) and \( n = -15 \) are resonant.
in the core, where the harmonic \( n = -12 \) is resonant. The blue lines correspond to the effect produced by the RMP with harmonic \( n = -15 \). The maximum velocity variation occurs at \( r/a = 0.4 \), where the harmonic \( n = -15 \) is resonant. This behaviour is consistent with the idea that the plasma flow braking is generated by the interaction of the static RMP with the corresponding rotating TM. Then, the TM velocity deceleration spreads from the resonant radius to the rest of the plasma via the viscous torque effect.

From a quantitative point of view, the plasma braking is related to the amplitude of the external perturbation and to plasma characteristics such has the viscosity. The viscosity \( \nu(r) \) can be estimated from the torque balance equation, in which the temporal change in the flow variation profile \( \Delta \Omega(r) \) is related to the electromagnetic torque \( T_{EM} \), which slows down the flow by acting on the TM velocity, and the viscous torque, which tends to re-establish the unperturbed velocity [6,7]:

\[
\rho \frac{d \Delta \Omega}{dt} = \frac{1}{r} \frac{d}{dr} \left( r v \frac{d \Delta \Omega}{dr} \right) - \frac{T_{EM}}{4 \pi \mu_0 r} \delta \rho \frac{r}{r_0}
\]

where \( \rho \) is the plasma density and \( R \) the major radius. To estimate the viscosity, a profile \( \nu(r) = \nu(T + cr) \) is assumed. The free parameters are calculated in order to obtain the best fit between simulated and experimental velocity variation profiles, as shown in figure 2.2.2-4(a). Errors are estimated by repeating the best fit changing the equilibrium and the density within the experimental uncertainties. The viscosity profile along with the classical viscosity are plotted in figure 2.2.2-4(b).

The viscosity is \( \approx 10^{-7} \text{kg/(m·s)} \) and is anomalous, being one order of magnitude larger than the classical viscosity.

**Publications section 2.2.2**

**Peer reviewed journals**


**Presentations at EPS conferences**

2. M. W. M. Khan, P. R. Brunsell, L. Frassinetti, *Effect on plasma flow of non-axisymmetric fields in EXTRAP T2R*
3. S. Menmuir, L. Frassinetti, P. R. Brunsell, J. R. Drake *Response of plasma velocity profile and ion temperature in EXTRAP T2R to variation of experimental conditions*
2.2.3 MHD stability

C. Wahlberg, in collaboration with CRPP Lausanne and UKAEA Culham

The main focus of this work is on the GAMs and global MHD oscillations and instabilities existing in the core region of tokamaks and, in particular, the influence of toroidal plasma flows on such modes. The activity during 2010 has dealt both with the effects of toroidal plasma flows on the internal kink mode and on the appearance of Kelvin-Helmholtz (KH) and combined KH-infernal instabilities in rotating plasmas with strong flow shear.

The sensitivity of the stability of the ideal $n = 1$ internal kink mode to variations in the plasma profiles has been analysed both analytically and numerically in rotating tokamak plasmas. These stability analyses have been carried out including the centrifugal effects of toroidal plasma rotation upon the equilibrium, and also inconsistently when the equilibrium is treated as static. The change in plasma stability due to rotation is partially (consistent equilibrium) or wholly (inconsistent treatment) determined by the radial profiles of the plasma density and rotation velocity. It is found that the internal kink mode stability is strongly influenced by small variations in these plasma profiles. The implications of this extreme sensitivity are discussed in Ref. [1], with particular reference to experimental data from MAST.

Some of the results from Ref. [1] are also presented in Ref. [2], highlighting especially the analytical aspects of the work and the influence of the rotation-induced GAM for the plasma stability. Here, analytical theory and results from two different MHD stability codes are compared. The numerical computations are based on one code where the centrifugal effects from the rotation are included in the equilibrium (CASTOR-FLOW), and on another code where such effects are not taken into account (MISHKA-F). In the absence of centrifugal effects, both the analytical theory and the code calculations show that the major flow effects on the internal kink instability come from the radial profiles of the plasma density and rotation velocity inside $q = 1$. With the centrifugal effects included, these profile effects persist, but an additional, stabilising effect appears from a coupling of the internal kink mode to the low-frequency GAM induced by the plasma rotation. While the latter effect dominates when the aspect ratio is large, the profile effects become increasingly important with decreasing aspect ratio.

Investigations of the appearance of Kelvin-Helmholtz (KH) type of instabilities and combined infernal-KH type of instabilities in tokamak plasmas with strong toroidal rotation and rotation shear, e.g. in NBI heated spherical tokamaks, was initiated during 2010. First results show the existence of such modes, localized in regions of strong flow shear, at Alfvénic flow velocities. Good agreement between analytical theory and numerical computations with CASTOR-FLOW has been obtained [3].

Publications section 2.2.3


### 2.3 Power and particle exhaust, Plasma-wall interaction

#### 2.3.1 Plasma-wall interaction

*M. Rubel, D. Ivanova (PhD student), P. Petersson, G. Possnert*

**Fuel Removal and Cleaning of Plasma-Facing Components**

Generation and in-vessel accumulation of carbon and metal dust are perceived to be serious safety and economy issues for a steady-state operation of a fusion reactor, e.g. ITER. The major concern is related to the risk of explosion under massive air or water leak on hot dust and to fuel retention in particles loosely bound to surfaces. Another point is the performance degradation of vital elements in spectroscopy and imaging systems, i.e. mirrors and windows. Therefore, the total dust and tritium inventory must be monitored and minimized by scheduled cleaning procedures. Among proposed methods for in-situ fuel removal there are: photonic cleaning and long-term annealing of plasma-facing components (PFC) at elevated temperatures.

This work was focused at: (a) dust generation associated with laser-induced removal of fuel and co-deposits from various carbon PFC; (b) fuel desorption by long-term annealing at 623 K, i.e. the maximum baking temperature of the ITER divertor; (c) desorption at 1223 K; (d) morphology of surfaces after different cleaning treatments. The study was performed with materials retrieved from TEXTOR and Tore Supra tokamaks (the typical deposits are shown in Fig. 2.3.1.1) and non-exposed plates of graphite and carbon fibre composites, used as reference targets in laser irradiation studies. To determine the distribution of ablated products a variety of dedicated dust catchers was used. The essential results obtained by of ion beam analysis, surface profilometry programmed thermal desorption and various microscopy methods are summarized by the following points:

(a) Qualitative analysis of the dust produced by laser induced ablation has shown the generation of fine (50 nm – 1µm) crystallites. The damage threshold values for ablative removal of several carbon materials have been determined.

(b) Quantitative analysis based on the profilometry of the laser-produced crater and on analysis of debris on the collectors allowed for determination of particle balance. Only 10% of the fuel retained in PFC is removed by baking at 623 K for 72 hours. Efficient removal requires temperatures exceeding 1070 K. Thermal desorption spectra are shown in Fig. 2.3.1.2.
(c) Annealing in vacuum at 1073-1273 K disintegrates thick layers, makes them brittle and the adherence to the substrate is reduced. Also photonic cleaning by laser pulses produces debris, especially under ablation conditions. Laser-produced craters and collected materials are presented in Fig. 2.3.1.3.

(d) Measurements of fuel retention in dust generated by laser-induced ablation show that co-deposits are disintegrated but fuel is not fully removed.

The impact of the results obtained on the development of wall cleaning methods will be discussed and possibilities of applying alternative mechanical methods will be discussed.

**Figure 2.3.1.1:** Morphology of deposits from the toroidal limiters in TEXTOR (a) and Tore Supra (b) tokamaks.

**Figure 2.3.1.2:** Temporal characteristic of fuel release during the long-term annealing of the ALT-II deposits from TEXTOR. (a) Start-up phase (623 K). (b) Final stage (1273 K).
Results of laser ablation studies. (a) Crater on polished graphite after 80 laser pulses at 0.7 J; (b) electron diffraction pattern of the dust, originating from ablation of pure graphite; (c) deposition pattern on a foil after irradiation of graphite; (d) carbon content along the tube collector normalised to one laser pulse.

Nitrogen and Neon Retention in Plasma-Facing Materials

Impurity seeding is a method for improved radiation especially in the case of operation with high-Z plasma-facing components (PFC). Either neon (Ne) or argon (Ar), or nitrogen (N₂) injection is performed. Massive puff of those gases is a key element in the development of disruption mitigation techniques. Nitrogen-assisted discharges are also considered as a route to reduction of fuel inventory. Among many issues related to the injection is the in-vessel residence of gas by implantation, co-deposition or by the compound formation with the PFC material. The latter is especially important in the case of nitrogen-tungsten combination. To address the change of PFC surface morphology in the presence of nitrogen or neon a series of dedicated experiments were performed in the TEXTOR tokamak. These were exposures of: (a) bulk tungsten (W) plates acting as test limiters during discharges with nitrogen injection and (b) graphite probes in the presence of neon injection. A series of injection and the spectroscopy measurements of the increased level of nitrogen in the machine is shown in Fig. 2.3.1.4 a,b. The exposures were followed by comprehensive surface studies carried out with time-of-flight heavy ion elastic recoil detection analysis (TOF-HIERDA) using a 40 MeV ¹²⁹I⁹⁺ beam, Rutherford backscattering spectroscopy, other ion beam techniques, X-ray photoelectron spectroscopy (XPS) and various microscopy methods. There are a few major results.

(a) With HIERDA one detects from 1.3x10¹⁵ to 3.4x10¹⁵ N at cm⁻² in the surface layer (20 nm) of the W plate. Besides tungsten and nitrogen, the other elements are: carbon, oxygen boron and deuterium. Nitrogen is measured even in hot areas free from deuterium. The HIERDA spectrum in Fig. 2.3.1.4c shows that nitrogen from the gas puff is co-implanted into the W surface.

(b) XPS data show that nitrogen occurs in the surface layer (10-20 nm) both in the elemental and compound form, i.e. tungsten nitride (WN/W₂N). From most, but not from all, areas on the plate nitrogen is removed from the surface by Ar⁺ sputter cleaning. This indicates
the presence of implanted or chemically bound N in the layer exceeding 10-20 nm in depth. The presence of tungsten tri-oxide (WO$_3$) and carbide (WC) is clearly proven by XPS, thus confirming previous studies of material mixing on PFC from TEXTOR. (c) Comparative HIERDA and XPS measurements have been performed on a not exposed reference W plate: (i) no nitrogen above the detection limit was found with HIERDA; (ii) nitrogen is detected with XPS on a reference plate but this adsorbed gas is fully removed by sputtering and, secondly, its amount is 2-3 times smaller than on the exposed plate. (d) Neon (2-4x10$^{15}$ cm$^{-2}$) is found in graphite probes exposed in the scrape-off layer during NBI-heated discharges with Ne injection. It shows that noble gases are not instantly desorbed from co-deposits.

Figure 2.3.1.4: Spectroscopy signals recorded during the nitrogen puffing experiment: (a) W(I) line at 400.9 nm and (b) N(II) line at 500.1 nm. (c) HIERDA spectrum recorded on the tungsten plate in an area without a thick co-deposit.

Fuel Retention in Carbon Fibre Composites from Tore Supra
Detailed examination of fuel retention in PFC is crucial for the assessment of tritium inventory in a D-T reactor-class machine. It is particularly important in the case of carbon PFC as long as the use of carbon fibre composites (CFC) is considered for the strike point region of the ITER divertor. Co-deposition is the main pathway for retention but it is also known that fuel migrates to the bulk of CFC. This work has been carried out in cooperation with the CEA Team within the DITS project (Deuterium Inventory in Tore Supra). The main emphasis has been on detailed account on the local fuel retention on the tiles retrieved from the machine. To obtain a comprehensive picture of retention, the D distribution has been determined on plasma-facing surfaces and in the bulk of four pump limiter tiles, which were representative for the erosion and deposition zones on that limiter. The study was carried out by means of nuclear reaction analysis (NRA) using a $^3$He$^+$ beam and detecting protons from the reaction D($^3$He,p)$^4$He. Three series of measurements were performed: (i) plasma-facing surfaces were analysed with a broad beam (1 mm diameter) to obtain an information about
variations in the D content and D/C ratio over the tiles and then; (ii) local variations on those surfaces were probed over an area of less then 1 mm² with a nuclear micro-beam of 10-20 µm resolution; (iii) D distribution in the bulk on cross-sections of cleaved samples was probed with the micro-beam. Prior to the cleaving, the tiles were tight-coated to avoid contamination of freshly open surfaces by D from the plasma-facing surface. The essential results are:

(a) On plasma-facing surfaces from the deposition zone, the D content reaches $2.5 \times 10^{19}$ cm$^{-2}$ in the about 8 µm thick top layer. The areal distribution of is highly inhomogeneous; the differences reach even one order of magnitude between points located 10 mm apart. This is related to the nature of the flaking layer. Results for tiles from different locations on the limiter are shown in Fig. 2.3.1.5 a-d.

(b) The erosion zone contains from $6.6 \times 10^{17}$ cm$^{-2}$ to $7.7 \times 10^{18}$ cm$^{-2}$ D atoms. These differences may be related to the local non-uniform structure of fibres.

(c) Significant variations of the D distribution are noted also on a micro-scale.

(d) Deuterium is detected in the bulk of all tiles under examination, i.e. both from the erosion and deposition zones. There are two common features: (i) fuel is detected up to the depth of 1 mm beneath the plasma-facing surface; (ii) it occurs in bands less then 100µm thick and several mm long which are located roughly parallel to the original plasma facing surface. The D atomic content in the carbon matrix varies largely, e.g. from 0.01 % to 2% in different layers.

![Figure 2.3.1.5: Inhomogeneous distribution of deuterium in various regions on the Toroidal Pump Limiter of Tore Supra measured with micro-beam nuclear reaction analysis: (a)area with thin deposit; (b) shadowed region; (c) thick deposits; (d) erosion zone.](image-url)
2.3.2 Dust collection using collector probes

H. Bergsåker, S. Ratynskaya

Studies of mobile dust with aerogel collectors

The possible accumulation of dust, particularly at hot surfaces, is a critical issue for ITER and other future big devices. The presence of dust in the plasma may also interfere with plasma confinement and other operations aspects. The most common diagnostics for dust in fusion plasmas include tracking of glowing particles with fast cameras, laser scattering and simply collecting and analysing the dust that has accumulated at different places in a device after extended periods of operation.

A new diagnostic technique is to collect dust in a controlled way with silica aerogel collectors, preferably with time resolution. This technique allows counting of particles that are moving in the edge plasma, even if it is not glowing, and analysis of individual particles to determine their size distribution, composition and other properties. The silica aerogel collectors also make it possible to estimate the particle velocities, since fast particles penetrate into the ultra low density material without being significantly damaged, and the penetration depth is related to the particle velocity.

The experimental campaign in 2010 has focused on dust collection in the EXTRAP-T2R reversed field pinch. Metal dust particles has been collected in discharges with and without feedback control of the resistive modes and rotating tearing modes. The discharges without the feedback terminate abruptly with large impurity production towards the end. The dust particle fluxes were collected in the toroidal and radial directions. In addition the difference between upstream and downstream with respect to the edge ExB drift could be resolved (Fig.2.3.2.1). The exposed samples were analyzed by SEM and optical microscopy. In the latter, particular attention was paid to improvement of the accuracy of the dust detections/counting by well-controlled surface preparation and image processing the surface before and after exposure, see figure 2.3.2.2.

Figure 2.3.2.1. Examples of entrance holes at aerogel surfaces exposed in T2R. Left panel: a single hole at the surface exposed in the same discharges, but on the downstream side. The material at the bottom of the hole contains stainless steel. Right panel: two holes at a surface exposed to 13 discharges with feedback, upstream side.
The results of EXTRAP exposures were compared with measurements with the same method in the TEXTOR tokamak. The capture rate of slow particles larger than 10 micron was comparable in the two devices, in spite of large differences in edge profiles, geometry and plasma facing materials. In T2R, it was larger for discharges without active feedback mode suppression and hard termination, in agreement with the metal impurity flux. The inferred flux of small fast particles was larger in T2R than in TEXTOR and appeared to be mainly in the ExB drift direction. The inferred flux of larger fast particles was larger in TEXTOR than in T2R.

**Publications section 2.3**


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2.4 Physics of fast particles, heating and current drive

*T. Hellsten, T. Johnson, M. Laxåback, A. Hannan (PhD student), J. Höök (PhD Student), Q. Mukhtar (PhD student)*

The research is focused on studying wave-particle interactions relevant for fusion experiments, in particular for heating, current drive and excitation of waves by fast particles. The group develops codes for predicting the effects of ICRH, and validates them against experiments. The program is well integrated into the European fusion program through participation in: the Integrated Tokamak Modeling Task Force, the exploitation of the JET facility and the EU training programme GOTiT.

The main codes developed by the group are PION, FIDO, SELFO and SELFO-light. PION was the first self-consistent code for modeling ICRH and NBI heating using a model for the power deposition and solves a simplified Fokker-Planck equation for the distribution function (developed by L.-G. Eriksson and T. Hellsten). PION has become the standard code for routine simulation at JET. The Monte Carlo code FIDO calculates the distribution functions of the resonant ion species taking into account effects caused by finite orbit width and RF-induced spatial transport due to absorption of the momentum of the wave. The SELFO code calculates the wave field, using the LION code, and the distribution function, using the FIDO code, self-consistency is obtained by means of iterations. The FIDO code is being upgraded to include interaction with MHD waves allowing self-consistent studies MHD modes during ICRH; at the moment by using simple models of the MHD-modes. A new code, SELFO-light, for routine simulation to be used at the Gateway and for analysing ITER and DEMO experiments has been developed based on a simplified Fokker-Planck equation for the distribution function similar to that used in PION.

**Exploitation of JET**

Martin Laxåback have been long term seconded at JET planning, coordination and monitoring of the JET programme.

**EU training programme GOTiT**

The group was one of the applicants in the EU training programme GOTiT (Goal Oriented Training in Theory) covering 16 trainees at 6 institutes started in autumn 2008, where four of our students are involved and Torbjörn Hellsten participates in the training. The aim of the programme is to train modellers to the most recent mathematical and numerical methods and best practice in the use of high performance computers as well as to the state-of-the-art theoretical models developed and applied by the fusion community. Characteristics for the programme are monthly teleconference seminars, intense High Level Courses given at the members laboratories with participants from the whole programme.

**ITM task force**

The group participates in the Integrated Tokamak Modelling Task Force, where Thomas Johnson is Deputy Project Leader for IMP5; the integration project for heating, current drive and fast particle effects. The main contributions have been in the development of the ITM infrastructure, the adaptation of codes to the ITM infrastructure and the development of advanced Fokker-Planck models.
The ITM infrastructure related to the IMP5 has been significantly developed during 2010 (with most of the contributions coming from members of the group); all data structures for representing heating and current physics has been rewritten, modules for manipulating the data structures (generating and merging) has been constructed and workflows for integrated modelling of heating and current drive has been developed.

Work on integrating the ICRF code SELFO-light into the ITM infrastructure has been initiated. In addition, several new developments of the SELFO-light have made the code a very competitive code within both the ITM and the ICRF community.

**FIGURE:** Various representation of the 5D distribution function obtained from RFOF-ASCOT simulations. The graphs show the distribution function: (a) as function of velocity v; i.e. integrated over space and pitch angle, (b) and (c) fast ion density in [R,Z], (d) velocity distribution function (space integrated), as function of \( v_\parallel \) and \( v_\perp \). (e) and (f) show high and low field side distribution function of \( v_\parallel \) and \( v_\perp \).

**Development of codes for Modelling ICRH**

A fast code for routine simulations of ICRH has been developed, SELFO-light. The code is based on a similar formulation used for the PION code using a 1D –Fokker-Planck solver with a model for the parallel velocity. The upgrading of the code consists of replacing the formula for the power deposition with direct solution of the wave equation. This enables calculation of the power deposition with several cyclotron resonances or harmonics of it at different locations and to calculate electron current driven by the magnetosonic waves, which in the past have to be done with more advanced codes.

**Development of algorithms for simulating plasmas with Monte Carlo codes**

In order to speed up the calculation of distribution functions with Monte Carlo methods an adaptive \( \delta f \)-method has been developed and tested for one dimension [1]. The drawback with conventional \( \delta f \)-methods is either that they only allow a small local deviation of the distribution function or that new particles have to be continuously added. These problems are
solved by resampling the distribution function and improving the approximation of the distribution function in when it is resampled in order to minimize the source term and the number of simulated particles. Resampling of the distribution function in higher dimensions by projecting the distribution function on a FEM base is less attractive. Methods by combining particles have been tried out.

When simulating plasmas with Monte Carlo methods poor convergence is achieved at because of boundaries where the diffusion coefficients vanish. To speed up the calculations and/or to improve the convergence improved Monte Carlo schemes have been developed. An adoptive method that chooses between the different Monte Carlo schemes has been developed and tested for 1D problems reducing the computing time for the same accuracy with a factor of 10 [2]. Monte Carlo operators have been developed enabling simulation for which the density profile relaxes to a prescribed one e.g. given by transport codes or experimental data.

**RF-induced rotation**

Rotation in plasmas can have beneficial effects. For instance, it can enhance the stabilizing effect of a resistive wall. Shear in the rotation is also believed to be an important factor for transport barriers. In ITER and future reactors the ratio between the neutral beam injected momentum and energy will be low, because of the high energy required for central heating as a consequence the induced rotation by the beam is not expected to give rise to strong plasma rotation. It is therefore of interest to consider other mechanisms with a potential to produce rotation. Intriguing observations of rotation in plasmas heated by Radio Frequency (RF) waves with little or no external momentum input have been made in several machines. There is as yet no complete understanding of the mechanisms behind the observed rotation. Effects due to fast ions, MHD and transport have been proposed, but reliable theoretical predictions of rotation in ITER or a reactor with low momentum are not yet available.

Analysis of JET experiments to study the effect of ICRH on the toroidal rotation of the plasma has been carried out. In the outer part of the plasma a co-current rotation appears. This is in contrast to the central plasma rotation, which was affected by the frequency and wave spectrum. When mainly heating the electrons directly with different heating scenarios without creating high energy ion orbits hollow rotation profiles appear with counter rotation in the core. This effect seems to be more correlated with the electron heating rather than heating mechanism.

**Momentum transport by non-resonant wave-particle interactions**

The role of non-resonant wave-particle interactions in spatial dispersive nonuniform media on the momentum transport has been studied. These interactions describe the change in the wave momentum by the poloidal upshift, which is important for modeling RF-heating. The non-resonant interactions give rise to a torque on the plasma, which has been proposed as an explanation of the strong rotation induced in the Alcator C-mod experiment during mode conversion heating for which the upshift can become strong due to the short wave length of the mode converted wave. The torque, which is as strong as the quasi-linear torques when modeling heating, is obtained as the difference between the torques from the ponderomotive force and from the Reynolds stress arising from the fluctuations caused by the wave field. The torque equals the torque arising in the geometric optics from refraction as the wave momentum changes in a spatial inhomogenous medium [3].
**Intrinsic plasma rotation with toroidal field ripple**

The effect of toroidal field ripple on the plasma rotation has been studied [4] using the unique capability of JET to change the amplitude of the magnetic field ripple, without modifying other relevant equilibrium conditions. The effect of the ripple on the angular rotation frequency of the plasma column was investigated under the conditions of no external momentum input. The ripple amplitude was varied from 0.08% to 1.5% in Ohmic and ion-cyclotron radio-frequency (ICRF) heated plasmas. In both cases the ripple causes counter rotation (see figure 1), indicating a strong torque due to non-ambipolar transport of thermal ions and in the case of ICRF also fast ions. The effect was seen both in the edge and core. JET results suggest that ripple will affect rotation in ITER, and should be taken into account in extrapolation from present data. Furthermore, the effect of changing the position of the ICRF resonance was studied. Figure 2 shows the rotation profiles for pulses with 1.5% ripple. The largest edge and core counter rotation was observed when the ICRF resonance position was on the low-field side, i.e., where the ripple amplitude is stronger. This correlation between the ripple experienced by the fast ions and the rotation indicate that the fast ions receive a torque from the toroidal field coils.

![Figure 1](image1.png)

*FIG. 1* (a) Toroidal rotation profiles for ICRF heated H-mode plasmas with $I_p=1.5$MA and $<B_T>=2.2$T. Top: pulse #74688 with ripple amplitude $\delta=0.08\%$ and $P_{ICRF}=3.1$MW; bottom: pulse #74686 with $\delta =1.5\%$ and $P_{ICRF}=2.9$MW. The plasma center is at $R_0=3$m and the resonance at $R_{res}=2.71$m. (b) Toroidal rotation as a function of ripple for ICRF heated H mode pulses

![Figure 2](image2.png)

*FIG. 2.* Toroidal rotation profiles for L-mode pulses with $\delta =1.5\%$, 1.5 MA, and $P_{ICRF}=2$ MW, for different resonance positions: (1) #77010 with $R_{res}=2.38$ m, (2) #77014 with $R_{res}=2.71$ m, (3) #77009 with $R_{res}=3.13$ m.
Experimental study of ICRF scenarios for the non-activated phase of ITER
JET experiments have been performed to investigate two ICRF heating schemes proposed for the half-field operation phase of ITER in Hydrogen plasmas; fundamental H majority and 2nd harmonic 3He ICRF heating [5]. Although the same magnetic field and wave frequencies (B0 = 2.65T, f = 42MHz and f = 52MHz, respectively) were used, the density and particularly the plasma temperature were lower than those expected in the initial phase of ITER. The heating efficiencies obtained for the fundamental H majority heating scheme were around \( \eta = 0.3-0.4 \) with dominant fast wave electron damping and hints of enhanced efficiency with increasing plasma temperature were observed. For the 2nd harmonic 3He heating scheme, the efficiency varied from below \( \eta = 0.2 \) (for low X[3He]) up to \( \eta = 0.4 \) when X[3He] \( \geq 20\% \) was reached. It was shown that the increase in the heating efficiency with X[3He] is mainly due to enhanced 3He ion-cyclotron absorption, which exceeds the electron absorption at high 3He concentrations. Fast H ions up to 50keV and fast 3He ions up to 200keV were detected by the NPA diagnostics in the N=1 H and in the N = 2 3He heating experiments, respectively, when 5MW of RF power was applied. The low ICRF heating efficiencies associated to the rather poor confinement of the H plasmas in the experimental conditions described here also lead to relatively strong plasma wall interaction in both scenarios.

Fast ion losses during fishbone oscillations in JET
Experimental measurements of fast ion losses during fishbone oscillations have shown that non-resonant ions (ions not resonant with the fishbone) can be expelled from the plasma. To investigate this extensive modelling were performed using a combination of the SELFO code for calculating the fast ion population before the onset of the fishbone, and the HAGIS code for calculating the effect of the fishbone. The modelling was able to predict, with fair quantitative agreement, velocities and pitch angles of the lost ions [6].

Ion heat transport studies and relation to plasma rotation in JET
Detailed experimental studies of ion heat transport have been carried out in JET, exploiting the upgrade of Active Charge Exchange Spectroscopy and the availability of multi-frequency ICRH with He3 minority [7,8]. This work was lead by Paula Mantica (NEA) and the group contributed primarily with modelling of the ICRF heating. The work has shown that the ion stiffness decreases strongly in presence of toroidal rotation when the magnetic shear is sufficiently low. This effect is dominant with respect to the well-known \( \Omega_{E	imes B} \) threshold up-shift and plays a major role in enhancing core confinement in Hybrid regimes and Ion Internal Transport Barriers. The effects of \( T_i/T_e \) and \( s/q \) on ion threshold are found rather weak in the domain explored.

Ion cyclotron current drive for monster sawtooth control in JET
Experimental evidence from the JET tokamak was presented in [15,16] supporting the predictions of a recent theory on sawtooth instability control by toroidally propagating ion cyclotron resonance waves. This work verifies that the kinetic response of highly energetic ions on the internal kink mode is sufficient to explain highly effective sawtooth control techniques by toroidally propagating ICRF waves with resonance tangential to the q = 1 surface. This has been achieved by creating experiments capable of eliminating all other known control mechanisms. Furthermore, more advanced experimental verification was undertaken by variation of the amplitude of the analytically derived fast ion mechanism. That fast ions can so dramatically, and directly, affect sawteeth is encouraging for ITER, especially where control solely via the magnetic shear is expected to be more challenging. The work was supported by numerical simulations showing that the fast ion modification of \( \delta W \) is destabilizing and consistent with the experimentally observed changes in the sawtooth period.
**Sawtooth control for ITPA task on in ITER and JET**

The group has been closely involved in the experimental verification of a recently proposed model for sawtooth stabilisation and destabilisation by fast ions [9,10]. For this purpose a He3 minority heating scenario was adopted, to minimize the effects of ion cyclotron current drive, which if present could provide an alternative explanation for the sawtooth behaviour. As part of the validation, detailed ICRF modelling were performed by the group and then analysed with the Hagis code to establish the effect on the sawtooth stability, which was then compared with experimental results.

The group also contributed to the ITPA MHD Working Group 3 project “Power requirements for ICRH and ECCD for sawtooth control in ITER”, primarily with modelling of ICRF generated fast ions and their influence on the sawtooth stability. Here the primary objective of sawtooth control is for avoiding the triggering of neoclassical tearing modes. Thus, detailed modelling of the fast ion populations from both ICRF and NBI was performed. The response of the fast ions to kink perturbations was studied then with the Hagis code, to identify their impact on the sawtooth stability. To validate the results the SELFO results were benchmarked with the ICRF codes AORSA/CQL3D and SCENIC.

**Benchmark between the SELFO and the SCENIC codes for ICRH**

In collaboration with CRPP a benchmark between the ICRF codes SELFO and SCENIC was initiated in 2009 and completed in 2010 [11]. While the activity for 2009 focussed on the input equilibrium and wave field solvers, the 2010 activity has been focussed on the fast ion physics. The benchmark was successful in the sense that, although errors were found in both codes, the end result showed good agreement for the benchmark case.

**Publications section 2.4**


2.5 Energetic particle physics

2.5.1 Physics of burning fusion plasmas

*M. Lisak, M. K. Lilley, R. Nyqvist (PhD student)*

**Introduction**

The research programme is concentrated on the study of stability and transport in magnetically confined, impure plasmas with a special emphasis given to transport driven by microinstabilities and fast particle effects in fusion plasmas. The aim is to develop improved theoretical models within this area, to be able to explain experimentally observed phenomena in existing magnetic confinement devices and to make predictions on the implications for the design of future ones.

One of the main objectives of tokamak devices such as JET and the planned ITER project is the study of alpha particle production, confinement and consequent alpha particle heating of D-T plasmas. Up to now the fusion experiments were in a sub-critical zone with the product of the plasma density $n$, plasma temperature $T$ and the energy confinement time $\tau_e$ being of the order of $nT\tau_e < 8.3 \times 10^{20} \, \text{m}^{-3} \, \text{keVs}$ and without burn or with a small burn, e.g. with $Q=0.61$ achieved at JET. In a burning DT plasma with $Q>10$, the energetic alpha particle population will contribute a considerable part of the total plasma pressure (typically 10-20%) and the alphas can be expected to give rise to fundamentally new physics phenomena which may have a substantial impact on achieving and maintaining high temperatures, thus modifying the beta limits and energy confinement times of tokamak reactors. Furthermore, the alpha particle population itself may have a considerable impact on plasma stability, possibly affecting the alpha thermalisation as well as the transport of high-energy alphas and thermal ions. From a practical point of view, both the confinement capability, wall loading and fusion power density of a tokamak may be affected by the presence of the alpha particles. For a number of years the Chalmers group has carried out theoretical research concerning the physics of burning fusion plasmas. The research has mainly been devoted to the following questions:

- What fraction of the energy associated with the alpha particles will be transferred to the plasma and what are the direct losses of the alphas to the wall?
- Which collective effects and associated macroscopic processes will result from the alpha particles and what is the impact of alpha particles on plasma stability and transport?
- Is it possible to draw conclusions about fast ion behaviour already from existing experimental data?
- Is (quasi-) steady state thermonuclear burn possible and which methods are required for burn control?

Both the design features and the operation regime of a fusion reactor depend on the answers to these questions. The confinement properties of energetic alpha particles are of fundamental importance for alpha heating, burn control and alpha particle diagnostics. The next generation of tokamaks will produce a large amount of thermonuclear alpha particles, which may excite wave instabilities. The basic reason for these instabilities is the deviation of the alpha distribution function from thermodynamic equilibrium, and the instability drive may thus be spatial as well as velocity space gradients, trapped as well as passing particles. The presence of thermonuclear instabilities may in turn cause anomalous losses of plasma energy and high-
energy particles. Many aspects of fast ion collective effects are well understood, including excitation of MHD modes such as fishbone oscillations and Alfvén gap modes in toroidal devices. However, a "hot" topic is the role of kinetic and nonlinear Alfvén instabilities. Promising topics for future advances include explanations for rapid frequency chirping, unambiguous identification of energetic particle modes and a quantitative understanding of the radial structure of Alfvén modes. Generally, to achieve an improved understanding of nonlinear dynamics of the thermonuclear wave instabilities driven by energetic ions and to determine the wave saturation mechanism and its implications fast ion confinement, requires a further development of the theory of the nonlinear evolution of kinetic systems by H. Berk and B. Breizman.

**VR Association Work Programme**

**Alfvénic Eigenmodes within the q = 1 Radius in Sawtoothing Plasmas**

R. Nyqvist, B. N. Breizman, M. Lisak, S. E. Sharapov

In JET discharges with high power ICRH and a central safety below unity, a hot ion population inside the q = 1 radius can stabilize the sawtooth crash during periods of up to 1 s or longer. During these periods, the central electron temperature builds up to significantly larger values than during ordinary sawtooth oscillations, finally saturating or even decreasing. The sawtooth free periods end with giant sawtooth crashes, preceded by magnetic activity in the toroidal Alfvén eigenmode (TAE) and sub-TAE frequency range (Figure 1), exhibiting a characteristic frequency evolution where cascading modes are transformed into “tornado modes” [P. Sandquist, S. E. Sharapov, M. Lisak and T. Johnson, Physics of Plasmas 14, 122506 (2007)]. In this contribution, ideal and non-ideal Alfvén eigenmodes (AEs) within the q = 1 surface are investigated analytically and numerically with a suite of MHD codes. It is found that in typical JET equilibria, the cascading modes can be identified as core-localized Alfvén cascades (AEs supported by reversed or extremely low shear) achieving their maximum amplitude close to, but below, the frequency range of a collection of low shear toroidal AEs, exhibiting “tornado” behavior due to the proximity of the magnetic axis (Fig. 2).
Radiative Damping of Low Shear Toroidal Alfvén Eigenmodes
R. Nyqvist, S. E. Sharapov, M. Lisak

Radiative damping due to finite ion Larmor radius effects was considered analytically for core-localized low shear toroidal Alfvén eigenmodes (TAEs) in the center of a large aspect ratio tokamak. The introduction of the short ion Larmor radius length scale admits for wave tunneling into extremely short perpendicular wavelengths. The tunnelled wave amplitude is propagated radially away from the low-shear region and considered lost. It was found, using a ballooning space formulation corresponding to a radial Fourier transformation, that radiative damping of the upshifted TAE mode in the toroidally induced Alfvén gap was negligible, while the downshifted mode could experience some damping in the very low shear limit (due to a somewhat more narrow potential barrier). These results are currently being reproduced using a real space representation.

Effect of dynamical friction on nonlinear energetic particle modes
M. K. Lilley, B. N. Breizman, S. E. Sharapov

The present day experiments on Alfvén eigenmodes AEs reveal a rich family of nonlinear scenarios with several types of evolution of the wave amplitudes and frequencies. AEs excited by ion cyclotron resonance heating ICRH-produced ions show usually the “soft” excitation regimes. In these nonlinear scenarios, the mode frequency remains close to the linear AE eigenfrequency and the structure of the mode remains similar to the linear one. In contrast, in the case of neutral beam injection NBI, a “hard” nonlinear regime is observed more often, resulting in bursting amplitude evolution and in rapid frequency sweeping. Spontaneous formation of phase space holes and clumps is typical of the NBI-driven scenarios. The holes and clumps in the energetic particle distribution function correspond to resonant particles that are trapped in the field of the wave. These nonlinear structures can be viewed as long living Bernstein–Greene–Kruskal BGK modes. The disparity between experiments on AE excitation by ICRH and NBI has recently been attributed to the role of dynamical friction drag as a relaxation process for resonant particles. This observation was made in the bump-on-tail model for a near threshold instability. In the past, such a model was successfully used to explain nonlinear bifurcations and frequency sweeping events observed in various AE experiments. It was found that the mode grows explosively when dynamical friction drag dominates over velocity space diffusion in the vicinity of the wave-particle resonance.

\[
\alpha/(\gamma_L-\gamma_d)=1.5
\]

![Fig. 3. Establishment of steady state holes by the removal of the drag sink at t \times \gamma_L =800.](image)
This previous analysis was limited to a weakly nonlinear regime with a perturbative treatment of the energetic particle response to the wave field. The present work extends the analysis to describe the full nonlinear behavior in drag dominated scenarios. This is largely based on numerical modeling, which generalizes an earlier code to include the effect of drag. We observe that the drag continues to play a destabilizing role in the fully nonlinear problem. Specifically, in the early nonlinear phase of the instability, the drag facilitates the explosive scenario of the wave evolution, leading to the creation of phase space holes and clumps that move away from the original eigenfrequency. Later in time, the electric field associated with a hole is found to be enhanced by the drag, whereas for a clump it is reduced. This leads to an asymmetry of the frequency evolution between holes and clumps. The combined effect of drag and diffusion produces a diverse range of nonlinear behaviors including hooked frequency chirping, undulating, and steady state regimes. An analytical model is presented, which explains the aforementioned diversity. A continuous production of hole-clump pairs in the absence of collisions is also observed.

**Fig 4. Establishment of a steady state hole in the presence of both drag and diffusion.**

**Fig 5. Hooked frequency spectrum with drag and an increase in diffusion from that in Fig 4.**

**Adiabatic Description of Long Range Frequency Sweeping**

R. M. Nyqvist, M.K. Lilley and B. N. Breizman

Magnetically confined plasmas exhibit a wide range of kinetic instabilities. The role of dissipation in such systems is far from trivial. In particular, the near threshold regime exhibits spontaneous formation of nonlinear phase space structures in the driving species distribution function, corresponding to a transformation from unstable cavity eigenmodes to beam-like,

**Fig 6. Spectrogram showing bursting-type TAE modes observed on the spherical tokamak MAST.**

**Fig 7. Spectrogram from bump-on-tail simulation of bursting-type MAST modes.**
self-sustained energetic particle modes with time dependent frequencies. Such sweeping events are commonly observed experimentally, see e.g. Fig. 6, and have traditionally been modelled using a 1D bump-on-tail framework (Fig. 7). Recently, emphasis has been put on asymmetric, long range frequency sweeping (previously theoretically reported only in the presence of drag collisions, which act to slow down the driving species), see Fig. 7. The Chalmers group has developed an efficient tool (the ABBOT code) for analyzing such events. The model takes advantage of the fact that after the formation of the phase space structures, the modes evolve slowly as compared with the mode oscillations. Taking into account the deviation from the unstable mode frequency and particle trapping in the wave field, the model predicts asymmetric frequency sweeping even in the collisionless limit (Fig. 9). Moreover, the model can accurately follow modes over a large span of frequencies, thus permitting quantitative analysis of sweeping rates and amplitude evolution (Fig. 10).

**Fig. 8.** Spectrogram showing long range frequency sweeping on JET, so called hooks.

**Fig. 9.** Spectrogram of hooks on MAST.

**Fig. 10.** Spectrogram showing asymmetric long range frequency sweeping in the collisionless case, as predicted by the ABBOT code.

**Fig. 11.** Spectrogram of hook, ABBOT code.

**Wave-Particle Interaction in Toroidal Axisymmetric Systems**

R. Nyqvist, M. Lisak, D. Anderson, J. Zalesny

The interaction between charged particles and plane electromagnetic waves in a large aspect ratio tokamak is investigated analytically by calculating the induced change in the constants of motion per poloidal transit period due to the presence of the wave. The wave-particle resonances are treated locally using a stationary phase method, and by forming a system of difference equations, a condition for stochastic particle motion is derived in terms of the wave
amplitude. If the motion is stochastic, either due to large wave amplitude or other effects such as e.g. magnetic field ripples, an appropriate Fokker-Planck equation can be derived with the effect of the wave-particle interaction treated as a diffusive process. The associated diffusion coefficient can be calculated in terms of the changes in the constants of motion of the unperturbed particle orbits due to the wave.

**Nonlinear evolution of two fast-particle-driven modes near the linear stability threshold**

J. Zalesny, I. G. Galant, M. Lisak, S. Marczyński, P. Berczyński, A. Gałkowski, S. Berczyński

A system of two coupled integro-differential equations is derived and solved for the nonlinear evolution of two waves excited by the resonant interaction with fast ions just above the linear instability threshold. The effects of a resonant particle source and classical relaxation processes represented by the Krook, diffusion, and dynamical friction collision operators are included in the model, which exhibits different nonlinear evolution regimes, mainly depending on the type of relaxation process that restores the unstable distribution function of fast ions. When the Krook collisions or diffusion dominate, the wave amplitude evolution is characterized by modulation and saturation. However, when the dynamical friction dominates, the wave amplitude is in the explosive regime. In addition, it is found that the finite separation in the phase velocities of the two modes weakens the interaction strength between the modes. In a further work, the two-wave analytical model has been simplified presenting it in a form based on purely two coupled differential equations of fifth order. Here, the effects of the Krook, diffusion and dynamical friction (drag) relaxation processes are considered, whereas shifts in frequency and wavenumber between the modes are neglected. In spite of these simplifications the main features of the dynamics of the two plasma modes are retained. The numerical solutions to the model equations show competition between the two modes for survival, oscillations, chaotic regimes and ‘blow-up’ behavior.

**Publications section 2.5.1**

**Peer reviewed journals**


**Presentations at EPS conferences**

Other workshops and conferences


2.5.2 Runaway electrons in tokamak disruptions

*T. Fülöp, T. Fehér (PhD student), G. Papp (PhD student)*

Due to a sudden cooling of the plasma in tokamak disruptions a beam of relativistic runaway electrons is sometimes generated, which can cause damage on plasma facing components due to highly localized energy deposition. This problem becomes more serious in larger tokamaks with higher plasma currents and understanding of the processes that may limit or eliminate runaway electron generation is very important for future tokamaks, such as ITER. During 2010, our model for runaway dynamics in the presence of impurity injection have been applied to ITER-like plasmas. Furthermore, we have developed a tool to assess the effect of resonant magnetic perturbations on runaway losses.

Runaway electron generation in tokamak disruptions

Fast plasma shutdown by impurity injection has been studied in an ITER-like scenario, with a 1-dimensional model of electric field, temperature and runaway current. The runaway generation model includes Dreicer, hot-tail and avalanche mechanisms. Our study showed that high neon and argon concentration can cause short thermal and current quench times, but usually a high runaway current is produced. With lower concentration of impurities this can be avoided, but the inhomogenities in cooling can result in a strong runaway generation by the avalanche mechanism also in that case.

![Current sheets are created after injection of neon into the plasma (nNe /n0 = 0.5).](image)

(a) Electron temperature as a function of time and normalized radius. (b) plasma current decay including the current driven inside the sheets (dash-dotted) and the runaway current.
**EFDA Task Force and Topical Group highlights**

As part of task WP10-MHD-02-01-xx-01/VR/PS we developed a model for 3D runaway drift orbits in 3D magnetostatic perturbed fields to assess the effect of RMP coils on loss enhancement. The relativistic drift equations for the runaway electrons were solved with the ANTS (plasmA simulatioN with drifT and collisionS) code, modified for our purposes to include the effect of synchrotron radiation losses, handle relativistic particles correctly, and we also included a new collision operator valid for arbitrary energies. In order to benchmark the code, we used a TEXTOR-like equilibrium with typical parameter profiles. The magnetic field perturbations were modelled to be like the ones produced by Dynamic Ergodic Divertor (DED) coils at TEXTOR. For the work presented here, the DED coils were operated in the (6,2) DC operation mode, hence the formation of magnetic islands with n=2 toroidal mode number was the most favoured. On the plasma edge, due to the steep q profile, the overlapping of islands generates a large ergodic zone, thus enhancing radial transport and runaway electron losses (Figure 1a).

![Magnetic field line tracing, poloidal cut](image)

**Figure 1.** (a) Magnetic structure with I_{DED} = 6 kA. Large ergodic zone forms at the edge of the machine, marked with red. (b) Time dependence of particle losses. The DED influences the onset time of the losses.

We found that runaway electrons in the core of the plasma are likely to be well confined. For low-energy (≈ 1 MeV) particles closer to the boundary, the onset time of the losses is dependent on the amplitude of the magnetic perturbation (see Figure 1b). The runaway current damping rate is, however, insensitive to the magnetic perturbation level, and its experimentally measured value is consistent with our simulations. Our results indicate that the loss of high-energy (> 10 MeV) runaways in the simulation is mostly due to the fact that their orbits are wide, which allows them to intersect the wall. The loss is dominated by the shrinkage of the confinement region, which is independent of the DED current.

**Publications section 2.5.1**

**Presentations at international conferences and workshops**


3 Development of plasma auxiliary systems - diagnostics

3.1 Neutron diagnostics

E. Andersson Sundén, M. Cecconello, S. Conroy, G. Ericsson, C. Hellesen, M. Weiszflog, J. Eriksson (PhD student), M. Gatu Johnson (PhD student), Siriyaporn Sangaroon (PhD student), M. Skiba (PhD student)

3.1.1 Instrumental developments for JET and MAST

The TOFOR instrument at JET

TOFOR is a neutron time of flight spectrometer at JET. A collimated beam of neutrons escaping the fusion plasma scatters in a first detector and a fraction of the scattered neutrons interact with a second detector. From the time between these two scattering interactions, the energy of the incoming neutron is deduced. During 2010, detailed work on the effect of the finite sight line on the measured spectra has been carried out.

Investigation of how the finite Larmor radii of energetic fuel ions affect fast ion measurements

In the interpretation of data from fast ion measurements it is normally assumed that the entire Larmor orbits of the ions are visible to the measuring instrument. However, if the Larmor radii of the fast ions are comparable to the width of the viewing cone of the instrument, this approximation, and hence the interpretation of the results, may be invalid. The effect is investigated in an experiment at JET using radiofrequency heating at the 3rd harmonic of the deuterium cyclotron frequency, in combination with 100 keV deuterium neutral beams. The fast ions produced by this heating scheme were measured by analyzing the neutrons from the d(d,n)^3He reaction with TOFOR. The experiment was setup so that the resonance layer of the radiofrequency heating was placed close to the outer limit of the field of view of TOFOR. This creates a situation where only a part of the Larmor motion of the fast deuterons is visible to TOFOR, which affects the measured neutron energy spectrum.

In order to investigate the importance of the finite Larmor radii (FLR) effects, a model of the fast ion distribution, including the Larmor motion, has been developed. From this model it is possible to calculate neutron energy spectra, with and without taking the FLR effects into account, and compare the results with TOFOR data. One example is shown in Figure 1. It is seen that the FLR effects need to be included in order to get a good agreement with the data. If these effects are not included, the low energy (long time-of-flight) part of the neutron spectrum is overestimated up to a factor of 10. This behavior is expected since the large Larmor radii of the fast deuterons cause these low energy neutrons to be emitted slightly outside the field of view. A systematic investigation of these effects, using data from all the successful discharges from the JET experiment under consideration, has been performed and the results are similar to the example in Figure 3.1.
The result shows that knowledge of the entire Larmor motion can be important for the interpretation and understanding of fast ion measurements for certain plasma scenarios. This will be important for future fast ion measurements with TOFOR, and applies to other fast ion diagnostics, such as γ-ray spectroscopy and neutral particle analysis, as well.

Assessment of the effect of upgraded data acquisition hardware on the TOFOR performance

TOFOR records neutron interaction times in two sets of plastic scintillators: S1 and S2. By calculating the time elapsed between the associated S1 and S2 events, a time-of-flight spectrum can be constructed. Some of the recorded events will, however, not be true coincidences arising from the same neutron, but random coincidences. These events give rise to a flat background in the time-of-flight spectrum. Since TOFOR presently only records time information, it is not possible to discriminate against this background. If more modern data acquisition boards capable of recording both time information and energy deposition data were to be installed, some of the accidentals could be removed based on the relation between the TOFOR geometry, neutron time-of-flight and energy deposition in S1. For a given energy of the incident neutron, the scattering angle in the S1 detector depends on the energy it deposited in that detector. The S2 detectors cover only a finite scattering angle interval. These facts can be used to construct energy discrimination windows, as functions of the scattering angle, for each measured flight time. These window functions can then be used to discriminate against random coincidences, where the deposited energy of the neutrons in S1 might be incompatible with a neutron flight-time. The main purpose of this work was to construct such windows, in order to provide cleaner spectra and enable visibility of previously hidden spectral features. This was done with simulated TOFOR data, producing spectra with significant decreases in background level, see Figure 3.2.

In addition to reducing the background in the time-of-flight spectra, a new data acquisition system storing pulse shapes instead of only time information might improve the TOFOR time resolution. The resolution could be affected by an energy-dependence in the shape of the neutron-induced scintillator pulses. Stored pulse shape data can be used to develop a better timing technique. As an introductory study, a parametrization of the energy-dependent
scintillator pulses has been done and compared to experimental data. The results provide a better understanding of the dependence between energy and pulse shape and enable a further analysis of timing-issues in TOFOR.

![Figure 3.2](image)

Figure 3.2. TOF spectrum constructed from Monte Carlo simulated TOFOR neutron data. The green line shows an unmodified, simulated TOF spectrum while the black line shows the effect of geometry-based background discrimination as described above.

The MPRu instrument at JET - Upgrade of the MPRu radiation shielding and electronics

During the shutdown of JET the upgraded Magnetic Proton Recoil spectrometer (MPRu) was upgraded. The detector array of the MPRu consists of 32 phoswich scintillators. Each scintillator is read out by two photo multiplier tubes (PMTs). During the JET campaigns since the major upgrade of the MPRu (2005), a five of these PMTs have broken down. These PMTs were replaced with new ones during the summer of 2010. In addition, the radiation shield of the MPRu was upgraded. The background of the MPRu mainly consists of gamma radiation, which originates from thermal neutron capture in the vicinity of the hodoscope. Gadolinium paint has been applied to the lead shield of the hodoscope, to absorb these thermal neutrons before reaching the hodoscope area. Both upgrades were performed successfully.
The neutron camera at MAST

The collaboration with the Culham Centre for Fusion Energy (CCFE) during the years 2008 – 2009 has resulted in the preliminary installation during 2010 of the proof-of-principle neutron camera with spectroscopy capabilities on the MAST experiment as shown in figure 3.3. This preliminary installation included only two of the four detectors and only the equatorial lines of sight were populated. The neutron camera collected its first data during the 2010 MAST experimental campaign confirming the expected performances. The first results were reported at the 18th Topical Conference on High-Temperature Plasma Diagnostics, Wildwood, New Jersey, May 2010 and later published in the journal Review of Scientific Instruments.

Figure 3.3. The neutron camera in its final installation at MAST.

The neutron diagnostic was commissioned in early 2010 for measurements at fixed location, i.e. without radial scan possibilities, with the detectors and acquisition system fully operational. As expected no signal was detected in ohmic discharges and first data were acquired with NBI heated plasma during parasitic experiments. The neutron collimation and shielding capability of the thick polyethylene blocks were verified taking data with open and closed collimators as shown in figure 3.4. The results clearly show how the shielding is able to suppress the neutron contribution to the detector signal as well as showing the gamma/neutron discrimination capabilities of the detectors. The neutron-capture gamma background is quite large and a better lead shielding around the detector would be beneficial.

Figure 3.4. Signals recorded on a detector with the collimators closed (left) and open (right): the neutrons’ footprint on the detector signal is clearly seen. The signals with closed collimator are from neutron-capture gammas and from the Na22 calibration source.
3.1.2 Physics achievements - Experiments at JET

Measurements of fast ion and their interactions with plasma instabilities

Diagnosis of fast ions in fusion plasma at JET has been a focus of attention for the Uppsala neutron diagnostic group for many years, and during 2010 we published a series of results on fast ion diagnostics using neutron spectrometry. One of the papers, a letter published in the IAEA journal Nuclear Fusion, was one of the most downloaded during 2010 and featured in the journal’s list of Highlights of 2010 edition.

Knowing the behavior of fast ions in a fusion plasma is of great importance, since in an ignited (self-sustained) plasma the heating is produced by the slowing down of the 3.5 MeV alpha particles born in the DT fusion reaction. Energetic ions in the MeV energy range are known to provide a drive for plasma instabilities, and these can in turn redistribute fast ions and cause losses of particles and energy. In that case the heating efficiency decreases with reduced plasma performance as a consequence; in the case of severe losses there might also be damages to the first wall of the reactor. In present day fusion experiments tritium is typically not used as plasma fuel and to study interactions between fast ions and plasma instabilities one relies on creating populations of fast ions using auxiliary heating systems. A very special plasma heating scheme was tested for the first time at JET with the aim of accelerating as many deuterons as possible to energies of several MeV and thereby provoking plasma instabilities.

Neutron spectrometry of the emission from the DD reaction is particularly well suited to use as a diagnostic of deuterons in this energy range, and from this JET experiment we obtained data of unprecedented quality using the time of flight neutron spectrometer TOFOR. This allowed us to follow the time evolution of the fast deuterium population in several energy regions, and we could clearly correlate the existence of plasma instabilities with certain mode numbers to how well these energy regions were populated by deuterons. Some of the instabilities were also seen to be responsible for considerable redistributions and losses of fast ions, and from the spectral information of the neutron emission we could conclude that only fast ions at certain energy intervals were affected. All results are well in line with the resonant nature of the particle-wave interactions. Finally, the results we obtained are also important for development of fast ion diagnostics for next step fusion reactors such as ITER. Although one of the aims of ITER is to approach ignition conditions, and therefore self-sustained plasmas, ITER will still need a substantial amount of auxiliary heating (at least 1/3). One of the heating systems at will have conditions resembling the experiment studied here. Due to the high energies involved in the ITER heating system the neutron emission from the ITER heating will be similar to that from the experiment studied here. The very promising results obtained today at JET therefore bode well for using neutron spectrometry for diagnosis of the NBI heating at ITER.

Observations of neutron emission from impurity reactions at JET

When speaking about the neutron emission from fusion plasmas one typically refers to the emission from reactions between the two hydrogen isotopes deuterium and tritium (DT reactions) or reactions between two deuterium nuclei (DD reactions). For this reason neutron spectrometry of fusion plasmas is typically mentioned as a tool to diagnose populations of deuterium and tritium ions, see e.g. the contribution above. The DT reaction that produces
neutrons at 14 MeV is dominating in plasmas with equal mixtures of deuterium and tritium, which is planned for next step fusion reactors such as ITER. On the other hand, in today's experiment reactors running mainly with deuterium plasmas, the DD reaction producing 2.5 MeV neutrons is typically dominating.

![Figure 3.5](image-url)  
*Figure 3.5. Illustration of the neutron emission from DD reactions (solid red at $E_n = 2.5$ MeV) and from $^3$He + $^9$Be reactions (broken back from 1 to 9 MeV).*

However, in recent JET experiments the emission from other neutron producing reactions was investigated. This concerns reactions between beryllium and one of the two helium isotopes $^3$He and $^4$He. Because the Coulomb barrier is much higher for heavier elements it is required that the helium nuclei have energies of at least 1 to 2 MeV for these reactions to take place. In a number of experiments performed at JET either $^3$He or $^4$He was accelerated to energies in the MeV energy range in plasmas where beryllium was intentionally present as an impurity. Indeed, for these plasmas a prominent and anomalous neutron production in a multi-peaked spectrum between 1 and 9 MeV was observed and ascribed to reactions between energetic helium and impurity beryllium, see Figure 3.5. The results show that neutron spectrometry can be used not only to diagnose deuterium and tritium ions but energetic helium as well, if the conditions are appropriate. Reactions involving beryllium will also become more important in future operations since JET has recently replaced the first wall of its reactor vessel with plasma facing components made of beryllium. Beryllium will therefore be a natural impurity in the plasma. It is also foreseen that beryllium is to be used in the first wall at ITER. Because of this, neutron production from beryllium reactions need to be accounted for in the early stages of ITER operation where hydrogen ($^1$H) plasmas will be used to produce “neutron free” plasmas.
Experiments at MAST

The effect of MHD activity on the neutron emission has been measured with the neutron camera in parasitic experiments and an example is shown in Figure 3.6, where the effect of sawteeth is clearly seen on the neutron camera. An example of the neutron count rate from two different plasma regions can be seen in Figure 3.7, where the count rate from the two equatorial channels is shown together with other plasma parameter for a high neutron yield discharge. In addition to the collimated neutron count rate the camera can also provide basic spectrometric information as shown in Figure 3.8, where the pulse height spectra of the recoil proton in the detector is plotted for the two different lines of sight shown in Figure 3.7. This pulse height spectrum results from the convolution of the neutron energy spectrum with the detector response function. While pulse height spectra required long integration times for sufficient statistical accuracy, the count-rate can be measured with sub millisecond time resolution.

Figure 3.6. Sawtooth activity in a MAST discharge has observed by the fission chamber (top panel), the neutron camera (2nd panel), a SXR detector (3rd panel) and a Mirnov coil (bottom panel, spectrogram).
Figure 3.7. High neutron yield discharge at MAST: plasma current (top panel), line integrated density (2nd panel), SXR signal (3rd panel), neutron yield as measured by the fission chamber and NBI total power (4th panel), neutron camera count rate for two channels (bottom panel).

Figure 3.8. Recoil proton pulse height spectra generated by neutrons in the detectors of different lines of sight for the discharge shown in figure 3.7.
The neutron camera will be equipped with additional two detectors in 2011 for the off-axis measurement of the neutron emissivity profiles and radial scan capabilities will be implemented. Data analysis software is still under development as integration with the MAST data acquisition system.

3.1.3 Work on ITER related tasks

Within the European EFDA framework, the UU-ND group has conducted three different projects pertaining to instrumental and method developments for ITER.

1. Development of the method to determine the fuel ion ratio by neutron spectroscopy
2. Development of the time-of-flight neutron spectrometry technique
3. Development of the thin-foil (non-magnetic) proton recoil technique for neutron spectrometry

The projects were conducted within the EFDA Topical group for diagnostics (TG-DIA) and they received economical support both as Baseline support and Priority support, indicating the high significance attributed to these tasks. The three projects are discussed in some detail below.

Determination of the fuel ion ratio using neutron spectroscopy

Introduction

In an ITER plasma heated with deuterium NBI, the direct neutron emission seen by a neutron spectrometer will come from 6 different reaction channels. These are thermonuclear DT, DD and TT, beam-thermal DT and DD as well as beam-beam DD reactions, and the intensities scale as

\[ I_{th,dt} = n_d n_t \langle \sigma_{dt} v_{th} \rangle \]  
\[ I_{th,dd} = \frac{n_d^2}{2} \langle \sigma_{dd} v_{th} \rangle \]  
\[ I_{th,tt} = n_t^2 \langle \sigma_{tt} v_{th} \rangle \]  

\[ I_{bl,dt} = n_{nb} n_t \langle \sigma_{dt} v_{nb-th} \rangle \]  
\[ I_{bt,dd} = n_{nb} n_d \langle \sigma_{dd} v_{nb-th} \rangle \]  
\[ I_{bb,dd} = \frac{n_{nb}^2}{2} \langle \sigma_{dd} v_{nb-nb} \rangle \]

where the expressions within brackets are the reactivities.

In addition to these components, any measurement will also detect the indirect, continuum back-scatter emission. As can be seen by the equations above the emission intensities depend on the reactivities as well as on the deuterium and tritium bulk \((n_d \text{ and } n_t)\) and beam \((n_{nb})\)
densities. Using the values of one or several of the emission components, the fuel ion ratio \( n_t / n_d \) can be deduced.

The **traditional method** is to compare the thermonuclear DT and DD emissions (eq. 1 and 2). The ratio \( n_d / n_t \) can then be deduced as

\[
\frac{n_d}{n_t} = \frac{1}{2} \frac{I_{th,dt} \langle \sigma_{dd} v_{th} \rangle}{I_{th,dd} \langle \sigma_{dt} v_{th} \rangle}.
\]  

(7)

If the neutron emission is purely thermonuclear it suffices to measure the total DT and DD contributions. However, if supra-thermal fuel ion velocity components contribute to the neutron emission, e.g. from NBI heating, the thermonuclear and beam emission components must be separated with dedicated DT and DD spectrometers.

A **second option** is to combine the ratio \( I_{th,dd} / I_{nb,dd} \) (eqs. 2 and 5) with the ratio of the total DT and DD emissions \( I_{DT} / I_{DD} \). Since only the total DT emission is used this method does not require a dedicated DT spectrometer; it suffices with e.g. a time of flight spectrometer optimized for DD spectroscopy, such as TOFOR.

A **third option** is to combine eqs. 1 and 4; the ratio \( n_d / n_t \) can then be deduced by

\[
\frac{n_d}{n_t} = \frac{I_{th,dt} \left( \frac{I_{nb,dd}}{I_{th,dt}} \right)^2 \langle \sigma_{dt} v_{th} \rangle}{\langle \sigma_{dt} v_{th} \rangle}.
\]  

(8)

This method requires a reliable modeling of the heating neutral beams and their interaction with the plasma.

**ITER simulations**

The spectral components of the neutron emission were calculated with a Monte Carlo code that samples particle velocities and positions from thermal distributions as well as from a beam slowing down distribution [ref]. A model of the ITER S2 scenario was used as reference, where \( n_e = 1.1 \times 10^{20} \text{ m}^{-3} \), average core temperature \( T \approx 20 \text{ keV} \) and \( Z_{\text{EFF}} = 1.7 \). In the model, temperatures and densities for the thermal populations were available as function of minor radius, and the beam slowing down distribution was available in 4 dimensions: \( R,Z \) positions in the poloidal plane as well as energy and pitch angle. In 3.9 the calculated total neutron emission profile for the S2 scenario is shown together with the modeled viewing cone of the spectrometer system.

**Figure 3.9.** Calculated emission profile for the ITER S2 scenario. The space coordinates of the distribution functions are shown with small circles and the spectrometers viewing cone is indicated by horizontal lines.

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A parameter scan over temperature $T_i$, $n_e$ and $n_t / n_d$ was made by scaling the temperature and density profiles and recalculating the emission spectrum. The density of beam ions, $n_d$, was also scaled to reflect changing slowing down times when temperature and electron density changes. The scaling for $n_{nb}$ is approximately $T_e/n_e$.

![Figure 1.10. Calculated neutron emission from ITER S2 scenario for a $n_t / n_d = 1.0$ (left) and 0.01 (right).](image)

The capability to deduce the fuel ion ratio $n_d / n_t$ from the neutron spectrum was assessed using synthetic data for two spectrometer systems by folding the calculated spectra with the response function and performing the analysis on the synthetic data. A time of flight (TOF) spectrometer was used for spectroscopy of the DD emission and a thin foil proton recoil (TPR) spectrometer for DT spectroscopy. The experimentally verified response functions of the TOFOR and MPRu instruments at JET were here used to represent these spectrometer types. Both are well known, and experimentally verified in a large number of experimental conditions.

**Results on $n_d / n_t$**

**TOFOR spectrometer**

Some examples of the ability of a TOF spectrometer to distinguish different components in the neutron emission are shown in Figs 3.11 and 3.12. In ITER scenarios with appreciable tritium content, a component of energy degraded (originally 14-MeV) neutrons that have undergone backscattering at the far wall will be present. This emission component is here modeled by a polynomial:

$$
\frac{dN}{dE} = N \cdot \exp \left( C_1 E + C_2 E^2 \right)
$$

![Figure 3.11. Example of time-of-flight spectrum from the TOFOR neutron spectrometer including both the 14-MeV emission (at 27 ns) and the 2.5-MeV emission (at 65 ns).](image)
Figure 3.12: Analysis of TOFOR data in terms of emission components. Included are the thermal DD (full red), beam-thermal DD (dashed red), far-wall backscattered (dashed black) DD neutrons, as well as scattered DT neutrons (full black).

Figure 3.13 Capability to deduce $n_t/n_d$ ratio using a TOF with maximum rate handling set at $R_{NT} = 5 \times 10^{18}$ and $5 \times 10^{19}$.

**MPRu spectrometer**

Using equation 8, the fuel ion ratio can be deduced from the DT emission alone by separating the thermonuclear and beam-thermal components. The main contributor to the uncertainty is how well the ratio $I_{th,dt}/I_{th,dt} (R_{DT})$ can be extracted from the neutron spectroscopy measurements.

Figure 3.14 Capability to deduce $n_t/n_d$ ratio using the MPRu spectrometer for electron densities $n_e = 3$ and $8 \times 10^{19} \text{ m}^{-3}$. 
Continuing with the results from $I_{th,dt}/I_{nb,dt}$, the parameter scan is presented in 3.14 for $n_e = 3 \cdot 10^{19} \text{ m}^{-3}$ (left) and $n_e = 11 \cdot 10^{19} \text{ m}^{-3}$ (right). Starting with the low-density case we see that the method covers the range $T > 7 \text{ keV}$ and $n_t/n_d > 0.3$. The lower limit in $T$ is the result of the thermonuclear emission being drowned in the beam-thermal emission at low temperatures. The lower limit for $n_t/n_d$ stems from the subtraction in eq. 8, which is why this method works best for high tritium densities (i.e., $n_{dt} \gg n_d$). For low tritium densities ($n_{dt} \approx n_d$), the subtraction is made with two numbers of similar size, and small relative errors in $n_d$ could result in very large relative errors in $n_t$. On the other hand, had tritium beams been used the situation would be the opposite.

Figure 3.15: The thermal (full) and beam-thermal (dashed) spectral components used in determining the fuel density ratio from the 14-MeV emission only (equation 8). MPR data from DTE1 used.

At higher densities, the range in $n_t/n_d$ that can be measured shrinks considerably with increasing temperature. This can be understood by looking at the synthetic MPRu data presented in 3.15 for the two scenarios at $T = 20 \text{ keV}$ and $n_t/n_e = 1.0$. We see that $I_{th,dt}$ has increased by more than a factor 11 (eq. 1) while $I_{nd,dt}$ is unaffected by the density. At high densities the thermonuclear emission has a tendency to drown the beam-thermal emission and the number of data points that are used to determine $I_{nb,dt}$ decreases. Hence the error in $I_{th,dt}/I_{nb,dt}$ increases. It can be noted however that this is not a hard limit such as that imposed on the DD measurement by the DT backscatter.

Figure 3.16 Capability to deduce the fuel ion ratio $n_t/n_d$ using all combined neutron spectrometers in this investigation for electron densities $n_e = 3, 8, 11$ and $14 \cdot 10^{19} \text{ m}^{-3}$. 
Conclusions

Using a single TOF spectrometer, the fuel ion ratio nt/nd can be extracted within 20% accuracy from nt/nd = 10⁻³ to around 0.1. This also requires a variable collimation to increase the dynamic range of the TOF spectrometer. For higher tritium concentrations, nt/nd > 0.1, the backscattered neutrons from the DT emission will drown the DD emission, which can then not be measured. Using a high-resolution DT spectrometer of TPR type, the fuel ion ratio can, in a neutral beam heated plasma, be extracted using the information in the thermonuclear and beam-thermal DT emission. The parameter range that can be covered with 100 ms time resolution and a 10 cm collimator aperture is nt/nd = 0.2 to 9 for T > 10 keV and ne > 5×10¹⁹ m⁻³.

With a combined TOF/TPR spectrometer system, the fuel ion ratio can thus be obtained over the range nt/nd = 10⁻³ to 10², with the exception of the interval nt/nd = 0.1 – 0.2, where there seems to be no overlap between the two systems. However the lower limit for the TPR spectrometer is only due to insufficient statistics. A 4 times better statistics in the neutron spectrum improves the situation considerably and fills the gap between the spectrometers. The improved statistics can be achieved with either a longer integration time, and hence relax the requirement on 100 ms time resolution, or using a larger collimator in the spectrometer viewing cone. In that case, the combination of a TOF and TPR spectrometer can be used to measure the fuel ion ratio over the entire interval nt/nd = 10⁻³ to 10², and the temperature and density ranges increases to T > 5 keV and ne > 3×10¹⁹ m⁻³.

Time-of-flight neutron spectrometry

The work from July 2010 to the present has focused on the hardware side of the ToF technique development. For this purpose, collaboration with the Swedish data acquisition electronics company SP (Signal Processing) Devices has been established. The aim of the collaboration is to produce combined time and pulse height acquisition electronics of suitable performance for implementation with a time-of-flight neutron spectrometer in fusion research. The collaboration has been successful and a set of 3 data acquisition cards have now been ordered.

Specifications for the boards were worked out by analyzing the signal situation expected in a ToF application based on fast plastic scintillators, as is the most commonly used in fusion research so far. Here we have used the TOFOR system at JET as an indication of the experimental situation, and also used real data from this spectrometer to test our assessments. As has been shown with TOFOR, spectroscopy of the full energy range 1-20 MeV (i.e., including the main emission “lines” at 2.5 and 14 MeV) is possible with this technique (albeit with reduced resolution at the high energy end). For an adequate coverage of this energy range, a dynamic range in pulse heights of about 0 – 3 MeVee is required. The internal electronics contribution to the resolution for the 2.5-MeV measurement should be about 3% or less in order to contribute at a level similar or smaller than other effects, photon statistics for example. The smallest expected proton recoil energy in TOFOR is about 50 keVee, giving a statistical uncertainty in photon statistics of about 4%. Thus the pulse height uncertainty due to sampling digitization should not be more than 1.5 keVee, which, distributed over 3 MeVee gives a required bit resolution of 3000/1.5 = 2000, and consequently require a smallest number of effective bits (ENOBs) of 11 bits. Considering an internal loss of one bit resolution in the device itself (internal noise), we conclude that a nominal 12 bit ADC is sufficient.
The sampling rate should be selected to fulfill the Shannon sampling rule, which dictates that an adequate sampling frequency is (at least) twice the expected highest frequency of the sampled signal. Inspection of TOFOR signals (Fig. 3.17) shows that, with the 25m of RG58 signal cables used, no frequencies higher than about 250 MHz are present in the acquired waveforms, and consequently a sampling rate of 500 MHz (or better) should be sufficient. Indeed, an analysis of TOFOR waveforms show that data taken with a fast sampling oscilloscope (2 GHz) can be down-sampled to 500 mega-samples per second (MSPS) without loss of information in the subsequent reconstruction of the original pulse. In the example shown in Fig. 3.17a we have used a linear sum of sinc=sin(x)/x with time base 2 ns to adequately describe the pulse (except for a small contribution from noise); compare the original data (black line) with the reconstruction using the down-sampled data (red points and line). This can be contrasted with a situation where the sampling frequency is insufficient (200 MSPS) and clear signs of aliasing are visible, as shown in Fig. 3.17b (red curve deviates significantly from the original black). We conclude that a sampling frequency of (at least) 500 MSPS is required.

Other requirements on the acquisition boards include flexible, event-by-event triggering, sufficient on-board memory and fast transfer data rates.

A survey revealed that no commercial product available at the time (summer 2010) fulfilled all the requirements. Contacts with the Swedish electronics manufacturer SP Devices resulted in a fruitful collaboration, where they concluded that with modifications, their board design should be able to fulfil all the set requirements. An agreement was set up, where a 2-channel card of reduced capability (but based on the same fundamental design) was purchased for tests, with an understanding that it could be returned with full refund if at least two cards of full specification boards were eventually ordered.

Measurements were performed in order to test if the SP board design can indeed be used as envisaged in a ToF system. A crucial aspect of the required card is the ability to faithfully store the event time (stamp) of each waveform (trigger) in an event-by-event type acquisition. An initial test revealed a problem with the time stamping in this mode, where an uncertainty of (plus-minus) one sample in the time information was observed. This was identified as a
software problem in the programming of the boards and it was corrected by the company. In subsequent tests, data from a waveform generator (sine wave of 1.001 MHz frequency) were taken in continuous acquisition mode. The positive-slope zero-crossings were found (linear interpolation used between samples) and the time difference between zero-crossings was calculated and histogrammed. A period of 999 ns ± 150 ps (FWHM) was found; this also includes a contribution from the linear interpolation. The result was compared with acquisition in time-stamping mode where exactly the same result was obtained (see figure 3.18). The test also show that the uncertainty contributed by the board (<150 ps) is negligible compared to other sources, like geometry (2 ns).

![Graph of zero-crossing section of one waveform (blue), with the linear interpolation (red).](image)

**FIGURE 3.18:** Test of event timing using time stamps in event-by-event triggered mode. Top graph shows the extracted distribution of time differences between subsequent zero-crossings. Bottom graph is an example of the zero-crossing section of one waveform (blue), with the linear interpolation (red).

Finally, a preliminary test used both card channels in a coincidence set-up, where a pair of scintillators (each with dimension 10 x 100 mm) was arranged to measure cosmic muons. The observed width of the time difference distribution between the two detectors, including the full chain of scintillator – PMT – cable – acquisition card was about 800 ps, where waveform constant-fraction discrimination was used to pick the event time. The result again indicates that the electronics contribution to the ToF timing uncertainty can be much improved from the TOFOR present estimate of 1.7 ns. Thus we conclude that the event-based acquisition mode of the SP Devices boards performs as required.

Further discussions with SP Devices have fixed the remaining design parameters: a system based on the PXI bus standard for data transfer will be used and the final PXI cards will have 4 acquisition channels per card, each channel of 1 GSPS and 12 bit amplitude resolution. Three such cards have been ordered and will be tested as a final task within this contract.

The thin foil proton recoil (TPR) neutron spectrometer utilizes a thin foil upon which collimated neutrons impinge. Recoil protons with energy, $E_p$, are produced in the elastic scattering of the neutron in the foil related to the neutron energy, $E_n$, as:

$$E_p = E_n \cos^2 \theta_{np}$$
where $\theta_{\text{np}}$ is the scattering angle, i.e., the of the recoil proton with respect to the velocity vector of the impinging neutron. An annular silicon detector is positioned at some distance behind the foil. The pulse height spectrum of the protons in the detector is used to determine the proton (and thus the neutron) spectrum.

This work estimates the energy resolution and the efficiency, as well as the intensity of the background, of a TPR placed in Port Cell 1 at ITER using neutron transport codes (e.g. MCNPX) and FISPACT (activation code). The assumed scenario is a 500 MW ITER plasma, which results in a neutron yield of $1 \cdot 10^9$ n/cm$^2$ on the foil of the TPR (given the LOS assigned to a high resolution neutron spectrometer at ITER). The background events which arrive at the detector primarily consist of gamma particles and neutrons which have scattered in e.g. the foil, the collimator or other structures. The gamma particles originate from the scattered neutrons absorbed in the surrounding material, such as shielding and vacuum vessel. The MCNPX model of the TPR spectrometer installed at ITER, together with the simulated neutron flux in the spectrometer system, is depicted in figure 3.19.

In the simulations, the radius as well as the material of the cylindrical vacuum vessel of the MCNPX and FISPACT model has been altered. The materials considered are aluminum and stainless steel. The integrated signal to background ratio is shown in Figure. It can clearly be seen that the signal to background ratio is quite independent of the vacuum vessel radius and that aluminum should be chosen to achieve the best signal to background ratio.
Figure 3.20. Signal to background ratio as function of radius of vacuum vessel and its material. It can clearly be seen that aluminium (red) is preferred over stainless steel (black) as construction material for the vacuum vessel.

A Monte Carlo code simulates the neutron and proton transport in the spectrometer system. The simulation data is used to calculate the energy resolution, $\sigma/E$, and efficiency, $\varepsilon$, of the spectrometer system. The simulations have been performed with several settings. In the simulations, the annual detector radii $r_{in}$ and $r_{out}$ are fixed at 24 and 48 mm, respectively and are of silicon type. The thickness of the polyethylene foil has been varied from 0.01 to 0.5 mm (with a density of 1.05 g/cm$^3$) and the area of the foil has been varied from 100 to 1000 mm$^2$. Furthermore the foil-to-detector distance is varied from 50 to 1000 mm and the detector thickness is altered from 1 and 2 mm. It should be noted that a 2 mm silicon detector fully stops an 18-MeV proton.

The result of the simulation using a 2 mm thick detector is shown in Figure. In Figurea all sampling points are plotted as a function of energy resolution and efficiency (black dots). A few points in the, so called, Pareto frontier are colour coded. In this case, the Pareto frontier corresponds to the optimal efficiency given a certain energy resolution. In Figureb all sampling points are plotted again, but now as a function of foil thickness and foil-to-detector distance. Those points that correspond to the colour coded points from Figurea are highlighted with the same colour in Figureb. These set of points correspond to optimal settings. For example, if a resolution of 0.05 is needed the best achievable efficiency would be just below $1 \times 10^{-3}$ cm$^{-1}$ (red dot in Figurea) by choosing the thickness and distance combination of 0.36 mm and 150 mm (red dot in Figure 3b), respectively. The foil area of all points in the Pareto frontier is 1000 mm$^2$. The typical reciprocal behaviour of efficiency and resolution of the thin foil technique is clearly visible.

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1 These values are those of the S1 detector of Micron Semiconductor. (http://www.micronsemiconductor.co.uk/pdf/s.pdf)
Figure 3.21. (a) Efficiency and resolution of all sampling points (black points). It can clearly be seen that for a given resolution there is an optimal efficiency. A set of these optimal settings has been colour coded. (b) The sampling points (black dots) as a function foil thickness and foil-to-detector distance of the ideal case together with the colour coded set from (b).

In Figure 3.22, the same quantities as in Figure 3.21 are shown for the two simulations with different detector thicknesses, but omitting the sampling points. In Figure 3.22a, the Pareto frontiers are indicated for the simulations, where the 2 mm detector case is represented by those points that are connected by a red line and the thin detector case frontier a black line. From Figure 3.22 it can be deduced that the 1 mm detector can reach the same resolution as the 2 mm thick detector (apart from very low resolutions) given that we can increase the foil-to-detector distance far enough. However, it should be noted that the efficiency is always lower for each optimal setting for the 1 mm thick detector compared to the 2 mm detector.

Ideally, a TPR spectrometer is capable of varying its foil thickness and foil-to-detector distance. This would allow for flexible measurement conditions, where a high efficiency can be chosen when the demands on resolution are lower and vice versa.

The TPR is a promising system and the spectrometer concept will be further studied. More detailed simulations will be undertaken the coming year. Furthermore, concept drawings of the systems will be produced, describing mechanical methods of alterations of the foil-to-detector distance as well as the foil thickness. In addition, a cost estimate of different technical solutions will be conceived.
Publications section 3.1

Peer reviewed journals


Measurements of fast ions and their interactions with MHD activity using neutron emission spectroscopy, Nucl. Fusion 50 (2010) 084006


Reviewed conference proceedings and pre-prints


**Theses**

1. M. Gatu Johnson, Fusion plasma observations at JET with the TOFOR neutron spectrometer; Instrumental challenges and physics results, Ph.D. thesis, Acta Universitatis Upsaliensis 2010, Digital comprehensive summaries of Uppsala dissertations from the Faculty of Science and Technology 715

2. C. Hellesen, Diagnosing fuel ions in fusion plasmas using neutron emission spectroscopy, Ph.D. thesis, Acta Universitatis Upsaliensis 2010, Digital comprehensive summaries of Uppsala dissertations from the Faculty of Science and Technology 716


4. J. Eriksson, Calculations of neutron energy spectra from fast ion reactions in tokamak fusion plasmas, Master Diploma Thesis in Engineering Physics, Uppsala University, 2010
3.2 Plasma Spectroscopy

3.2.1 Dielectronic recombination resonances observed at the divertor of the JET tokamak

C. Jupén, L. Engström, H. Nilsson

We report on a rapid cooling mechanism triggered by the influence of the electron density on dielectronic recombination (DR) of C III, observed at the divertor of the JET tokamak for the first time (1). Briefly the DR process is initiated by interaction between a free electron and a positive ion producing continuous bremsstrahlung as a result of the energy loss with an amount depending on the distance between the electron and the ion. If the resulting kinetic energy of the free electron, relative to the rest system of the ground state of the ion in addition to its binding energy is matching the energy required for excitation of an inner shell electron, a resonant radiation-less dielectronic capture (DC) creating doubly excited configurations of the next lower ion is taking place. In present work we have studied doubly excited configurations of C II produced by DC involving capture of a free electron by the ground state 1s^22s^2 1S_0 of C III and an simultaneous inner shell excitation 2s to 2p producing bound doublet and quartet states of C II with the electron configurations 1s^22s2p(3P)nl L±1 with L as the orbital angular momentum of the captured electron and L the total orbital angular momentum of the electrons from the three open shells. There are two different paths for decay of these doubly excited states shown below schematically for a quartet state;

C III [1s^22s^2 1S] + e^- \rightarrow C II [1s^22s2p(3P)nL L±1] \rightarrow C II [1s^22s2p(3P)nL L±1] + h\nu \ (1)

The intermediate state [1s^22s2p(3P)nL L±1] can either de-excite by a photon emission into a lower bound state, 1s^22s2p(3P)nL L±1, which completes the DR process (the arrow to the right). The intermediate state [1s^22s2p(3P)nL L±1] can also make a radiation-less transition (the arrow to the left in (1)) to a continuous state with a subsequent auto-ionization which means ejection of an electron (Auger electron) having the same energy as the kinetic energy of the incoming captured electron, i.e. which means that no change of the C II ion has occurred.

In the theoretical expression of the intensity (I_{ph}) of an auto-ionizing satellite line given by the expression below of equation (2), p and b means upper and lower levels respectively. The intensity is proportional to electron density (n_e) and to the density of the ground state of C III (n_{CIII}) multiplied with the dielectronic capture rate coefficient (C_d) and branching ratio of captured states that stabilize before auto-ionization.

I_{ph} = n_e \cdot n_{CIII} \cdot C_d \cdot \frac{A_e}{(A_u + A_d)} \ (2)

The results are based on observations of lines of the 1s^22s2p(3P)3l L±4L - 1s^22s2p(3P)nL L±4L characters of C II for resonant upper states lying between 0.4 and 2.5 eV above the first ionization limit 1s^22s^2 1S_0, the ground state of C III. Figure 1 shows two of the investigated 2s2p4p ^4D and 2s2p4s ^3P quartet terms containing the ^4D_{3/2,5/2} and ^3P_{1/2,3/2} meta-stable states.
Spectra were recorded by a vertically viewing mirror link spectroscopic system with three Czerny–Turner spectrometers covering the near-UV through the near-IR wavelengths up to 11000 Å by use of Mark IIA, Mark IIGB and Mark IIHD divertors. In figure 1 is shown two decay branches starting from levels closely above the first ionization limit obeying this concept. One of these branches contains seven possible transition lines centered at 3585 Å. At lower electron density you are just seeing three of them, those labeled J=3/2-5/2, J=5/2-5/2 and the strongest one J=7/5-5/2. However, at a certain increase of the electron density other lines of the multiplet are becoming visible as can be seen in figure 2.

**Fig. 1 Transition scheme for some C II lines.**

**Fig. 2 Spectral recording of the 2s2p(3P)3p 4D – 2s2p(3P)4s 4P during the DR mode at 13.925s and metastable autoionization mode at 14.025s with Mark IIA**

**Atomic spectroscopy at the Astrophysical Department at Lund**

At the Lund astrophysical department laboratory spectroscopy is being performed for support of understanding of astrophysical spectral observations. Parameters required for analyses of astrophysical spectra of atomic and molecular nature comprise accurate wavelengths and wavelengths. Other parameters needed for as complete analyses as possible are transition probabilities and absorption cross sections ranging from UV up to IR wavelength region. There are plans trying to make an investigation of the Be I and C I spectra by means of a hollow cathode instrument. Of particular interest are the doubly excited states lying above the first ionization limit of these ions being subject to the counteracting dielectronic recombination and auto-ionization. As many of these states so far are unknown, theoretical calculations have been made.
3.2.2 Spectroscopy in support of ITER-like wall project (SIW)

T. Elevant and J.R. Drake

Spectroscopy in Support of the ITER-like Wall (SIW) is part of a package of diagnostic enhancements, which is to be implemented and used at JET during operation with an ITER-like wall during year 2011 and onwards.

**Background**

The primary material choice for ITER is a full beryllium main wall with CFC (Carbon Fibre Composite) at the strike points together with tungsten at divertor baffles and dome. The layout for all eight octants is shown in Fig.1.

![Fig. 3.2.2-1 New inner wall layout. The black circle marks the area that is observed KS3-diagnostic and the black square the area observed by KS8. The two vertical li. represent the area scanned by the two KT1 systems (not to scale).](image)

Beryllium

W-coated CFC

(3Be-coated) Inconel
Since this material combination has not been tested in a tokamak previously, the ITER-like Wall project has been launched at JET implying replacement of the present first wall by an ITER-like wall together with an adequate set of diagnostics.

Installation of the ITER-like wall and a new divertor on JET will result in qualitatively new spectroscopic needs and technical challenges. The metal surfaces can be subjected to large power density and thus high erosion rates. In case the full W divertor option is selected, it is conceivable that C levels will be so low that Be has to act as the reference species for low Z elements. Spectra from W are very rich, and may result in blending with spectral lines from other elements in all wavelength ranges. The spectral line best suited to study W erosion is the one at 401 nm because modelled and experimental results are available to convert photon fluxes into erosion rates. However, at this wavelength transmission losses by optical fibres are high, quantum efficiency of detectors is low and absolute calibration more difficult than for the spectral lines that best characterise C. For Be, spectral lines in the normal visible wavelength range exist, but many other suitable lines are in the blue as well. It will also be necessary to quantify the amount of Be released from W surfaces, the amount of W released from Be surfaces, and later in the experimental campaigns the release of Be and W from Be/W alloys once these have formed on the various surfaces.

Features of the enhanced systems
To meet the new demands a number of spectroscopic diagnostic systems are modified and enhanced. All instrumentation that is not installed in the torus hall will be located in a dedicated laboratory, Spectrometer Room (SR) in J1D, with a number of optical fibres routed from the various locations around the torus. Fibre patch panels in the spectrometer room will make the instrumentation interchangeable between different lines of sight should such need arise. The core emission from W is spread from the VUV (10nm-13nm) and XUV (4nm-7nm) to the Xray (0.1nm-0.4nm), with each spectral region being representative of a different part of the plasma, depending on the degree of ionisation and hence electron temperature.

The diagnostic uses 10 horizontal and 20 vertical viewing lines, connected to instrumentation via optical fibres. Light from these fibres is split between spectrometers with ~50 ms time resolution, and a set of Polychromator Assembly (PA) units with filters for ELM resolved fast time resolution. The horizontal lines of sight will observe Be limiters. The vertical lines of sight provide a spatial resolution in the divertor of ~30 mm using 10 lines for the inner and 10 for the outer divertor. Light in each optical fibre needs to be split further into more channels than previously, to cover simultaneously BeI, CII, WI and D<sub>α</sub>. Thus additional light channels with filters and fast PM tubes have been acquired and installed.

Role of Fusion Plasma Physics Laboratory
The responsibilities of VR are dealt with by the Fusion Plasma Physics Laboratory at KTH and include technical specifications, purchasing, installation, test, calibration and commissioning of the following installations.

24 units of Polychromator Assemblies (PA)
Briefly, each PA consists of a light fibre input, four beam splitters and four output channels, each one with a band pass filter and a fast PM-tube, overall 96 channels.

Previous tests have shown that all 96 beam line geometries are in agreement with specifications. Wavelength throughput have also been measured for all channels using a
calibrated tungsten-halogen lamp, a 0.6 mm optical fibre for light transmission and an Ocean Optics HR2000 detector.

During the 2010 commissioning of the 24 PA-units response functions of all PM-tubes to HV-variations were measured. The dark current as well as the stray light was compensated for. Ideally, response functions should show a linear dependence in a double log graph. Fig. 2 shows results for channel #1 in a set of 10 PM-tubes together with corresponding equations of each linear approximation and related least square fit, in terms of R-square numbers.

![Graph showing linear dependence in a double logarithmic diagram for one channel in 10 of the 96 PM-tubes.](image)

**Fig. 3.2.2-2 Gain vrs. high voltage show near linear dependences in a double logarithmic diagram for one channel in 10 of the 96 PM-tubes.**

R-square numbers imply a maximum deviation of 1% from linear dependence for all PM-tubes, which is well within specifications.
4 Concept improvements

4.1 Computational methods and beta limits

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In collaboration with
D.D. Schnack, Univ. Wisconsin-Madison, USA

Introduction
This project combines two areas of fusion research; a new method for efficient solution of initial-value problems, in particular for cases with multiple characteristic time scales, and applications of the method on confinement operational limits related to pressure driven instability.

There is a general need for fast and efficient computational solution of initial-value problems to model stability and transport problems in the presence of several characteristic time scales. This becomes increasingly urgent with the interest in confinement physics for high Lundquist numbers. We have developed a novel solution to this problem, namely spectral representation of the time domain. The resulting Generalized Weighted Spectral Method (GWRM) thus extends earlier spectral methods that were associated with the spatial domain only. It eliminates time step problems and allows for averaging over fast time scales in order to determine long time scale behaviour. The method has been shown to be applicable to a wide range of initial-value problems [1].

Pressure limits play major roles for operational beta limits in the tokamak (neoclassical tearing modes) and in the reversed-field pinch (resistive g-modes). Thus, we wish to apply the GWRM to investigate important recent theoretical results [H. Lutjens, J. F. Luciani and F. Garbet, Phys. Plasmas 8, 4267 (2001), and A. Bruno, J. P. Freidberg, and R. J. Hastie, Phys. Plasmas 10, 2330 (2003)] that imply that the modelling of these pressure driven modes is adequate only if heat generation and conduction effects are properly accounted for. In the particular case of the RFP, the generic instability to resistive g-modes (due to unfavourable curvature) is allegedly removed at low beta. As a consequence, the RFP could operate, free from resistive tearing, g-mode or rippling instabilities in steady state at low beta. This would strongly contribute to the reactor potential of the RFP. In this project, the theoretical results are compared with data from a fully resistive MHD plasma model including effects of ohmic heating as well as perpendicular and parallel heat conduction, using the GWRM.

Improved modeling of resistive g-modes
Because of the complexity of the strongly nonlinear MHD phenomena and the strongly separated Alfvén and resistive time scales, relatively limited physical effects have so far been included in numerical computations of the RFP. In this study we therefore wish to extend the understanding of operational limits of confinement.

Our earlier numerical simulations of RFP plasmas in the advanced regime, where current driven tearing modes are eliminated by current profile control, have provided favourable scaling of on-axis temperature and poloidal beta. Energy confinement scaling remains, however, too weak for reaching into the reactor domain. Pressure driven modes at moderate
and high beta are expected to play major roles here. To better understand and possibly find modifying effects to the limited energy confinement scaling, the MHD modelling should be carried out with more physical effects included. Thus ohmic heating, perpendicular and parallel heat conduction as well as the Hall and diamagnetic contributions to Ohm’s law need be modelled. Of particular interest is heat conduction. Using the classical, simplified energy equation (adiabatic assumption) in cylinder geometry, the following criterion was derived early. It says that when pressure is confined \((D > 0)\), resistive instability in the form of \(g\)-modes always results:

\[
\begin{split}
r_s \Delta' &= 2 \pi \frac{\Gamma(\frac{1}{4})}{\Gamma(\frac{3}{4})} Q^{5/4} (1 - \frac{\pi D}{4 Q^{3/2}}) \\
&= \frac{2 \pi}{\Gamma(\frac{3}{4})} Q^{5/4} (1 - \frac{\pi D}{4 Q^{3/2}}) 
\end{split}
\]  

(4.1)

Here \(r_s\) is the radial location of the resonant surface, \(D\) is a normalized pressure gradient, \(Q\) is the normalized growth rate and \(\Delta'\) is the standard jump across the resonance of the logarithmic derivative of the radial component of the perturbed magnetic field. At zero plasma beta \((D = 0)\) this relation predicts tearing instability if \(\Delta' > 0\), and at finite beta instability occurs also for \(\Delta' < 0\); these modes are called resistive \(g\)-modes. An RFP thus may be tearing mode stable but is, in this theoretical framework, always resistive \(g\)-mode unstable at high beta.

In the more recent work referred to above, heat conduction has been included into the energy equation. Using a number of orderings and approximations (including that of low beta), heat conduction effects were found to dominate over convection and compressibility, and the resistive layer was found to be characterized by a scale length \(\delta_x \propto (\chi_\perp / \chi_\parallel)^{1/4}\), being larger than the classical tearing mode scale length. This allegedly introduces a flattening of the perturbations and a stabilization of the resistive \(g\)-mode. The resulting criterion becomes

\[
r_s \delta_x = 2 \pi \frac{\Gamma(\frac{1}{4})}{\Gamma(\frac{3}{4})} Q^{5/4} \frac{r_s D}{2 \delta_x}. 
\]

(4.2)

Given the amount of orderings and assumptions in this model, in particular that of low beta, we wish to compare the results of Eqs. (4.1) and (4.2) with results for realistic beta using a linearised resistive MHD code which assumes resistivity in the entire plasma domain, where no ordering assumptions have been made and where ohmic heating as well as perpendicular and parallel heat conduction effects are fully accounted for. The GWRM is used as the basic method for this code, and the development has been quite a challenge, given that this method is new. As a next step, Hall and diamagnetic contributions to Ohm’s law will be included.

We have, during 2010, also developed \(\Delta'\) codes for zero and finite beta that may be of use also for other researchers. Preliminary results [2] show, contrary to the results of the authors mentioned above, the effects of heat conduction on resistive \(g\)-modes to be limited. More results are presently being produced and will be submitted for publication during late 2011.

**Time-spectral solution of initial-value problems – further development**

We have developed a new computational method for general initial-value problems that efficiently provides approximate solution of problems with several time-scales. Semi-analytical, explicit solutions are obtained for systems of both linear and non-linear partial differential equations. Employing time-spectral residual methods, approximate Chebyshev polynomial expansions are used to represent not only space but also time and physical
parameters explicitly. Time stepping is thus avoided completely. This eliminates numerical stability restrictions on the time domain, enabling problems with strongly separated time scales to be efficiently solved.

During 2010 the efficiency of the Generalized weighted residual method (GWRM) has been further improved through optimizing the use of temporal and spatial sub-domains. To avoid large memory allocation when inverting systems of Chebyshev coefficient matrices, the use of spatial sub-domains is essential. Ideally, these could be computed individually, with information provided for each internal boundary at each iteration of the solution. This is a non-trivial problem with respect to numerical stability, but substantial progress has been obtained during 2010. The employment of temporal sub-domains is simpler, but the characteristics of the solutions may influence how these should be implemented. A large number of problems have now been solved and benchmarked.

The GWRM work is now internationally recognized. As a result, one of us (Scheffel) was invited in 2010 by Nova Science Publishers, Inc to provide an account of the method as the first chapter in a new book on methods for partial differential equations [1]. The book is published in 2011.

The most challenging problem solved so far is the resistive MHD stability problem, including spatially dependent resistivity, ohmic heating and perpendicular and parallel heat conductivities as described above. A total of 14 coupled pde’s are solved simultaneously, yielding semi-analytical solutions that are functions of all variables time, space and resistivity.

In the neighbouring graph we show a result, using the GWRM code, for the time evolution of the perturbed radial magnetic field for a case with perturbation \((m,n) = (1,-2)\), including heat conduction at Lundquist number \(1.0 \times 10^3\) with \(\beta_o = 0.05\). A typical RFP equilibrium, being Suydam stable, is used. The growth rate exceeds that obtained when using the adiabatic energy equation. In the spectral representation, 5 temporal domains were used with 4 Chebyshev modes in each; furthermore 8 Chebyshev modes were used for the spatial domain and 2 for the resistivity domain.

**Publications section 4.1**

**Books**

**Other workshops and conferences**
5  Emerging technology

5.1 High purity ODS-tungsten materials

*M. Muhammed, S. Wahlberg, M. A. Yar.*

Tungsten and tungsten based materials are being considered as candidates for armour material facing the plasma in fusion reactor, but poor material properties are major hurdles. The less ductile nature or brittleness of tungsten materials is an obstacle during their processing as well as for their deployment in such extreme environment with high temperature and neutron radiation.

Material’s properties are directly related to its composition (purity) and microstructure. In recent years, some research has showed that mechanical properties in tungsten can be improved through grain-refinement. Further, nanostructured materials also show higher radiation resistance. Therefore research has been focused on the development of nanostructured but thermally stable oxide dispersed strengthened (ODS) tungsten composites with reduced brittleness. On one hand, plasma-wall interaction studies are progressive for evaluation of different tungsten based materials by high heat load testing and neutron radiation damage. On the other hand, on the basis of results, research is also proceeding to develop new techniques for fabrication and processing of tungsten based materials.

KTH is contributing for development of new powder metallurgy routes to fabricate high purity tungsten based materials with tailored microstructure. In this project we developed specific chemical powder metallurgical methods for the fabrication of nanoscale tungsten based powders. The processing is carried out in two steps:

i) Fabrication of doped tungsten containing precursors by a solvent mediated or a co-precipitation process, and

ii) Thermal decomposition and reduction of the obtained precursor into a tungsten composite powder.

Various chemical routes have been developed and studied for fabrication of both La and Y doped W powders and ODS-W composites sintered by spark plasma process (SPS) have been characterized using high resolution electron microscopy. The major findings have been published in well know journals of related field as mentioned below;


The materials produced by these developed methods, were not highly homogeneous. We, therefore, have undertaken a detailed study to identify the reason for the inhomogeneous distribution of the dispersed oxides.

Recently, we have developed another method for precursor fabrication also starting from APT and salts of Y, but carried out under acidic condition. The improved process involves a complete phase transformation in which the APT solid is converted into a tungstic acid based material. By control of the particle growth in solution, we are able to fabricate a nanostructure (<10nm) powder precursor, which can be reduced under hydrogen to a nanostructured tungsten-ODS powder. Major achievements regarding this research topic are listed below:

- Novel chemical routes have been developed to fabricate nanostructured tungsten powders and ODS-W composites.
- The resultant powders are highly homogeneous and uniformly dispersed by doping elements at molecular level and have been sintered into bulk material by SPS.
- Optimization of sintering (SPS) conditions for high density ODS-W-Y2O3 composites.
- New results have been presented at “1st FEMaS-Ca conference May 9-13 2011 Rosenheim, Germany/13th International Workshop on Plasma-Facing Materials and Components for Fusion Applications”.
- Development of samples for material’s evaluation for armour material application.
- High heat load testing of developed W-Y2O3 materials at Jülich, Germany, with other partners from the FEMaS-CA and EFDA program (results are under compilation for publication).

Developed nanostructured W-Y2O3 powder

Ultrafine grained Sintered W-composite.
The Swedish Association is very heavily involved in the JET activity. Some aspects of the scientific programme that support JET have been included in the earlier sections of this Annual Report covering the Work Programme of the Association. In this section the JET activity is described in more detail.

During 2010 JET was shut down for installation of the ITER-like wall. As a result there were no campaigns and no S/T orders in connection with campaigns. However the Swedish Association has been heavily involved in earlier campaigns and therefore substantial work was carried out under Art 6.3 Notifications.

The Swedish Association is involved in the Enhancement Programme in the areas of Erosion deposition diagnostics for the ITER-like wall project and Spectroscopy in support of the ITER-like wall project.

The RU is also involved in the JET Fusion Technology programme in two areas; Analysis of mirrors exposed in JET; first test mirror for ITER and Cross-sectional analysis of deposits on JET tiles.

Involvement of the RU in the EFDA JET Notifications and Orders are summarized in Table 6.1.

<table>
<thead>
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<td>Spectroscopy in support of ITER-like wall project</td>
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6.1 EFDA JET Enhancements

Erosion deposition diagnostics for ITER-like wall project

M. Rubel, D. Ivanova

The ITER-Like Wall (ILW) Project at the JET tokamak has an objective to explore plasma operation and plasma-wall interactions with a full metal wall: beryllium (Be) limiters and cladding on the central column in the main chamber and tungsten (W) in the divertor. Material erosion and fuel inventory studies are among top priorities. A large number of diagnostic tools has been developed and manufactured to elucidate on the overall material migration scenario. They are based either on transport tracers or on deposition monitors.

VR team has been responsible for the coordination of efforts focused on the development and procurement of several diagnostic tools for material migration studies. There were two major categories of items Beryllium marker tiles and Erosion-Deposition Probes (EDP). Specific scientific goals are:

- to determine material migration: erosion and deposition pattern in the main chamber and the divertor.
- to assess fuel inventory in the divertor.
- to improve understanding of the beryllium and tungsten behaviour and their impact on material migration and fuel inventory in a fusion reactor.

Beryllium marker tiles are the standard bulk limiter components (15 x 30 cm) additionally coated first with a thin nickel film (2-3 micrometers interlayer) and then with several micrometers of a dense Be layer obtained by thermo-ionic vacuum arc deposition. Markers tiles for tungsten erosion studies are contain a thin molybdenum interlayer between the bulk divertor tile and W overlayer. Both types of markers are embedded in regular limiter arrays. The qualification process of the markers comprised detailed analyses of the layers before and after high heat flux (HHF) testing. The results of high-heat flux testing (HHF) and detailed characterisation of materials will be presented and discussed. The characterisation, before and after HHF, was carried out using a large set of material analysis methods from ultra-high resolution microscopy and surface sensitive electron spectroscopy to ion beam analysis and X-ray diffraction. Coated W-Be probes are also inserted in the Be-covered inconel cladding tiles on the central column. The project is has been realised as a collaboration between the Associations VR, FZJ, TEKES, UKAEA and the JET Operator (JOC).

Within the project on Erosion–Deposition Probes several tools have installed on the main chamber wall and in the divertor:

- Rotating Collectors (divertor and main chamber wall)
- Quartz Micro-Balance (QMB) devices (in the divertor)
- Deposition Monitors (divertor)
- Wall Inserts (main chamber wall)
- Louver Clips (shadowed area in the divertor)
- Cassettes with Mirrors (divertor and main chamber wall).

The project has been realised in the cooperation of Associations VR, FZJ, IPP, UKAEA, CRPP and the JET Operator (JOC). All items have been procured, delivered to JET and installed for the ILW operation. Images in Fig. 6.1.1 show some probes installed on the inner divertor carrier (a) and under the full tungsten load bearing plate of the JET divertor (b).
Figure 6.1.1: Assembly of Erosion-deposition Probes in the divertor: (a) rotating collector and a cassette with in the inner divertor carrier; (b) cassettes with mirrors installed under the full tungsten load bearing plate of the divertor.

6.2 Fusion Technology at JET

6.2.1 First Mirror Test

*M. Rubel, D. Ivanova*

Metallic mirrors will be essential components of all optical systems for plasma diagnosis in a reactor-class device. First Mirror Test has been carried out at JET on the request of the ITER Design Team. Up to date two exposures have been performed in JET with carbon walls: 35 h of plasma operation (Step 1 in 2005-2007) and recently accomplished Step 2 (2008-2009): 45 h exposure with 32.7 h of X-point operation. The entire test (i.e. two steps) was performed with over 60 mirrors made of stainless steel (16 samples only in Step 1) polycrystalline molybdenum (Mo-poly) including 4 specimens coated with a 1 mm thick layer of rhodium
(Rh). Mirrors were installed in carriers (cassettes with channels) placed on the outer wall and in the divertor: inner leg, outer leg and base plate under the load bearing tile. Before and after exposure mirrors underwent detailed surface analysis with optical methods and ion and electron spectroscopy.

The aims of this work were: (a) to provide an overview of results based on the outcome of the two campaigns; (b) to discuss the performance of Rh-coated mirrors; (c) to consider options for mirror maintenance and cleaning in a steady-state reactor. Essential results are summarized by several points.

a) **Divertor**: Reflectivity of all mirrors in the divertor region has been degraded by 80-90%. It is caused by deposition of thick (> 35 mm), flaking-off coatings on surfaces as shown in Fig. 6.2.1.1a. The growth of a new layer is observed in places where such thick deposit peeled-off (Fig. 6.2.1.2). This leads to dust formation and large local differences in surface roughness and composition.

b) **Outer Wall** (Fig. 6.2.1.1b,c): The most important result is that only small reflectivity losses (5-10%) occur on Mo-poly mirrors at the channel mouth. This is due to the in-situ removal of deposited species by charge exchange (CX) neutrals.

c) **Composition**: Deuterium and $^{12}$C are the main species detected on surfaces, but other isotopes ($^{9}$Be and $^{13}$C) are also found in some locations thus indicating differences in the material migration.

d) **Rh-coatings** have 30% better initial reflectivity than Mo-poly. The coatings survived the test without detachment from the Mo substrate, but co-implantation of D and impurity atoms in mirrors on the outer wall caused material mixing: resultant reflectivity of pure Mo and Rh-coated mirrors was the same. It may indicate a limited use of coated mirrors.

e) **Deposition in channels** in the divertor cassettes is pronounced at the very entrance; the deposition sharply decreases with depth in the channel, $\lambda \sim 5-7$ mm. For the mirrors exposed on the outer wall the opposite tendency is observed: deposition increases with depth in the channel.

---

**Figure 6.2.1.1**: Visual observation of mirrors exposed: a) in the divertor base, b) on the outer wall at the cassette channel mouth, c) on the outer wall deep into the cassette channel (4.5 cm). Square-shaped dots on the mirrors are areas where SIMS measurements were performed.
6.2.2 Ion beam microanalysis of divertor surfaces in JET

H. Bergsåker, P. Petersson, G. Possnert

The erosion and migration of first wall material gives rise to several critical plasma-surface interactions issues for ITER and for other future big and high duty cycle fusion devices. The balance between erosion and deposition at different surfaces in the device determines the net erosion rate and consequently the life time of the plasma facing component. Deposition of thick layers at some surfaces is linked with co-deposition of fuel and consequently to the tritium inventory in a reactor. The build-up of thick deposited layers also entails dust production due to subsequent breaking and flaking of the layers. Materials migration and mixing may also modify the erosion rate and other surface properties. To study these issues at JET, microscopy, ion beam analysis techniques and SIMS have been frequently used for post mortem analysis of plasma facing surfaces. In JET with carbon wall, layers with thicknesses up to about one mm are produced in the divertor over extended periods of plasma operation. A recently developed method to study deposited layers on a microscopic scale employs micro ion beam analysis. This way elemental mapping of cross sections of deposited layers yields interesting information. Figure 1 shows an outline of the JET divertor configuration in different versions since 1999. Following every shut down at JET, samples of the divertor tiles have been removed for analysis. Figure 2 shows an example of D- and Be-distributions in layers that were deposited on tile 6, position 7, during the whole period from 1999 to 2007. The elemental composition shows very significant variation, in the first place in the direction perpendicular to the tile surface, where the layered structure is very pronounced both for D and Be, but also in the tangential direction on a scale length of a few hundred µm. Clearly, in order to estimate e.g. the total fuel content of the layers it is necessary to take into account the variations in elemental composition, it is not adequate to base any estimates e.g. just on the composition near the surface. It is also desirable to
understand what causes the variations in composition in the deposited layers, in order to be confident in any extrapolations to new conditions (ITER). Preliminary analysis suggests that the layered variations in the composition can largely be related to JET operations history, whereas in regions with large surface heating the deuterium had been mobile and segregated towards the surface and interface.

Figure 2. Elemental maps of a cross section of the deposited layer at position 7 on tile 6 (bottom of the outer divertor). a, Deuterium distribution. b, Beryllium distribution. c, Optical micrograph of layer cross section. d, Projected average depth profile of D and Be throughout the layer.

Publications section 6.2

Peer reviewed journals


Presentations at conferences

3. P. Petersson, Cross sections of deposited layers investigated by micronuclear reaction analysis, Conference on Plasma Surface Interactions in Controlled Fusion Devices, San Diego, California, USA, 24-28 May
The EFDA Work Programme for 2010 included activities specified in Task Agreements and Topical Groups. VR had activity in two EFDA Task Forces: Task Force Integrated Tokamak Modelling (TF-ITM) and EFDA Task Force Plasma Wall Interaction (TF-PWI). VR is active in both of these Task forces. In particular, the Task Force Leader for TF ITM was Pär Strand, the Deputy Task Leader for TF-ITM in the area ICRH Heating, Current Drive and Fast Particles was Thomas Johnson. Also Marek Rubel was a task co-coordinator in the TF PWI for the task area of dust generation. VR had activity in four Topical Groups: Materials, Diagnostics, MHD and Transport. The scientific information for the activity is reported in the earlier sections covering the Association Work Programme. Here we list the tasks for the respective Task Forces and Topical Groups.

### 7.1 Task Force Integrated Tokamak Modelling

An overview of the activity in the Task Force Integrated Tokamak Modelling follows:

**WP10-ITM-TFL1 Task Force Leadership**

WP10-ITM-TFL-VR: Task Force Leadership: 0.75 PY priority support (P. Strand)

WP10-ITM-PL-IMP5-01-VR Coordination of Integrated Modelling: Project IMP#5 H& CD 0.25 PY priority support (T. Johnson)

**WP10-ITM-IMP12**

WP10-ITM-IMP12-ACT 9-T1-01/VR. Resistive wall modes: 0.1 PY Baseline Support (D. Yadikin)

**WP10-ITM-IMP3. Transport Code and Discharge Evolution.**

WP10-ITM-IMP3-ACT 1-T2-01/VR: Finalisation IMP3 core modules: 0.08 PY Baseline Support (P. Strand)

WP10-ITM-IMP3-ACT 1-T4-01/VR: ETS V&V: 0.04 PY Priority Support (P. Strand)

WP10-ITM-IMP3-ACT 2-T1-01/VR: ITER Scenario modelling: 0.20 PY Baseline Support (P. Strand and H. Nordman)

**WP10-ITM-IMP4. Transport Processes and Micro stability.**

WP10-ITM-IMP4-ACT 5-T1-01/VR: Maintenance of transport modules: 0.08 PY Baseline Support (P. Strand)

**WP10-ITM-IMP5. Heating current Drive and Fast Particles.**

WP10-ITM-IMP5-ACT1-T1-01/VR: Adaption of IMP5 codes for use with ITM: 0.35 PY Priority Support (T. Hellsten, J. Höök and A.1 Hannan)

WP10-ITM-IMP5-ACT1-T1-02/VR: Adaption of IMP5 codes for use with ITM: 0.1 PY Baseline Support (A. Hannan)

WP10-ITM-IMP5-ACT2-T4-01/VR: Data structure, ICRH: 0.05 PY Baseline Support (T. Hellsten)
WP10-ITM-IMP5-ACT3-T1-01/VR: Benchmarking and validation of codes: 0.7 PY Baseline Support (T. Hellsten, A. Hannan and T. Johnson)
WP10-ITM-IMP5-ACT4-T1-01/VR: Development of advanced 3D Fokker-Planck solver for ions: 0.6 PY Baseline Support
WP10-ITM-IMP5-ACT4-T1-02/VR: Development of advanced 3D Fokker-Planck solver for ions: 0.4 PY Priority Support (ACT4-T1-01 and -02 together) T. Hellsten, T. Johnso, J. Höök, Q. Mukhtar and A. Hannan

7.2 Task Force Plasma Wall Interaction

An overview of the activity in the Task Force Plasma Wall Interaction follows:

**WP10-PWI-01. Fuel Retention as a function of wall materials foreseen for ITER**
WP10-PWI-01-02-01/VR/BS: Characterisation of retention mechanisms: 0.05 PY (M. Rubel)
WP10-PWI-01-02-01/VR/PS: Characterisation of retention mechanisms: 0.20 PY Priority Support (G. Possnert)
WP10-PWI-01-02-01/VR/PS: Characterisation of retention mechanisms: 10 kEuro hardware Priority Support Art 8.2b (M. Rubel, G. Possnert)
WP10-PWI-01-02-02/VR/PS: Characterisation of retention mechanisms: 0.40 PY Baseline Support (M. Rubel)

**WP10-PWI-02. Exploration of Fuel Removal Technologies Compatible with Retention in Mixed Materials and Metals, including Be**
WP10-PWI-02-01-01/VR/BS: Effectiveness of fuel removal by post mortem: 0.10 PY Baseline Support (M. Rubel)
WP10-PWI-02-03-01/VR/PS: Effectiveness of photonic cleaning: 0.25 PY (M. Rubel, G. Possnert)
WP10-PWI-02-03-01/VR/PS: Effectiveness of photonic cleaning: 10 kEuro hardware Priority Support Art 8.2b (M. Rubel, G. Possnert)
WP10-PWI-02-03-02/VR/PS: Structure of surfaces treated by various: 0.5 PY Baseline Support (M. Rubel)

**WP10-PWI-03. Dust generation and characterisation in different devices**
WP10-PWI-03-00/VR/PS: Leadership of the SEWG on dust in fusion devices M. Rubel 0.25 PY Priority Support (M. Rubel)
WP10-PWI-03-01-01/VR/BS: Collection of dust in TEXTOR: 0.05 PY Baseline Support (M. Rubel)
WP10-PWI-03-01-01/VR/PS: Collection of dust in TEXTOR: 0.15 PY Priority Support (M. Rubel, G. Possnert)
WP10-PWI-03-01-01/VR/PS: Collection of dust in TEXTOR: 5 kEuro hardware Priority Support Art 8.2b (M. Rubel, G. Possnert)
WP10-PWI-03-01-02/VR/PS: Comparison of deposits: Assess: 0.30 PY Baseline support (M. Rubel)
WP10-PWI-03-03-01/VR/BS: Characterisation mobile dust…aerogel samples…: 0.15 PY Baseline Support (H. Bergsåker)
WP10-PWI-03-03-01/VR/PS: Characterisation mobile dust…aerogel samples…: 0.30 PY Priority Support (H. Bergsåker)
WP09-PWI-04. Erosion, transport deposition of low Z
WP10-PWI-04-01-01/VR/PS: Studies of local re- and co-deposition: 0.10 PY Priority Support (M. Rubel, G. Possnert)
WP10-PWI-04-01-01/VR/PS: Studies of local re- and co-deposition: 5 kEuro hardware Priority Support Art 8.2b (M. Rubel, G. Possnert)
WP10-PWI-04-01-01/VR/BS: Studies of local re- and co-deposition: 0.15 PY Baseline Support (M. Rubel)

WP10-PWI-05. Development of the PWI basis in support of integrated high-Z scenarios for ITER
WP10-PWI-05-01-01/VR/BS: Fuel retention in bulk tungsten exposed in TEXTOR: 0.05 PY Baseline Support (M. Rubel)
WP10-PWI-05-01-01/VR/PS: Fuel retention in bulk tungsten exposed in TEXTOR: 0.05 PY Priority Support (M. Rubel)
WP10-PWI-05-01-02/VR/PS: Structure of tungsten damaged…: 0.15 PY Baseline Support (M. Rubel)

WP10-PWI-06. Determination of expected alloys and compounds and their influence on PWI processes and fuel retention.
WP10-PWI-06-01-01/VR/PS: Formation and composition on mixed: 0.10 PY Priority Support (M. Rubel)
WP10-PWI-06-01-01/VR/PS: Formation and composition on mixed: 5 kEuro hardware Priority Support Art 8.2b (M. Rubel, G. Possnert)
WP10-PWI-06-01-02/VR/BS: Formation and properties on mixed: 0.15 PY Baseline Support (M. Rubel)

7.3 Topical Group Fusion Materials

An overview of the activity in the Topical Group Materials follows:

WP10-MAT-WWALLOY: Tungsten Tungsten alloys development
WP10-MAT-WWALLOY-03-02/VR/BS Novel fabrication of high purity nanostructured ODS tungsten materials: 2.0 PY Baseline Support (S. Wahlberg)

7.4 Topical Group Diagnostics

An overview of the activity in the Topical Group Diagnostics follows:

WP10-DIA-01 Diagnostic for burning plasmas
WP10-DIA-01-03-xx-01/ VR/ BS: Measurement of fuel ratio: 0.3 PY Baseline Support (G. Ericsson).
WP10-DIA-01-04-xx-01VR/ PS High resolution neutron spec for ITER 0.70 PY Priority Support (G. Ericsson).
WP10-DIA-01-04-xx-01VR/ PS High resolution neutron spec for ITER 46 kEuro hardware and consumables Priority Support Art 8.2b (G. Ericsson).
WP10-DIA-01-04-xx-02VR/ BS High resolution neutron spec for ITER 1.65 PY Baseline Support (G. Ericsson).

### 7.5 Topical Group MHD

An overview of the activity in the Topical Group MHD follows:

**WP10-MHD-01 Fast particle physics**
WP10-MHD-01-01-xx-01/VR/BS: Co-ordinated experiments fast particles MAST: 0.40 PY Baseline Support (M. Cecconello)

**WP10-MHD-02 Disruptions**
WP10-MHD-02-01-xx-01/VR/PS: Numerical simulations runaway electrons: 0.40 PY Priority Support (T. Fülöp)
WP10-MHD-02-01-xx-02/VR/BS: Numerical simulations runaway electrons: 0.60 PY Baseline Support (T. Fülöp)

**WP10-MHD-03: NTMs ELMs, RWMs**
WP10-MHD-03-03-xx-01/BS Stability at high beta…RWM control: 1.75 PY Baseline Support (P. Brunsell)
WP10-MHD-03-03-xx-02/BS Stability at high beta…plasma rotation braking: 0.75 PY Baseline Support (P. Brunsell)
WP108-MHD-03-03-xx-03/PS Stability at high beta…TM control: 0.50 PY Priority Support (P. Brunsell)

### 7.6 Topical Group Transport

An overview of the activity in the Topical Group MHD follows:

**WP10-TRA-03: Particle and impurity transport**
WP10-TRA-03-03-xx-02/VR BS: Quasilinear impurity transport GYMES: 0.6 PY Baseline Support (T. Fülöp)

**WP10-TRA-04: Role of neoclassical and turbulent mechanisms in plasma rotation**
WP10-TRA-04-02-xx-01/VR BS: Predictive simulations of poloidal and toroidal rotation: 0.65 PY Baseline Support (J. Weiland)
WP10-TRA-04-02-xx-01/VR PS: Predictive simulations of poloidal and toroidal rotation: 0.2 PY Priority Support (J. Weiland)

**WP10-TRA-05: Statistical properties of edge turbulent transport**
WP10-TRA-05-03/VR BS: Theory and modelling of turbulence: 0.2 PY Baseline Support (J. Weiland)
7.7 Goal Oriented Training (GOTiT).

An overview of the activity in the Goal Oriented Training Programme in Theory follows:

GOTiT is under Art 8.2e of the Contract of Association and is included in the Statement of Accounts and Article 7 of the EFDA WP08-GOT-GOTiT (multi-year contract)

T. Hellsten co-ordinator. 24 ppm (91.2 kEuro) cost for trainees, 4.5 ppm (39.2 kEuro) cost for mentor, 16.5 kEuro cost other. Total cost 146.9 kEuro eligible for special support (ceiling 58.76 kEuro)
8 ITPA

8.1 Overview of ITPA activities.

The International Tokamak Physics Activity (ITPA) provides a framework for internationally coordinated fusion research activities. The ITPA continues the tokamak physics R&D activities that have been conducted on an international level for many years. This has resulted in the achievement of a broad physics basis essential for the ITER design and useful for all fusion programs and for progress toward fusion energy generally.

The ITPA operates under the auspices of ITER. The Participants in the ITPA are the Members of ITER. The organizational structure of the ITPA consists of a Coordinating Committee (CC) and several Topical Physics Groups.

The role of the ITPA Coordinating Committee is to oversee the Topical Physics Groups in conducting their tasks and to interface the ITPA with the ITER Organization. It is composed of three representatives from each Participant and the ITER Organization.

Representatives of the ITPA and the experimental fusion facilities meet annually with representatives of the International Energy Agency (IEA) to encourage collaborations. The IEA sponsors several Implementing Agreements that foster joint collaborations. The resulting process has grown to involve all the IEA implementing agreements and nearly all tokamaks.

In the fall of each year, the ITPA, through its Topical Groups, prepares a report on the previous year's joint experiments and a proposal for a set of joint experiments for the coming year. The ITPA CC chair presents this proposal to the world's tokamak program leaders in a meeting in the December time frame. At this meeting, the joint experiments are discussed, and commitments are sought from the various tokamak program leaders. An international participant team is identified and a spokesperson defined. The tokamak leaders seek to implement these joint experiments within their normal experimental planning processes.

Since 2008 there has been an increased focus on addressing remaining uncertainties in the physics related to ITER design and operation.

The Swedish Research Unit has participated in the following ITPA Topical Groups:

- Diagnostics (p. 85-88)
- MHD, Disruptions & Control (p. 44)
- Scrape Off Layer and Divertor (p. 30)
- Transport and Confinement (p. 9-10)

See respective page reference for the scientific information related to the Swedish ITPA activities.
9 Other activities

9.1 Training and education

PhD training
The physics programme of the Swedish Fusion Research Unit is university based. There are PhD programmes at CTH, KTH and UU. During 2010 there were 17 PhD students included in the Research Unit. On the average there are about 3 to 4 PhD examinations per year. This is slightly less than in previous years, due to a decrease in funding for PhD students. However there is no difficulty in recruiting students when funding is available.

Masters programmes
In addition to the PhD programme the three universities have Master of Science programmes where students can select fusion plasma physics topics for their thesis work. Annually there are about 15 MSc thesis students.

9.2 Public information

The Swedish Public Information Network (PIN) officer within EFDA is professor Jan Scheffel at the Division of Fusion Plasma Physics, KTH in Stockholm.

The Swedish PIN activities are mainly carried out in the form of lobbying, concentrated towards contact with government, parliament and energy administrators with the intention of strengthening fusion funding. The national funding for fusion is unchanged since the early 1990’s, and the situation is quite difficult with regards to retaining competence and enrolling younger scientists. The Swedish Energy Authority has a mission to consider fusion but has sofar essentially not supported fusion research.

The year 2010 has been an intense year with substantial contact also with the public. There is presently a strengthened interest in fusion among school pupils and university students, most likely because of the debate on global warming and the energy future.

Performed PIN activities
Lobbying
- Personal contact with politicians and administrators in the energy sector
- Articles on fusion, to Swedish newspapers or produced by independent journalists

Contacts with the public
- Interview of PIN officer in Swedish national television’s most viewed news program, where the lack of fusion support from the Swedish Energy Authority was questioned
• A stand with fusion information at the largest Swedish conference on energy, with over 2000 attendants gathered for two days

• Talk at ABB Industry in Västerås, Sweden, and discussions concerning possible cooperation with industry

• Lectures on fusion to school children, students, associations and at other universities (for example; Lund and Linköping Universities)

• Tutorship for several groups of upper secondary pupils doing project work in fusion

• Cooperation with the science center “House of Science” on “science points”, where upper secondary school children come to visit the Alfvén Laboratory, see the fusion experiment Extrap T2R and do basic experiments.

Contacts with students, taking university courses

• Students, to become upper secondary school teachers, take course on fusion and energy

• A popular university summer course on energy includes lectures on fusion. Professor Scheffel provides these lectures.

Publications for a general audience

Appendix I:  
Fusion for Energy Grants

The Joint Undertaking for ITER and the Development of Fusion Energy, called fusion for Energy or F4E, has the responsibility for procuring the necessary research and development as well as the equipment for ITER. F4E can provide support to groups in the member states in the form of Grants, which cover 40% of the costs of the Research and Development.

This F4E activity is not a part of the EURATOM fusion programme, which provides contributions to the Research Units as established in the Contract of association and the European Fusion development Agreement. However it is important that the EURATOM programme maintains an effective contact with F4E since the EFDA programme must have a programme where preparations for ITER have the highest priority. Therefore information of Association VR activity for F4E is provided here for information.

During 2010, Studsvik Energy AB has had the following activity summarised in Table A.1.

<table>
<thead>
<tr>
<th>Contract No.</th>
<th>Title</th>
<th>Type</th>
<th>Comments</th>
</tr>
</thead>
<tbody>
<tr>
<td>F4E-2008-GRT-ES-SF-01</td>
<td>New analyses of fire reference events scenarios</td>
<td>Grant</td>
<td>Ongoing</td>
</tr>
<tr>
<td>F4E-2008-GRT-021 (MS-VV)</td>
<td>Corrosion issues</td>
<td>Grant</td>
<td>Ongoing</td>
</tr>
<tr>
<td>F4E-2009-GRT-031 (ES-FS)</td>
<td>Safety assessment of EU TBM</td>
<td>Grant</td>
<td>Ongoing</td>
</tr>
<tr>
<td>F4E-GRT-243 (ES-MF)</td>
<td></td>
<td>Grant</td>
<td>Accepted</td>
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<tr>
<td>F4E-OPE-244 (ES-MF)</td>
<td></td>
<td>Grant</td>
<td>Accepted</td>
</tr>
</tbody>
</table>

Total 4231729 SEK
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