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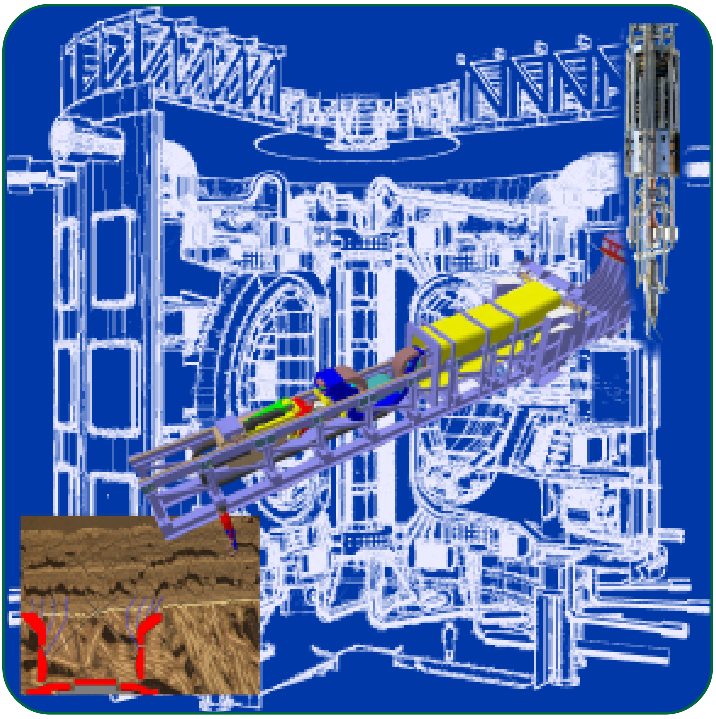

FUSION YEARBOOK

Association Euratom-Tekes Annual Report 2005

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Association Euratom-Tekes
Annual Report 2005

Seppo Karttunen, Karin Rantamäki & Anne Marttila (eds.)

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Valopaino Oy, Helsinki 2006

FOREWORD

The FUSION technology programme by Tekes provides the national frame for the fusion research activities of the Association Euratom-Tekes. It is fully integrated into the European Fusion Research Area in the 6th Framework Programme. The emphasis of the Finnish fusion activities is on EFDA technology research and development work, which is carried out in close collaboration with Finnish industry. The focus in our plasma physics and plasma-wall research is in the EFDA JET Workprogramme and in the AUG experiments at IPP Garching.

The year 2005 was remarkable for the international fusion research community. In June, the six ITER parties agreed to site the ITER facility in Europe. The European site at Cadarache in Southern France provides excellent settings for ITER and its team in terms of technical infrastructure and social environment. In December 2005 India joined the ITER project as the seventh Party, which means that more than one half of the mankind is now behind ITER. The final stage of the ITER negotiations has been successful and the ITER Joint Implementing Agreement is expected to be completed in early 2006. In Europe, preparations for ITER are advancing and *the European Joint Undertaking for ITER and the Development of Fusion Energy* will be established in Barcelona later in 2006. The fusion research budget for the seventh Framework Programme is not yet consolidated. It is extremely important to ensure also sufficient funding in order for a vigorous fusion research programme in parallel with the ITER construction. This is the only way to keep Europe in the leading position in the worldwide fusion research.

The research activities of the Association Euratom-Tekes are focused into two areas: (i) Physics work for EFDA JET and AUG experiments and (ii) vessel/in-vessel field of the EFDA technology programme. The Tekes contributions to JET in 2005 comprised modelling of real time control of transport barriers, radio-frequency heating, predictive integrated modelling of tokamak plasmas, diagnostics and studies on material transport in the edge plasmas supported by the surface analysis of the divertor and limiter tiles from JET.

The technology work in the vessel/in-vessel field included preparations for the ITER Divertor Test Platform (DTP2) at VTT in Tampere. The DTP2 host activities are managed by VTT and the Tampere University of Technology. The other vessel/in-vessel activity is materials research, which covered mechanical testing of reactor materials under neutron irradiation, characterisation of irradiated CuCrZr/SS joints, further development of beam welding with filler wire and demonstration of inter-sector welding robot. Some effort was also devoted to modelling of radiation damage in Eurofer, neutron shielding for IFMIF design and socio-economic studies.

Finally, I would like to express my sincere thanks to the whole team of the Finnish Research Unit and to the companies involved, for the hard work and successful contributions to the European Fusion Programme.

Seppo Karttunen
Head of the Research Unit
Association Euratom-Tekes

CONTENTS

1 INTRODUCTION

The “FUSION” technology programme for 2003–2006 is the national frame for the fusion research activities of the Association Euratom-Tekes. The FUSION programme is fully integrated into the European Fusion Programme in the 6th Framework Programme or the European Fusion Research Area. Financing is provided by Euratom and nationally by Tekes (National Technology Agency of Finland), Finnish Academy, participating institutes and industry. The Association Euratom-Tekes was established in 1995 and the present Contract of Association between Euratom and Tekes extends to the end of 2006. The total budget of the Fusion Research Unit is about 3.5 M€ corresponding to the manpower over 35 ppy. Other agreements of the EU Fusion Programme include the multilateral European Fusion Development Agreement (EFDA), JET Implementing Agreement (JIA) and the Staff Mobility Agreement. The Research Unit of the Association Euratom-Tekes covers research groups from

- Technical Research Centre of Finland (VTT)
- Helsinki University of Technology (TKK)
- Tampere University of Technology (TUT)
- Lappeenranta University of Technology (LUT)
- University of Helsinki (UH)

This Annual Report summarises the fusion research activities of the Finnish Research Unit of the Association Euratom-Tekes in 2005. The activities of the Research Unit are divided in the Fusion *Physics Programme* and the Fusion *Technology Programme*.

The Physics Programme is carried out at the Technical Research Centre of Finland (VTT) Helsinki University of Technology (TKK) and University of Helsinki (UH). The research areas of the Physics Programme are

- Heat and particle transport, MHD physics and plasma edge phenomena
- Radio-frequency applications – heating, current drive and diagnostics
- Plasma-wall interactions and material transport in SOL region

Association Euratom-Tekes participates actively in the EFDA JET Workprogramme 2003–2005 and JET operations. Two persons (Johnny Lönnroth and Marko Santala) were seconded to the UKAEA operating team in 2005. One person (Tuomas Tala) acted as a Deputy Task Force Leader in TF T (transport). S/T Order/Notification activities continued the work that started in 2000. A major part of the physics activities of the Research Unit are carried out in collaboration with other Associations with the focus on EFDA JET work. In addition to EFDA JET activities, the Tekes Association contributed to the experimental programme of ASDEX Upgrade (AUG) in 2005.

The Technology Programme is carried out at the Technical Research Centre of Finland (VTT), Helsinki University of Technology (TKK), Tampere University of Technology (TUT) and Lappeenranta University of Technology (LUT) in close collaboration with Finnish industry. The main companies involved in 2005 activities are: Fortum (Finnish EFET partner), Outokumpu Poricopper, Metso Powdermet, Hollming Works, Diarc Technology, Creanex, Hytar, Adwatec, Finpro and Prizztech Oy.

The technology research and development in 2005 is focused on the fusion reactor vessel/in-vessel area:

- Multimetal in-vessel components, joining technology
- In situ materials testing and characterisation under neutron irradiation
- Plasma facing materials and coatings including erosion studies, material transport and tritium issues
- Beam welding and welding/cutting robotics
- Remote handling of divertor maintenance
- Preparation of the ITER Divertor Test Platform (DTP2)

In addition, activities in Underlying Technology, Breeding Blanket Materials and Socio-Economic studies will continue. Two persons (Hannu Kaikkonen and Hannu Rajainmäki) were seconded to the EFDA CSU – Garching in 2005.

Technology collaboration was active with Euratom Associations FZK Karlsruhe (IFMIF, ITER neutronics), Risø Roskilde and SCK-CEN Mol (In-reactor materials testing), UKAEA Culham (JET Technology), CEA Cadarache and ENEA Brasimone (divertor maintenance tools and manipulators).

The following Staff Mobility actions took place in 2005

1. Markus Airila from HUT to IPP Garching, 41 days, 24 April – 3 June 2005 and to the Task Force E meeting at CRPP Lausanne, mission 4 days, 18-21 January 2005.
2. Ville Hynönen from HUT to to IPP Garching, 30 days, 11 July – 9 August 2005.
3. Timo Kiviniemi from HUT to UKAEA Culham, 28 days, 16 January – 12 February 2005.
4. Taina Kurki-Suonio from HUT to IPP Garching, 52 days, 17 February – 2 March, 11 July – 2 August and 3 – 18 October 2005.
5. Jari Likonen from VTT to UKAEA Culham, 30 days, 29 June – 24 July, and 23-29 October 2005, to the Task Force E meeting at CRPP Lausanne, mission 4 days, 18-21 January 2005 and to the EU Task Force PSI planning meeting at CEA Cadarache, mission 4 days, 16-19 October 2005.
6. Johnny Lönnroth from HUT (JOC Secondee) to the Task Force E meeting at CRPP Lausanne, mission 4 days, 18-21 January 2005 and to DIII-D San Diego, USA, mission 7 days, 6-12 November 2005.
7. Markus Nora from HUT to the Task Force T meeting at UKAEA Culham, mission 6 days, 23-28 January 2005.
8. Karin Rantamäki from VTT to the Task Force T meeting at UKAEA Culham, mission 6 days, 23-28, January 2005, to the Task Force H meeting at CEA Cadarache, mission 4 days, 20-23 February 2005 and to the Task Force S2 meeting at UKAEA Culham, mission 6 days, 18-23 September 2005.
9. Tuomas Tala from VTT to the Task Force T meeting at UKAEA Culham, mission 6 days, 23 January – 3 February 2005.
10. Frej Wasastjerna from VTT to Forschungszentrum Karlsruhe, 84 days, 19 June – 30 July and 1 November – 12 December 2005.

11. Sami Saarela from VTT to SCK-CEN Mol, 40 days, 17 October – 4 November and 28 November – 18 December 2005.

The total volume of the mobility actions in 2005 was 362 days (11.9 person-months). More detailed Staff Mobility reports are given in Section 10.4. In addition, there were several shorter visits outside the Staff Mobility arrangements. These are listed in Section 10.

Table 1-1: Summarise the EFDA Technology activities by the Association Euratom-Tekes in 2005.

Task Reference	Task Area / Task Title
	Physics Integration / JET Technology
TW4-TPP-TILCAR	Characterisation of erosion/ re-deposition balance in ITER-relevant divertor tokamaks and of PFCs after hydrocarbon removal by oxidative methods
JW6-FT- 3.24	Material transport, erosion / re-deposition in JET torus
JW5-TA-EP-BEW-02	Development of W-coatings for JET divertor tiles (Art 5.1b)
	Vessel / In-Vessel
TW5-TVM-CUSSPIT	Testing of irradiated CuCrZr/SS joints produced under different blanket manufacturing conditions
EFDA 01-602 (Am 2)	Ultrasonic tests of primary first wall panels and mock-ups (Art. 5.1b)
TW3-TVM-COFAT2	In reactor fatigue testing of Cu alloys
TW4-TVR-DTP2.1	CMM and SCEE design updates (Art. 5.1b)
TW4-TVR-DTP2.2	DTP2 platform design adaptation and QA system development (Art. 5.1b)
TW4-TVR-WHMAN	Development of water hydraulic manipulator
TW3-TVV-EBEAMS	Further development of e-beam welding with filler wire
TW4-TVV-ROBASS	Upgrade robot to include linear track
TW4-TVV-ROBOT	Further development of stability, safety and accuracy of IWR robot
TW5-TVV-IWRFS	Demonstration of IWR feasibility
Underlying Tech 1	Further development of novel methods and studies on radiation effects and verification of specimen size effects
Underlying Tech 2	Water hydraulic components further development
	Tritium Breeding and Materials
TW5-TTMI- 004	IFMIF design integration – 3D radiation shielding analysis and neutronics
EBP - SM	European blanket programme: tritium breeding materials
(TW5-TTMS-007-13a	Radiation damage in EUROFER: FeCrHe thermodynamics
	Magnets
TW3-TMSC-ATEST	Test of advanced Nb3Sn strand (Art. 5.1b)
	System Studies
TW5-TRE-FESO-1	Identification and comparative evaluation of fusion and other possible future energy production alternatives – TIMES global modelling
TW5-TRE-FESO-2	Fusion as a part of energy system
	EFDA CSU Secondments
EFDA CSU Garching Secondments	Hannu Kaikkonen, Administration / project control Hannu Rajainmäki, Magnets

2 FUSION PROGRAMME ORGANISATION

2.1 Objectives of the Programme

The Finnish Fusion Programme, under the Association Euratom-Tekes, is fully integrated into the European Programme, which has set the long-term aim of *the joint creation of prototype reactors for power stations to meet the needs of society: operational safety, environmental compatibility and economic viability.* The objectives of the Finnish programme (FUSION) is to carry out high-level scientific and technological research and to make a valuable and visible contribution to the European Fusion Programme and to the international ITER Project in our focus areas. This can be achieved by close collaboration between the Research Unit and Finnish industry, and by strong focusing the R&D effort on a few competitive areas. Active participation in the EU Fusion Programme and ITER provides a challenging opportunity for technology R&D work in research institutes and Finnish high-tech industry increasing to build up know-how and benefit from the technology transfer.

2.2 Association Euratom-Tekes

The National Technology Agency of Finland (Tekes) is funding and co-ordinating technological research and development activities in Finland. The Association Euratom-Tekes was established on 13 March 1995 when the Contract of Association between Euratom and Tekes was signed. Other agreements of the Association Euratom-Tekes include multilateral European Fusion Development Agreement (EFDA), JET Implementing Agreement (JIA) and Staff Mobility Agreement. Tekes was a member of the JET Joint Undertaking from 7 May 1996 until its end December 1999. The fusion research co-ordinator in Tekes is Mr. Juha Lindén.

2.3 Fusion Research Unit

The Research Unit of the Association Euratom-Tekes consists of several research groups from VTT and universities. The Head of the Research Unit is Mr. Seppo Karttunen from VTT.

The following institutes and universities participated in the fusion research during 2005.

1. VTT - Technical Research Centre of Finland

VTT Processes (co-ordination, physics, materials, socio-economics)
VTT Industrial Systems (materials, remote handling, welding, DTP2)
VTT Information Technology (diagnostics)

2. Helsinki University of Technology (TKK)

Department of Engineering Physics and Mathematics (physics, system studies)

3. University of Helsinki (UH)

Accelerator Laboratory (physics, materials)

4. Tampere University of Technology (TUT)

Institute of Hydraulics and Automation (remote handling, DTP2)
Laboratory of Electromagnetics (superconductors)

5. Lappeenranta University of Technology (LUT)
Institute of Mechatronics and Virtual Engineering (remote handling)

The following industrial companies collaborated with the Fusion Research Unit: Fortum Nuclear Services Ltd. (Fortum is the Finnish EFET partner), Outokumpu Oyj, Outokumpu Copper, Hollming Works Oy, Metso Powdermet Oy, Diarc Technology Inc., Creanex Oy, Hytar Oy, Advatec Oy, Delfoi Ltd., TP-Konepaja Oy, Oxford Instruments Analytical Oy, Patria Oyj, Instrumentti Mattila Oy and Elektrobit. The fusion related industrial activities were co-ordinated by Prizztech.

The contact persons and addresses of the participating research institutes and companies can be found in the Appendix.

2.4 Association Steering Committee

The research activities of the Finnish Association Euratom-Tekes are directed by the Steering Committee, which comprises the following members in 2005:

Chairman 2005	Mr. Yvan Capouet, EU Commission, Research DG
Members	Mr. Chris Ibbott, EU Commission, Research DG
	Mr. Gregor von Rintelen, EU Commission, Research DG
	Mr. Reijo Munther, Tekes
	Mr. Jouko Suokas, VTT
	Mr. Harri Tuomisto, Fortum Nuclear Services Oy
Head of Research Unit	Mr. Seppo Karttunen, VTT
Secretary	Mr. Jukka Heikkinen, VTT

The Steering Committee had one meeting in 2005, at VTT, Espoo, on 27th October. In this meeting Mr. von Rintelen was replaced by Mr. Cosyns from the Commission.

2.5 National Steering Committee

The FUSION programme national steering committee advises on the strategy and planning of the national research effort and promotes collaboration with Finnish industry. It sets also priorities for the Finnish activities in the EU Fusion Programme. The national steering committee had the following members in 2005:

Chairman	Mr. Harri Tuomisto, Fortum Nuclear Services Oy
Members	Mr. Iiro Andersson, Prizztech Oy
	Mr. Juha Lindén, Tekes
	Mr. Reijo Munther, Tekes
	Mr. Ben Karlemo, Outokumpu Copper Oy
	Mr. Rainer Salomaa, Helsinki University of Technology
	Mr. Pentti Pulkkinen, Finnish Academy
	Mr. Jouko Pullianen, Metso Powdermet Oy
	Mr. Rauno Rintamaa, VTT
	Mr. Arto Timperi, VTT
Programme Manager	Mr. Seppo Karttunen, VTT
Secretary	Mr. Tuomas Tala and Mr. Jukka Heikkinen, VTT

The FUSION national steering committee had four meetings in 2005.

2.6 The Finnish Members in the EU Fusion Committees

Consultative Committee for the Euratom Specific Research and Training Programme in the Field of Nuclear Energy – Fusion (CCE-FU)

Reijo Munther, Tekes
Seppo Karttunen, VTT

Fusion Industry Committee (CFI)

Juho Mäkinen, Outokumpu Oyj

EFDA Steering Committee

Reijo Munther, Tekes
Seppo Karttunen, VTT

Euratom Science and Technology Committee (STC)

Rainer Salomaa, TKK

Administration and Financing Advisory Committee (AFAC)

Juha Lindén, Tekes
Rainer Salomaa, TKK

Science and Technology Advisory Committee (STAC)

Seppo Karttunen, VTT
Rauno Rintamaa, VTT
Rainer Salomaa, TKK

EFDA sub-Committee on Public Information (CPI)

Seppo Karttunen, VTT (CPI Chairman)

Finnish representatives in the following fusion committees and expert groups:

Reijo Munther is a member of the IEA Fusion Power Co-ordinating Committee (FPCC).
Rainer Salomaa is a member of the Programme Committee of the ASDEX-Upgrade, Max Planck Gesellschaft.
Jukka Heikkinen is a member of the Co-ordinating Committee for Fast Waves (CCFW).
Seppo Karttunen is a member of the Co-ordinating Committee for Lower Hybrid Waves (CCLH).
Rainer Salomaa was a member of the ad-hoc-group monitoring the heating and current drive systems for JET enhancements and ITER
Seppo Karttunen was a member of the ad-hoc-group for new EFDA and Contract of Association.
Rainer Salomaa is the Tekes administrative contact person in EFDA JET matters.
Seppo Tähtinen is a Materials Liaison Officer in the European Blanket Project (EBP).
Harri Tuomisto is a member of the International Organising Committee, of the Symposium on Fusion Technology (SOFT).
Jukka Heikkinen is a member of the Scientific Committee of the European Fusion Theory Conference.
Leena Jylhä is Industry Liaison Officer for Fusion-Industry matters.

2.7 Public Information and Media

Mass media – newspapers and television: Fusion and ITER received a lot of interest in Finnish mass media, mostly treated in a positively manner. The press conference in connection to the opening of the Divertor Test Platform (DTP2) hosted by VTT and Tampere University of Technology was very successful and generated newspaper articles and was well reported in the Finnish national TV channels including interview of Finnish and EFDA experts. The ITER site agreement in June attracted a great deal of interest in the main media (national TV channels and leading newspapers). The message from the media was mainly positive to fusion. National public TV channel YLE TV 1 presented a half hour programme on fusion energy research in their science series “Tutkittu Juttu (Research Case)”. The programme covers the recent fusion research activities and includes an interview of Sir Chis Llewellyn Smith and Finnish fusion experts. Video clips taken from the “Starmakers” movie are used in the program. Fusion was covered in a more general energy series including comments by the EFDA Associate Leader for JET Jerome Pamela.

Conferences and seminars: Public information actions included participation in the Public Information Group meeting in Cadarache in March 2005 and the chairmanship of the EFDA sub-Committee on Public Information Committee (CPI) which had two meetings in 2005 both at Cadarache.

The Annual Fusion Seminar of the Association Euratom-Tekes was held in Helsinki on 30–31 May 2005. The Seminar provided a summary of the research activities of the Finnish Fusion Research Unit and industrial R&D projects. The invited speakers from the EFDA CSU Garching were Alan Peacock, who gave a review of fusion materials research and Lawrence Jones presenting a summary on ITER vacuum vessel manufacturing.

The summary of the work carried out by the Finnish Research Unit is collected to the FUSION Yearbook 2003 – Annual Report of the Association Euratom-Tekes which was published prior the Annual Seminar.

Arto Timperi gave a talk on ITER remote handling and DTP2 in the big Power-Gen Conference in Milan, June 2005. This gave a good reason to bring an EFDA stand to the exhibition in the Power-Gen Conference. Rainer Salomaa gave a talk “Fusion – a solution for future energy problems?” in Studia Generalia Technologica in Espoo, in October 2005. Seppo Karttunen gave an invited talk “Fusion Energy and New Materials” in the Millennium New Materials Seminar in Mikkeli, in September.

A “Quick-Screen” poster on the Finnish fusion programme was prepared and shown at the exhibition at the “ITER – Opportunity for European Industry” Workshop in Barcelona, December 2005. Tekes Association with Finnish companies had a stand in the Barcelona Workshop.

Newsletters: Finnish FUSION Newsletter appeared three times in 2005 telling the main news and stories on national, EU and ITER research activities as well as the important political news related fusion research and ITER negotiations. EFDA Newsletter and JET Bulletin are distributed electronically to main target groups

Fusion courses: Lecture course on Fusion Physics and Technology was given at Helsinki University of Technology. Fusion was present on the open doors day at the University of Helsinki in celebration of Year of Physics 2005.

Brochures and www-pages: The new Expo booklet and an ITER brochure prepared by the Commission were translated in Finnish. Technology cards on remote handling and virtual modelling (in Finnish and in English) and on plasma-wall phenomena and plasma facing coatings to complement the general brochure on the FUSION technology programme were prepared. The target group for those is the Finnish and European industry. The EFDA and Commission brochures have been very welcome and they are widely distributed.

The www-pages of the Finnish fusion research activities can be found from the two web-addresses:

http://www.vtt.fi/pro/research/fusion2003_06/indexe.htm
http://akseli.tekes.fi/Resource.phx/enyr/fusion/en/index.htx

2.8 Funding and Research Volume 2005

In 2005, the expenditure of the Association Euratom-Tekes was about 3.8 Mio Euro including Staff Mobility actions. The major part of the national funding comes from Tekes. The rest of the national funding comes from other national institutions, such as the Finnish Academy, research institutes participating in the fusion research (VTT, TKK, TUT, UH, LUT) and industry. The funding was allocated as following: fusion plasma physics 32% including EFDA JET activities, Underlying Technology 8%, EFDA Technology Tasks (Art. 5.1a) 43%, EFDA Art. 5.1b Contracts 9% and EFDA CSU Secondments 5%. The hot cell work, capital investments and co-operation with other Associations under the Preferential Support exceeded k€ 780 and the expenditure on the Staff Mobility actions were k€ 49. The total volume of the 2005 activities was about 38 professional man-years.

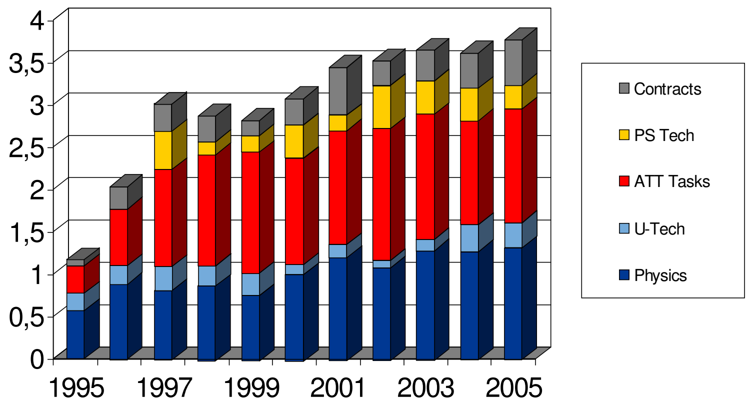

Figure 2.1: Breakdown of the expenditures of the Association Euratom-Tekes in 1995-2005. Numbers are based on Annual Accounts (1995-2005). Categories are from bottom: Physics Programme, Underlying Technology, EFDA Technology Tasks, Preferential Support items and EFDA Contracts/Orders.

3 PHYSICS PROGRAMME – FUSION PHYSICS

Research activities in fusion physics and plasma engineering are carried out in two areas: 1) fusion plasma physics and 2) plasma-wall interactions. The fusion physics work will be performed in very close co-operation between VTT Processes and Helsinki University of Technology. The main emphasis is in the participation in the JET Workprogramme 2005 and in the AUG programme at IPP Garching in collaboration with other Euratom Associations. In addition, some collaboration with CEA Cadarache and ENEA Frascati will continue.

The fusion plasma simulation groups at VTT and HUT maintain and provide an important modelling and support centre in fusion physics and plasma engineering for the European fusion programme and ITER. Simulation, design support and expertise in a fusion spin-off technology, material plasma processing in collaboration with Finnish industry will continue and will be strengthened.

Institute:	**VTT Processes** (VTT PRO)
Research scientists:	Dr. Seppo Karttunen (Head), Dr. Jukka Heikkinen (Project Manager, Dr. Jari Likonen (Project Manager), MSc. Johnny Lönnroth (seconded to UKAEA JOC), Dr. Karin Rantamäki, MSc. Tommi Renvall, Dr. Tuomas Tala (Deputy TFL T at JET) and Dr. Elizaveta Vainonen-Ahlgren Student: Markus Nora
Institute:	**Helsinki University of Technology** (HUT TF) Department of Engineering Physics and Mathematics, Laboratory of Advanced Energy Systems
Research scientists:	Prof. Rainer Salomaa (Head), Dr. Pertti Aarnio, MSc Markus Airila, Prof. Olgierd Dumbrajs, MSc. Svante Henriksson, MSc. Ville Hynönen, MSc. Salomon Janhunen, Dr. Timo Kiviniemi, Dr. Taina Kurki-Suonio, Dr. Mervi Mantsinen, PhD Francisco Ogando (Euratom Fellow, seconded from UNED), MSc. Antti Salmi, Dr. Marko Santala (seconded to UKAEA JOC), Dr. Seppo Sipilä, Students: Leena Aho-Mantila, Otto Asunta, Timo Ikonen, Simppa Jämsä, Pia Kåll, Tommi Kokki, Matti Kortelainen, Ville Tulkki
Companies:	**Fortum, Diarc Technology, Oxford Instruments Analytical**

3.1 EFDA JET Work Programme 2005

In the proposed Association Proposed Programme 2005 participation in JET activities is a key area within the physics activities. The following proposals for work in Campaigns for 2005 have been included:

1. Predictive transport modelling of advanced tokamak scenarios with ITBs (T)
2. Implement and use of real-time control algorithms in JETTO transport code (T)
3. Integrated predictive transport modelling of ELMy H-mode JET plasmas (T)
4. ICRH experiments and modelling (H)
5. LHCD experiments and modelling – long distance coupling, fast electron generation and hot spot formation and current profile optimisation (H, S2)

Notification work is allocated to perform the background investigations and the final analysis and publication of the experimental results. The emphasis in the notification work is in the ITB modelling, the analysis of LHCD and ICRH experiments.

Dr. Tuomas Tala is the Deputy Task Force Leader of TF-Transport (T).

3.1.1 Predictive transport modelling of the effect of ripple-induced thermal ion transport on H-mode performance

It has previously been shown that differences in the MHD stability of the pedestal cannot explain the differences in pedestal performance and ELM frequency between JET and JT-60U in a recent series of dimensionless pedestal identity experiments at the two machines. The effects of toroidal rotation and the explicit dependence on the aspect ratio were included in this analysis, since the two tokamaks feature rather different rotation profiles and there is a 10% difference in the aspect ratio. Given this result, attention has recently shifted to exploring the effects of ripple losses, the rationale for this being that the JT-60U plasmas are characterised by stronger toroidal magnetic field ripple than the corresponding JET identity plasmas. Losses of thermal ions have been identified as potentially important in H-mode plasmas, because it has been realised that ripple-induced ion thermal transport can be significant in comparison with the residual level of transport in the pedestal.

The influence of ripple losses on ion thermal transport has been studied in orbit-following simulations. These simulations have shown that ripple-induced transport can indeed considerably exceed the level of ion neo-classical transport and thereby affect the physics of the H-mode pedestal. Depending on the collisionality, both diffusive and convective (direct) losses can be significant. Diffusive losses were found to lead to a wide distribution of enhanced ion thermal transport, comparable in magnitude to or larger than the level of ion neo-classical transport, extending from the separatrix well beyond the top of the pedestal. Convective losses, on the other hand, were found to be very edge-localised.

The results of the orbit-following calculations have been applied in predictive transport simulations of ELMy H-mode plasmas. These simulations indicated that toroidal magnetic field ripple can influence the ELM behaviour and plasma performance very sensitively. In the case of convective losses, the so-called τ approximation was used, in which a convective energy sink term is included into the continuity equation for the ion pressure. In these simulations, a deterioration of plasma confinement as well as an increase in the ELM frequency were observed for a level of losses consistent with the orbit-following simulations. This result suggests that convective losses of thermal ions might play an important role in explaining the modest pedestal performance and high ELM frequency characterising many JT-60U plasmas. A similar result was also obtained by assuming a narrow edge-localised distribution of ion thermal transport induced by diffusive losses. However, all the orbit-following simulations so far indicate that diffusive losses tend to have a rather broad radial distribution extending well beyond the top of the pedestal. In predictive transport simulations with additional ion thermal transport matching such a distribution of diffusive losses, an improvement in confinement and a reduction of the ELM frequency were observed. This suggests that toroidal magnetic field ripple could become an important tool for ELM mitigation. The result also resembles the improved performance observed with the introduction of a stochastic magnetic boundary layer in DIII-D plasmas and might provide an explanation for it. A somewhat similar improvement of performance and reduction of the ELM frequency were observed at the start of the H-mode phase, when the ripple amplitude was increased only slightly, in a series of experiments with enhanced toroidal magnetic field ripple at JET in 1995. A further increase in the ripple amplitude led to a deterioration of confinement and an increase in the ELM frequency, followed by a back H-L transition, which could suggest an interplay between the mechanisms described here.

Above all, the study shows that ripple losses of thermal ions seem to be a highly important effect influencing the performance of tokamak plasmas in different and often counter-intuitive ways. These results may have profound implications for the design of future tokamaks and in particular for ITER, which is planned to operate with an intermediate level of toroidal magnetic field ripple and for which large divertor heat loads are a concern. A particularly interesting result is that ripple losses need not necessarily have a detrimental influence on plasma performance. On the contrary, better overall confinement than in the absence of ripple can probably be obtained by carefully choosing a suitable ripple amplitude profile.

3.1.2 Integrated predictive transport modelling of ELM heat pulse propagation

An ELM occurring at the outer midplane of a tokamak results in first an electron heat pulse and later an ion heat pulse arriving first at the outer target and then at the inner target. Modelling of the propagation towards the outer and inner targets of a heat pulse due to an ELM localised at the outer midplane has been undertaken with the integrated predictive transport code COCONUT. This code is a coupling of the 1D core transport code JETTO and the 2D edge transport code EDGE2D / NIMBUS, a set-up that makes it possible to self-consistently model the entire plasma.

It has previously been found that the heat fluxes measured at the targets and at the wall depend very sensitively on the assumptions of both perpendicular and longitudinal transport in the SOL during and after the ELM and that the assumption of large radial transport in the outer part of the SOL during the ELM leads to an enormous heat flux to the wall. Recent modelling indicates that because of the strong sensitivity of the heat fluxes on the parallel heat transmission coefficients, the fluid approach assuming temporally and spatially constant transmission factors is insufficient during the transient and has to be complemented by a kinetic approach. For this reason, the parallel heat flux limiting factors and sheath heat transmission coefficients have been determined in particle-in-cell simulations for some transient scenarios. It has been shown that these parameters vary strongly as a function of time during the transient. A parameterisation of the parallel heat flux limiting factors and sheath heat transmission coefficients has been done based on the kinetic results. The parameterisation is now being implemented for use in the integrated transport simulations.

3.1.3 Predictive transport modelling of edge ergodisation at DIII-D

In a recent series of experiments at DIII-D, the influence of edge ergodisation by a stochastic magnetic field on H-mode plasma performance and ELMs has been studied. The resonant magnetic perturbation in these experiments is provided by a set of external I-coils. A reduction of the ELM frequency or even a complete suppression of the ELMs as well as a slight improvement in pedestal performance have been observed in the presence of the stochastic field [T. E. Evans *et al.*, Phys. Rev. Lett. **92** 235003-1 (2004)]. Some of the results are unexpected and even counter-intuitive, such as the fact that the density drops and the temperature does not drop after the application of an even parity resonant magnetic perturbation.

In collaboration between JET and General Atomics, the DIII-D results have been explored in predictive transport modelling with the 1.5D JETTO transport code. It was realised that the ergodic transport provided by the resonant magnetic perturbation depends on an expression involving the magnetic diffusion coefficient, the thermal velocity, the collisionality and the parallel reconnection length and is strongly influenced by screening e.g. due to plasma rotation. The expression for ergodic transport was implemented in the JETTO transport code together with an exponential scaling expression thought to represent the screening. The ELMs were modelled using various ballooning models. It was demonstrated that introducing additional transport due to a resonant magnetic perturbation leads to a decrease in the ELM frequency and sometimes to a complete suppression of the ELMs, as seen in the experiments at DIII-D. It was demonstrated that the reduction of the ELM frequency in the presence of ergodic transport can be attributed to the specific evolution of the pressure gradient due to the perturbation. Without ergodic transport at the edge, the ELMs are typically triggered by a violation of the critical pressure gradient close to the separatrix, whereas in the presence of ergodic transport, the pressure gradient cannot build up in this region. Instead, the plasma evolves further, until the critical value of the pressure gradient is reached near the top of the pedestal. This reduces the ELM frequency and sometimes leads to complete ELM suppression.

It was found that the electron temperature does not drop with the application of an even parity resonant magnetic perturbation, because the ELMs play a stabilising role in energy transport by causing the time-averaged transport within the pedestal to increase with increasing heating power in such a way that the pressure gradient at the top of the pedestal remains roughly constant. Time-averaged transport, however does not increase with the application of a resonant magnetic perturbation, because the perturbation removes the ELMs altogether. For the decrease in density with the application of ergodic transport, no conclusive explanation has yet been found. It has been speculated that the disparity in temperature and density evolution could have to do with a movement of the strike point during the ELMs, which could significantly increase particle recycling or with a failure of the cryo-pump to cope with the large particle fluxes during the ELMs.

It should be noted that some of the modelling results resemble those obtained in work on the modelling of toroidal magnetic field ripple, which can be seen as a complementary method of ELM control. The similarities suggest that there could be a physics link between ripple transport and edge ergodisation.

3.1.4 Predictive transport modelling transient transport experiments

Rapid increase of the heat flux with respect to the inverse temperature gradient length above some critical threshold is a phenomenon observed in several tokamaks. This so called profile stiffness or resilience strives to keep the profile in marginally stable area. So far, there exists only a few descriptions of this phenomenon, mainly based on a hypothesis of an abrupt onset of strong turbulent transport, but the full picture is unknown. The implications are degradation of confinement and fusion performance, issues that are critical for the next generation tokamaks like ITER. Therefore, it is important to determine which factors affect on stiffness and how the detrimental effects can be circumvented.

Since the propagation of a temperature perturbation depends on the transport properties, and consequently the stiffness, of the plasma, transient transport studies offer an applicable method for studying profile stiffness. In this work the semi-empirical Bohm/gyroBohm and the theory based Weiland transport models, implemented in the JETTO transport code, were analysed in order to see how well they are capable of reproducing experimental stiffness levels and if they would reveal some new information about which plasma parameters the stiffness depends on. Two JET L-mode discharges with different T_e/T_i ratio and stiffness levels were used as the basis of the simulations. Modulated heating power and cold pulses induced by laser ablation or pellet injection had been applied in both of them.

The analysis of the Weiland model was emphasised, since it is, for a theory based quasi-linear fluid description of turbulence, computationally relatively light and found to give results reasonably consistent with experimental ones. The steep amplitude and phase curves, given by the Fourier analysis of the modulated electron temperature, showed that the model underestimates perturbative transport by a factor of 1.5-2. Rigorous analysis of the heat fluxes, inverse temperature scale lengths and instability growth rates led to a similar conclusion. However the stiffness level was observed to change between the two discharges even though the predicted temperature ratios were almost the same. This indicated that the stiffness might not be as strongly dependent on the temperature ratio as previously believed and led to a new idea of R/L_{Ti} dependence, which was also found to be supported by the experimental results. Next steps will be the search of theoretical explanation for the observation and finding if it can be utilised.

3.1.5 Predictive transport simulations of hybrid scenario plasmas

The simulations of the hybrid scenario discharges with the Weiland and GLF23 transport models showed limited agreement on a scan over several JET hybrid scenario discharges with varying density. For most of the discharges, either electron or ion temperature was well reproduced (roughly within measurement accuracy) while the other heat transport channel showed only limited agreement with the experiment. No systematic reason for this has been found so far. Moreover, the simulations have indicated some difficulties in predicting the experimental q-profile. JETTO transport code coupled with neo-classical transport code NCLASS predicts often central q below one while the q-profile in hybrid scenario experiments stay above one. This is most probably due to the lack of MHD phenomena, such as fishbones and NTMs redistributing the current in the JETTO transport code. However, the predicted radial region with $q<1$ is less than $r/a<0.2$. Therefore, it does not cause any significant problems in the interpretation of the simulation results, as the very central region is outside the so-called 'gradient' region ($0.2<r/a<0.8$) that is the region of most interest and importance from the transport point of view. The predictions of the hybrid scenario are similar to those in the ELMy H-mode scenarios and as a consequence, the transport models predict the experimentally observed fact that the core confinement is similar in hybrid and ELMy H-mode plasmas.

In addition to predicting the capabilities of the theory-based transport models in reproducing the experimental results, an extensive benchmark activity has been carried out using the JET hybrid scenario discharges. As the first step, JETTO and CRONOS transport codes have been benchmarked against each other with a simple case and secondly, the use of the GLF23 transport model in each code has been benchmarked. The results of the benchmark studies verified that when using the same input (initial and boundary conditions, power deposition profiles etc.) for the two transport codes, very similar predictions for the temperatures were achieved. Furthermore, the predictions for the temperature profiles and diffusion coefficients using the GLF23 transport model in both transport codes yielded very similar results, verifying their correct implementation in both of the codes.

3.1.6 Use of real-time control techniques to control *q* and the strength and location of the ITB in predictive transport simulations

An important experimental programme is in progress on JET to investigate plasma control schemes which could eventually enable ITER to sustain steady-state burning plasmas in an "advanced tokamak" operation scenario. The triggering and subsequent controllability of ITBs are major issues for fulfilling this goal, and their study is therefore an essential part of this programme. Recently, a multi-variable model-based technique was developed for the simultaneous control of the current, temperature and/or pressure profiles in JET ITB discharges, using lower hybrid current drive (LHCD) together with NBI and ICRH. The Real-Time Control (RTC) algorithms have been implemented in the JET control system, allowing the use of three actuators that are the power levels of NBI, ICRH and LHCD systems. Identical algorithms to those used in the experiments have been also implemented in the JETTO transport code.

In the closed-loop simulations with the real-time control, RTC can be applied as long as desired in the JETTO code, typically for more than one resistive current diffusion time. The power levels vary in the closed-loop simulations, as requested by the controller. The whole procedure of carrying out the open-loop power step-up simulations, determination of the controller matrix from the open-loop simulations and finally performing the closed-loop simulations with RTC is identical to that performed in JET experiments when applying the RTC technique. The benefits in the transport simulations with respect to experiments are that transport simulations are free from unpredictable events, such as MHD events, diagnostics problems or power systems failures occurring often in the experiments and 'polluting' the data. In addition, RTC techniques can be tested for several current diffusion times which is impossible with the present capabilities of JET heating systems. Therefore, the simulations serve as a simplified platform to test, validate and develop the real-time control algorithms techniques, with increasing degrees of complexity and completeness.

The earlier closed-loop simulation results with real-time control of q and ρ_{Te}^* confirmed the promising experimental results which demonstrated the successful RTC of those two quantities simultaneously on JET. And very importantly, the successful control could be continued for several resistive current diffusion times. Therefore, within the limits of the present transport model, the simulation with the reversed q-profile and strong ITB as the set-point ρ_{Te}^*-profiles showed a way to achieve strong ITBs in a controllable way, desirable and necessary for high fusion performance in long pulse experiments.

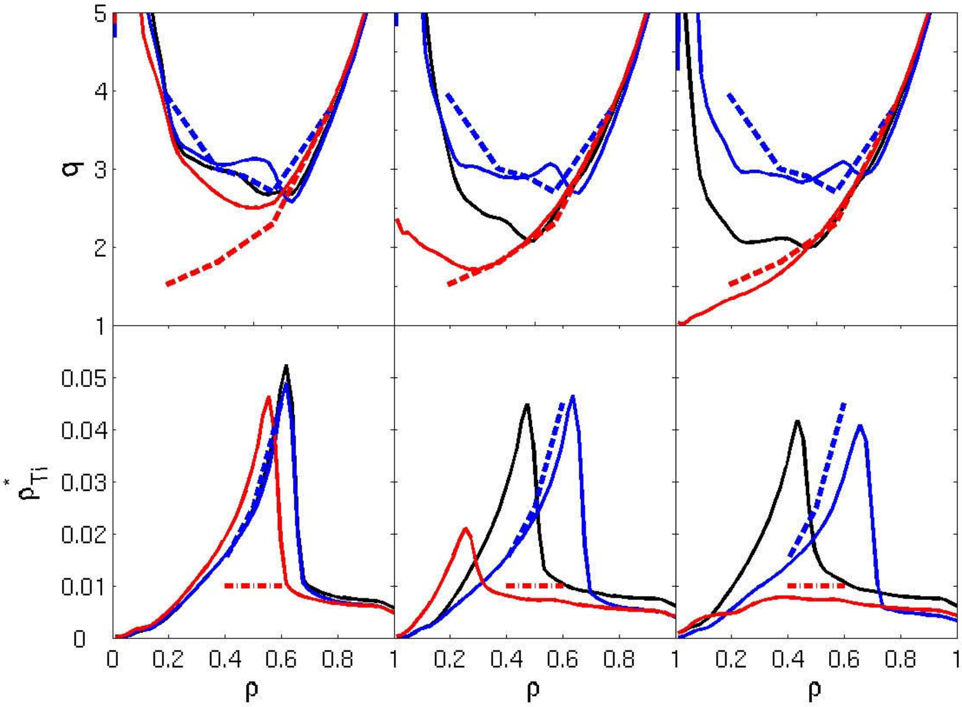

Figure 3.1: Upper frames: Predicted q-profiles for the two closed-loop simulations (red) and (blue) together with their set-point profiles (dashed red) and (dashed blue) at instants t=10s, t=20s and t=30s. Lower frames: Predicted ρ_{Ti}^* for two the closed-loop simulations (red) and (blue) together with its set-point profiles (dashed red) and (dashed blue) at instants t=10s, t=20s and t=30s. The black curves show the open-loop reference simulation without RTC at three instants in both frames.

Recently, a similar control algorithm for the strength and location of the ion ITB (ρ_{Ti}^*) was implemented in JETTO. The closed-loop JETTO simulations with the simultaneous RTC of q and ρ_{Ti}^* showed that it is significantly more challenging to control ρ_{Ti}^* than ρ_{Te}^*. This is illustrated in Figure 3.1. In the case with no ITB and monotonic q-profile as the set-point profiles (dashed red curves), the closed- loop simulation (solid red curves) approach well to the set-point profiles. In the case with strong ITB and reversed q-profile as the set-point profiles (dashed blue curves), the set-point q-profile is well achieved in the closed-loop simulation (solid blue curves), but the predicted ρ_{Ti}^{*} indicates that the controller finds an ITB located at too large radius. Thus, RTC of ρ_{Ti}^{*} is more difficult to achieve than that of ρ_{Te}^{*}, indicating that the ion ITBs are more challenging to control than the electron ITBs. At least the following two reasons could be identified: firstly, the strength of the ion ITB (magnitude of ρ_{Ti}^{*}) is controlled by NBI and ICRH while the location of the ion ITB (radial profile of ρ_{Ti}^{*}) is mainly controlled by LHCD via the LH driven current changing the q-profile. And secondly, the values of ρ_{Ti}^{*} vary typically a factor 2-5 just outside and inside the ITB while those ones of ρ_{Te}^{*} vary only a factor of 1.5. As a consequence, a larger non-linearity in the controlled ρ_T profile is created in the ion heat transport channel. These simulations are real predictions in a sense that the RTC of ρ_{Ti}^{*} has never been done experimentally so far.

3.1.7 Predicting the ITB dynamics and the link to poloidal rotation

Recent results from the measurements of carbon plasma rotation velocity across Internal Transport Barriers (ITBs) on JET show that the velocities are typically an order of magnitude higher than the neo-classical predictions. As a consequence, the radial electric field can be very different from that calculated using the neo-classical value for v_{pol}. This gives further rise to different ω_{ExB} shearing rates, than normally used in transport simulations to predict the ITB dynamics, location and strength. The 1D first-principle transport models, such as the Weiland model or GLF23, have so far had difficulties in reproducing satisfactorily the time dynamics, location and strength of the ITBs. The Weiland model does not typically predict a clear ITB at all while GLF23 often predicts the ITB at the wrong radial location or too weak an ITB. One of the obvious reasons is that the growth rates of the ITG/TEM modes significantly exceed the ω_{ExB} shearing rates calculated from the radial electric field E_r. In the present calculation of E_r in transport codes, the neo-classical value for the poloidal rotation velocity is assumed. However, after the recent measurements of v_{pol} on JET, the question to be addressed is whether the failure to predict ITBs could actually be caused by the incorrectly estimated ω_{ExB} shearing rates, rather than the oversize growth rates.

Two predictive simulations with the Weiland transport model are compared in Figure 3.2 before the ITB formation (left frame) and after the ITB formation (right frame). The only difference between the two simulations is that the first one (red curves) uses the neo-classical poloidal velocity from NCLASS whereas the second one (blue curves) takes the experimentally measured v_{pol} in the calculation of E_r and ω_{ExB} flow shearing rate. In the case when the experimental poloidal rotation is used, the Weiland model predicts the ion ITB just at the right radial location and right instant with roughly the same ITB strength as measured in the experiments. On the other hand, otherwise an identical simulation except with v_{pol} from NCLASS does not exhibit any sign of an ITB.

If the experimental poloidal velocity is used instead of the neo-classical one to calculate the radial electric field and the $\omega_{E\times B}$ shearing rate, E_r and $\omega_{E\times B}$ are found to be even qualitatively significantly different. This is most pronounced within the ITB layer. As a consequence, the simulation predictions for the dynamics, strength and location of ITBs change and may improve significantly, as shown here in the case of the Weiland model. In addition to changing the predictive transport simulation results, the non-neo-classical anomalous poloidal rotation velocities might raise the need for further assessment of the neo-classical transport theory in the presence of turbulence. Furthermore, understanding the causality between the onset of the ITB and large v_{pol} as well as the source for this large v_{pol} remain extremely interesting future challenges.

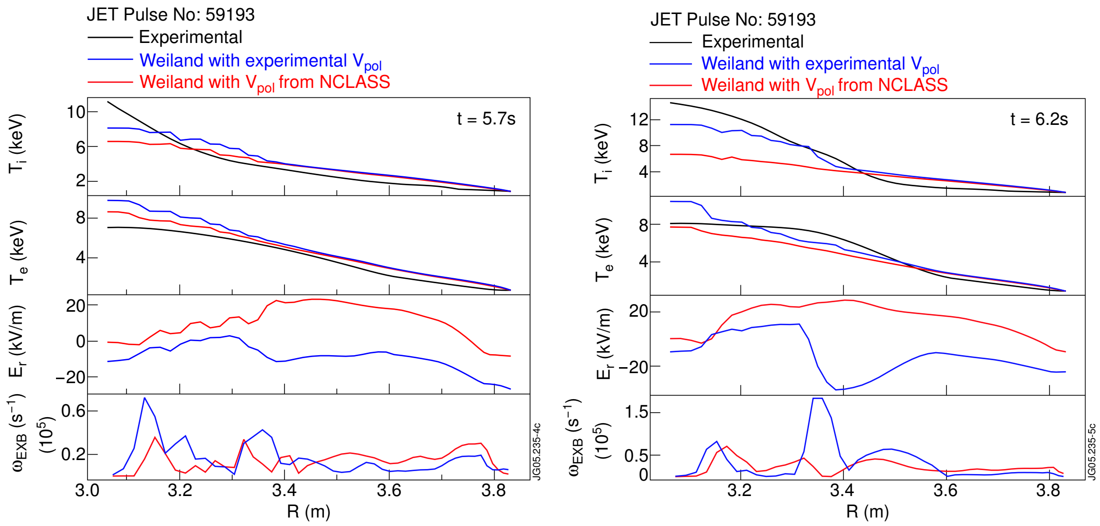

Figure 3.2: Predictions for the ion and electron temperatures, radial electric field and $\omega_{E\times B}$ shearing rate before the ion ITB formation (left frame) and after it (right frame) using the Weiland transport model.

3.1.8 Modelling of Lower Hybrid Current Drive (TF-H)

The aim of the work is to predict the LHCD efficiency, the power deposition profiles and levels in both current Advanced Tokamak scenarios and ITER - scenarios. The power deposition and its effects on the current profile and the q-profiles are important in many experiments. We participate in the work of modelling the LH power deposition by using the JETTO/FRTC code.

To investigate LH wave propagation in tokamaks, it is important to model scattering of lower hybrid waves by density fluctuations inside the plasma in addition to standard ray-tracing because LH wave propagation can be strongly affected by scattering from density fluctuations. The scattering can have a significant effect on the q-profile evolution, which is important in predictive transport modelling, especially for ITB or hybrid plasmas. In this work, a simple technique taking into account the scattering of the LH waves from the density fluctuations has been included in the fast ray-tracing code FRTC. This model couples the LH module with the Weiland transport model. The density fluctuations are obtained from the transport model and used in the LH module to calculate the scattering angle. The model can then be used in either interpretive or predictive transport simulations where LH heating is used. The first studies have been performed with the FRTC code coupled to the Weiland transport model in order to take into account the scattering within JETTO transport code.

In this first study, this model is used for two JET pulses. It is found that the inclusion of the effect of density fluctuation on the LH ray-tracing provides an important flattening influence on the driven current density profiles. This removes, in an essential way, spurious spikes in j_{LH}, which have been sometimes observed in the ray-tracing calculations, and thus changes the prediction for ITBs and temperature profile evolution in predictive transport modelling.

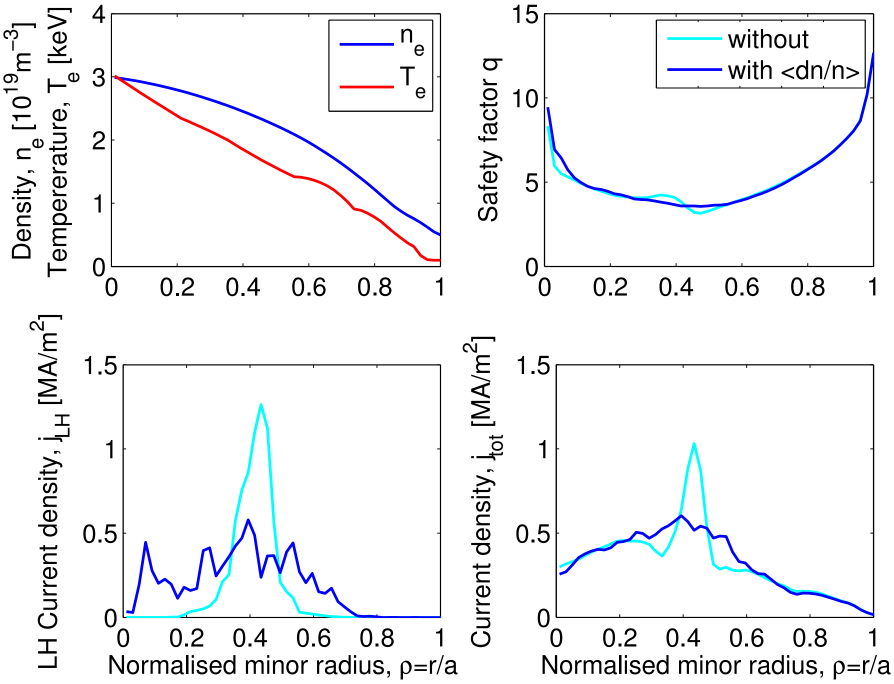

Figure 3.3: Density and temperature profiles together with the q-profile for the shots 57321 (t=13.4s). The lower frames show the LH driven current density and the total current density with and without the wave scattering from the density fluctuations.

3.1.9 JET-FT-3.24: Material transport and erosion/deposition in JET Torus

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Volume in 2005:	20 person-months

Background

A major concern with the proposed use of carbon divertor targets in ITER is retention of tritium in re-deposited films. This concern was highlighted by the deuterium-tritium experiment (DTE1) with the Mk I divertor in JET in 1997, when ~17% of the tritium fuelling was trapped in deposits at shadowed areas of the inner divertor. Another problem is that deposition in the JET divertor is highly asymmetric, with heavy deposition in the inner divertor but just small net erosion in the outer divertor, which is not understood. Post-mortem analysis of plasma-facing components (PFC) has shown that much of the carbon responsible for co-deposition in the divertor is sputtered from the walls of the main chamber, even though the primary plasma contact areas are in the divertor. Thus there is no model for erosion/deposition that fits the JET data and can thus be applied to ITER.

Under Task Force Fusion Technology (TF-FT) there has been a three-year programme to analyse JET Mk IIGB divertor tiles, wall tiles and other components removed from JET during the 2001 shutdown following the C1-4 operational campaigns. Firstly, the ion beam methods used prior to 2001 have limited depth range, so a facility has been installed in the Finnish Association (Tekes) for the cutting of samples from the Be- and T-contaminated JET components, and for analysis by secondary ion mass spectrometry (SIMS). SIMS analysis in combination with ion beam techniques has proved to be particularly powerful. Unexpectedly, the combination of analysis techniques has demonstrated that the nature of the deposition at the inner divertor changed from that previously seen at JET at some point towards the end of 1999-2001 experimental campaign. SIMS analyses have shown a double layer structure in the deposits on the inner divertor wall tiles; C-rich layer on the surface and a Be-rich layer underneath. Tiles from the inner wall of the Mk IIA divertor, and from the Mk IIGB divertor exposed 1998-1999 are covered only with the Be-rich film. Moreover, ion beam techniques and SIMS analyses have shown that the much lower Be/C ratio in the outer layer on the divertor tiles is similar to that in the deposits on inner wall guard limiter tiles (which have no double layer structure in the deposits). The reason for the change in the composition is not yet known and one possibility could be that the temperature decrease of JET main wall in December 2000 somehow “switched off” chemical erosion of carbon. The divertor configuration was changed during the 2001 shut-down to Mk II SRP, by replacing the septum with a septum replacement plate (SRP). All the 2002-4 campaigns have been at the lower vessel temperature, so it should become clear if that was the reason for the duplex film in the inner divertor

Surface analysis of JET tiles and sample handling

The batch of MkIISRP tiles delivered to VTT contained six divertor tiles. Hardware design of the SIMS instrument and the tandem accelerator used in the characterisation work require the sample size be below 20 mm in diameter. So, the CFC tiles were sampled in a glove box using a drill saw to cut cylinders of 17 mm diameter at the marker stripe. Pieces of about 10 mm high were cut from the core samples.

SIMS analysis was made with a double focussing magnetic sector instrument (VG Ionex IX-70S). A 5 keV O_2^+ primary ion beam was used. Some selected SIMS samples were measured with time-of-flight elastic recoil detection analysis (TOF-ERDA) technique to obtain elementary concentrations at the near surface region. In the measurements, the 5 MV tandem accelerator EGP-10-II with a 43 MeV beam of $^{79}Br^{8+}$ ions was used. Tiles have also been analysed by nuclear reaction analysis (NRA) and Rutherford backscattering spectrometry (RBS) methods with 2.5 MeV H^+ ions at the University of Sussex.

Erosion and deposition

The analysed MkIISRP tiles were coated with a tungsten stripe in 2001 prior to installation. General picture of deposition at the inner divertor during Mk IISRP phase is quite similar to that during the Mk IIGB divertor; deposition in the JET divertor is highly asymmetric, with heavy deposition in the inner divertor but just small net erosion in the outer divertor. The thickness of the deposit decreases from the top of tile 1 to the bottom and then increases on tile 3 reaching ~ 60 μm. There are even thicker deposits on the small sloping section of the floor that can be accessed by the plasma both at the inner and outer divertor legs (see Figure 3.4). According to optical microscopy deposit thickness on the sloping part of tile 4 is up to ~ 280 μm. Only small amounts of deposition are found elsewhere in the outer divertor.

The deposits on tiles 1 and 3 have a layer structure and high Be/C ratio in contrast to deposits on Mk IIGB tiles removed in 2001. Figure 3.5 shows SIMS depth profiles from the top of tile 1. W layer with thickness of about 3 μm is clearly visible. There is a deposit with thickness of about 30 μm. Be/C ratio varies in the layer. Near the surface Be/C ratio is lower than deeper in the deposit. Ni seems also to be associated with Be as Ni content is lower near the surface. There is some D near the surface, where Be/C level is lower, and at the interface between the W marker and the deposit.

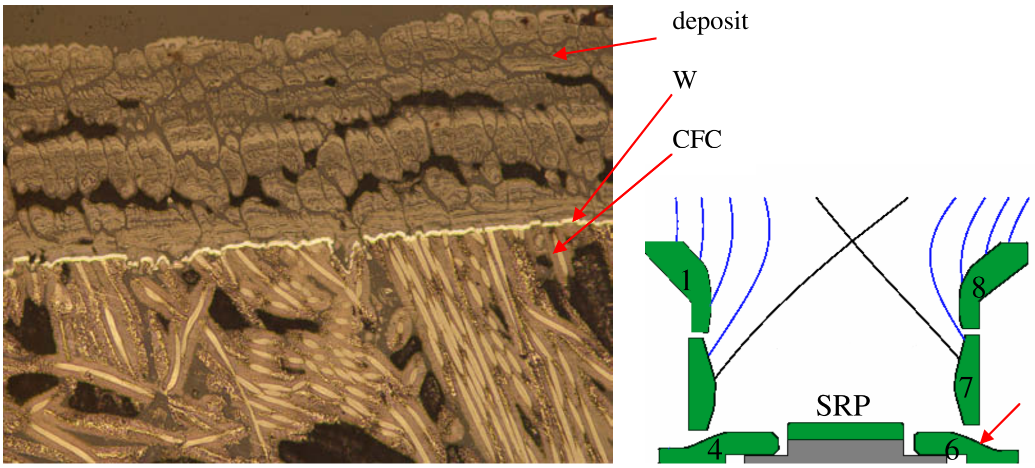


Figure 3.4: Optical microscope image from the sloping part of tile 6 (see red arrow). Thickness of the deposit is about 200μm.

It has been speculated that the duplex layers and the low Be/C ratio near the surface on Mk IIGB tiles were due to the reduction of the vessel wall temperature in 2001. JET was operated at reduced wall temperature (200°C) throughout 2001-2004 operations indicating that the temperature decrease did not explain the duplex nature. JET operated with He-fuelling for a month at the end of C4 campaign and this could be the reason for the duplex structure. The original W stripe on the inner and floor tiles has survived the 2001-2004 plasma operations intact.

W coated marker tiles installed in 2001 were removed from JET at the end of 2004. The original W stripe on the inner and floor tiles has survived the 2001-2004 plasma operations intact. The W marker coatings on the outer divertor tiles 7 and 8 seemed to have survived the experimental campaign in 2001-2004 and the W coatings were still clearly visible by eye although there are some toroidal deposition bands on tile 7 which in some places overlay the W layer. The W marker on tile 8 appeared to have survived the experimental campaign intact. Post mortem surface analyses have, however, shown that the W stripe has been eroded strongly and CFC is exposed on the strike point area on tile 7 and on the top part of tile 8.

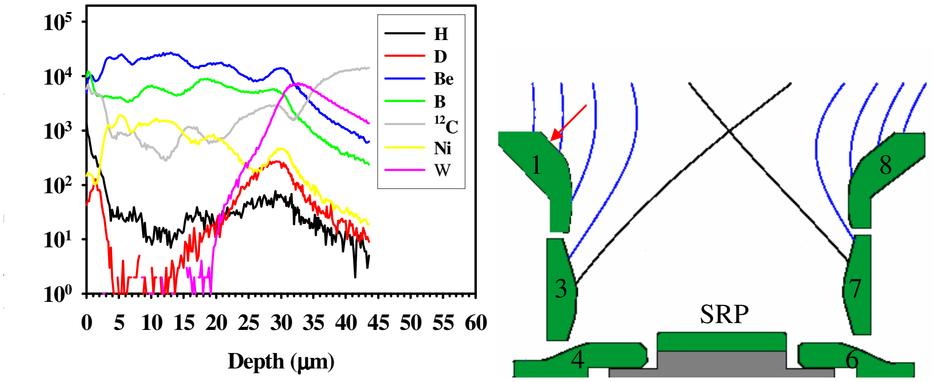


Figure 3.5: SIMS depth profiles from top of inner MkIISRP divertor wall tile 1 and MkIISRP divertor. Analysed sample marked with red arrow.

Material transport in scrape-off layer (SOL)

At the end of C14 campaign in 2004, Task Force E carried out an experiment to provide specific information on material transport and SOL flows observed at JET. $^{13}CH_4$ was puffed into JET from the outer divertor between tiles 7 and 8, and $B(CH_3)_3$ into the outer divertor floor. The largest ^{13}C amount was found on tile 7 just above the strike point and at the top of the tile and on the apron of tile 8. The deposition on tile 7 is due to local downstream impurity drag. The deposition on the apron of tile 8 could perhaps be due to upstream transport and far SOL downstream flow. The amount at the inner divertor is relatively small but there is somewhat higher level on the floor tiles. At the moment it is not known how much there is transport through the private flux region from the outer to the inner divertor. ^{13}C has been detected on the outer poloidal limiter tiles but no significant levels have been found. Overall amount detected with post-mortem analyses do not account for most of the injected ^{13}C and the location of the remaining unaccounted ^{13}C is not known.

3.1.10 The containment of and heating by fusion-born alpha particles in ITB plasmas

In fusion power reactors, the fuel plasma ought to be predominantly heated by the fusion reactions themselves, i.e. by the 3.5 MeV alpha particles born in the reactions. Therefore, these particles should be effectively confined until they have deposited most of their energy in the surrounding plasma. On the other hand, keeping the capital costs of fusion as low as possible, one ought to minimize the size of the fusion vessel because the toroidal magnetic coils surrounding it are responsible for a lion's share of the cost. Therefore, optimizing the performance of the plasma in a given device has a high priority.

Internal transport barriers (ITB) are one way of optimizing the plasma performance by allowing higher pressure for a given magnetic field. ITBs can be produced in plasmas where the current profile is modified so that the q-profile becomes reversed. In such a situation the poloidal magnetic field in the plasma centre becomes very small, which implies very wide drift orbits, especially for high energy particles. The concern thus is whether the confinement of fusion alphas becomes compromised.

The behaviour of 3.5 MeV alphas in JET-size geometry was studied using ASCOT simulations. We compared the confinement of and plasma heating by fusion-born alpha particles in a tokamak plasma with and without the ITB, i.e. advanced scenario with a reversed q-profile versus standard H-mode plasma.

The study showed that the alpha confinement is not compromised to a significant degree. The increased orbit widths are sufficiently compensated by an energy selective transport barrier formed where the stagnation orbits exist. The overall alpha heating was also found to be of the same order of magnitude for both normal and reversed q-profiles, but for the latter it was spread to a larger volume as per Figure 3.6. It should be noted that the simulations were performed with identical plasma backgrounds, yet in a realistic case the thermal particles are much better confined in the plasma with the ITB, thus increasing the number of born alpha particles.

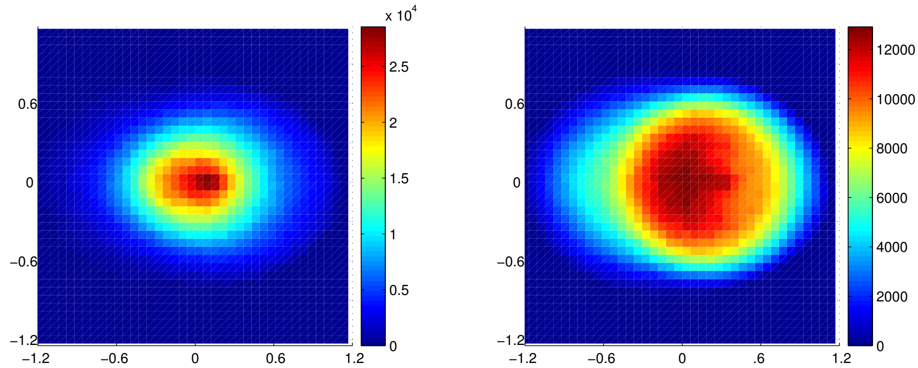


Figure 3.6: Alpha heating in the poloidal cross-section with monotonic (left) and revers ed (right) q-profiles, the scale being joules per 6x6~cm grid. The heating is more widely dispersed in the reversed q-profile case than in the H-mode case, and the poloidal heating distribution has a distinct peak at the location of the alpha particles' stagnation orbits in both cases.

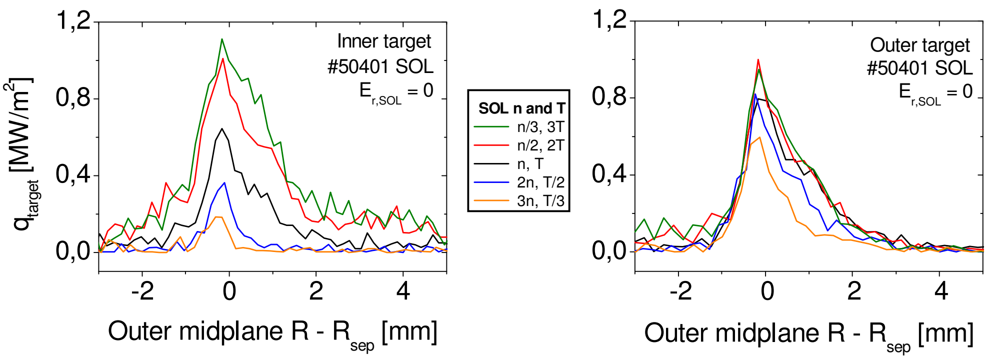


Figure 3.7: The effect of SOL/divertor collisionality on divertor target loads (ASCOT simulation).

3.1.11 Effect of SOL/divertor Collisionality on Target Loads

The ASCOT code has been applied to study the effect of scrape-off layer (SOL) and divertor collisionality on the divertor target loads in JET. The effect, shown in Figure 3.7, was found to be much more pronounced on the inner target, consistent with the longer path of the loss ions to that target (with B×∇B drift towards the X-point). Based on independent density and temperature scans, the ion-ion collisionality was shown to be the governing parameter for the peak heat load profiles, e.g. peak in-out asymmetry. The effect of charge exchange collisions was found to be relatively minor in comparison. Although the inclusion of ion-ion collisions offers improved agreement of simulation results with experiment, it can only account for at most 20% of the observed ELM-averaged peak outer target heat loads. Enhanced radial transport, either collisional or anomalous, is clearly necessary to explain the observed load magnitudes.

3.1.12 Ripple at JET

Ideally the magnetic field in a tokamak is axi-symmetric. However, due to the finite number of toroidal field generating coils the toroidal magnetic field strength alternates with the period of the coils. Toroidal field is strongest just under the coil and weakest half way between subsequent coils. The losses of energy and particles that emerge due to the finite number of coils are called ripple losses. There are two ways how ripple can induce particle losses. Firstly particles can get trapped in a local magnetic well between the coils resulting in rapid vertical losses due to grad B drift. And secondly ripple destroys the axi-symmetry and leads to stochastic radial diffusion at the banana tips thereby enhancing the loss rate.

Recently there has been a renewed interest in ripple. It is thought that by controlling the ripple amplitude one could influence the ELM frequency which is very important in fusion power plants as ELMs create high level heat pulses that can damage vessel walls quite rapidly. The physical mechanism for ELM control is such that the ripple increases transport close to the edge of the plasma and thus slows down the rate at which the plasma pressure increases to a critical value that will make the ELM unstable.

ASCOT code has been used to study this phenomenon by evaluating the transport coefficients in the presence of ripple for varying ripple amplitudes and collisionalities using the so-called pulse spreading technique. Figure 3.8 shows loss rates as a function of normalised minor radius. As expected losses tend to increase towards the edge showing approximately exponential growth. Loss rates increase as a function of ripple amplitude but clearly ripple is not dominating the losses. More work will be done in year 2006 to elaborate many more dependencies that effect the loss rates.

An NBI power deposition code PENCIL used at JET has been recently interfaced with ASCOT. This allows one to calculate ripple losses using a realistic beam line geometry and NBI power deposition at JET. There are also other European codes which can calculate ripple losses but that either do not follow particle orbits (and thus do not know where to power goes) or have a simplified geometry, ripple model or magnetic background. ASCOT can simulate a real tokamak in a realistic magnetic equilibrium and calculate the amount of heat load each vessel component receives.

These simulations will be done in preparation for the ripple campaign planned at JET during 2006. So far the initial results indicate that with a maximum ripple amplitude ripple trapped particles can cause large stress on the vessel with up to ~0.2 MW/m2 per 1 MW of NBI power. Thus it seems that injecting around 10 MW of NBI would already compromise the vessel safety and therefore limited NBI power might have to be used in the coming experiments. Figure 3.9 shows an example of a heat load on the low field side vessel wall just above the divertor resulting from a 1 MW of 120 keV NBI from octant 8 beam line #1.

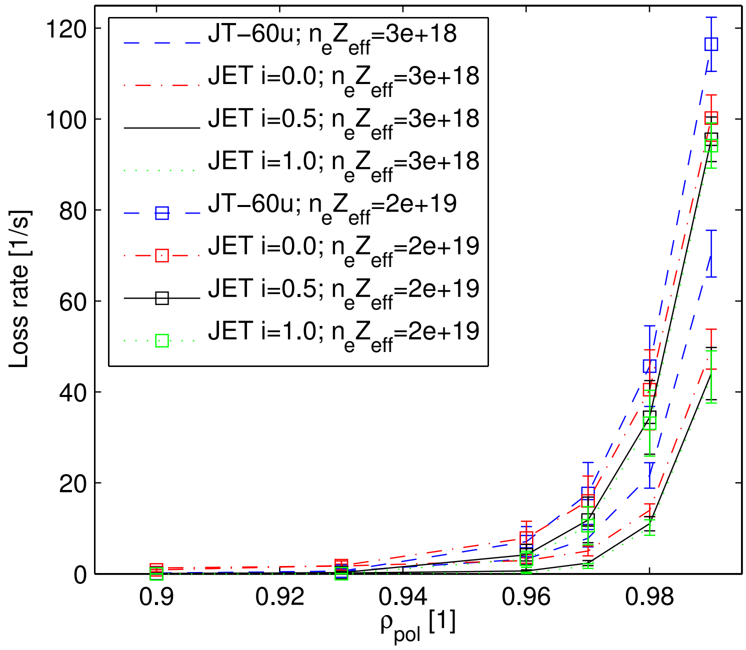


Figure 3.8: Ion loss rate in the presence of toroidal ripple as a function of normalised minor radius.

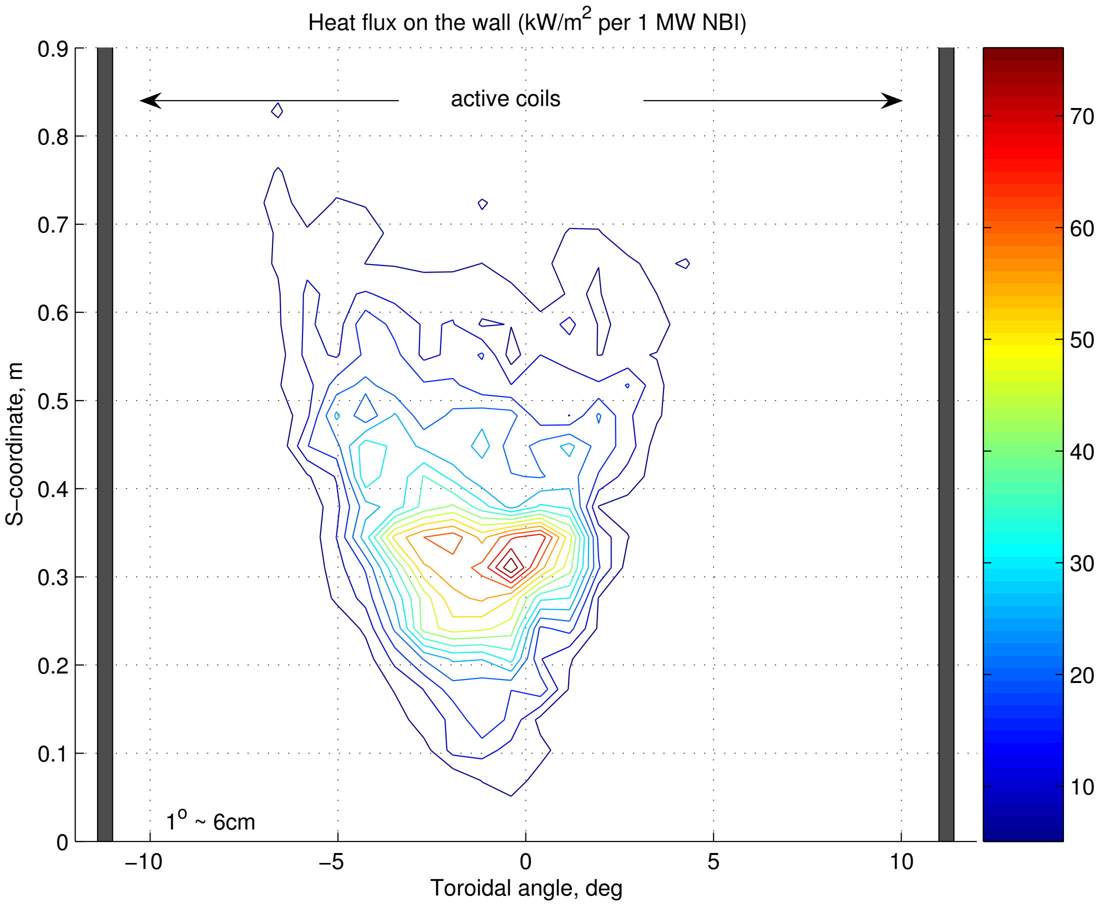


Figure 3.9: Estimated heat load from 1 MW of 120 keV NBI beam ions due to ripple trapping. S-coordinate follows the wall poloidally and \Phi is the toroidal angle.

3.1.13 Neutral Particle Analysers

There are two neutral particle analysers (NPA) at JET. The high energy NPA (GEMMA-2M, diagnostic ID: KF1) is installed on top of the JET machine and has a vertical line-of-sight. It can be configured to measure one ion species on eight energy channels with energy of 250-1600 keV for hydrogen isotopes and up to 3500 keV for He. The low energy NPA (ISEP, diagnostic ID: KR2) has a horizontal, radial line-of-sight through plasma centre. It measures simultaneously all three hydrogen isotopes on a total of 32 channels. The energy range can be configured from 5 keV to 750 keV (for H) by varying the electric and magnetic fields within the diagnostic. The diagnostic hardware as well as all data collection electronics has been supplied to JET by Ioffe Institute, St. Petersburg.

During year 2005 Tekes Association has continued to provide technical support for the JET NPAs. As there were no experiments during 2005, the support has concentrated on maintenance and commissioning of the diagnostic for experimental campaigns. After the long shutdown, the NPAs have been brought back into operation without major problems.

A major new development during 2005 has been the proposal and commencement of a project to develop thin silicon detectors for the NPAs to replace the present CsI(Tl) scintillator detectors. This is one of the JET EP2 diagnostic upgrade projects. Tekes is the leading association in this project and the collaboration involves Helsinki Institute of Physics, VTT Microelectronics, Helsinki University of Technology and Ioffe Institute.

The first phase of the project has started with numeric modelling of the new detectors. The aim is to develop detectors using Silicon-on-Insulator (SOI) technology. Thickness of the detectors should not exceed by much the range of the ions to be detected to achieve best possible background rejection. For high-energy NPA, the needed thickness is about 5-25 μm, and even thinner for the low energy NPA. After modelling, prototypes are to be manufactured and tested. If performance is sufficiently good, the diagnostic upgrade will be designed and built to be installed in JET during the 2008 shutdown.

3.2 Participation in the IPP AUG Programme

3.2.1 Fast ion distribution in the AUG edge pedestal region

Fast ion effects on the edge region and material surfaces: The effect of a radial electric field is generally considered to arise only from its inhomogeneity, but even a constant electric field can have a net effect on NBI-born ions. Another important factor in the behaviour of fast ions is the finite toroidal ripple. Using the three-dimensional ripple model recently implemented in ASCOT, it is possible to study both effects separately or simultaneously.

Both counter- and co-injected neutral beams have been investigated in a series of eight simulations for ASDEX Upgrade Quiescent H-mode (QH-mode) discharge 17695 at t = 5.6 s. The magnetic and plasma background for the counter-injection simulation came directly from the actual discharge. The corresponding co-injection simulation background data were obtained by reversing the current, the toroidal magnetic field and the pitch ($v_{\parallel}/v$) of the particles, thus creating a virtual co-injection discharge which otherwise is identical to the actual counter-injection discharge. Four simulations were carried out for both co- and counter-injection case:

a) No magnetic ripple, no radial electric field
b) No magnetic ripple, typical QH-mode radial electric field
c) Finite toroidal ripple, no radial electric field
d) Both finite ripple and typical QH-mode radial electric field

There is a radical difference in the wall load between co- and counter-injection simulations. In the former case (H-mode), the wall load is negligible for exisymmetric situation, but with counter-injection (QH-mode) there is significant load which more than doubles when ripple is switched on. However, at the same time the divertor load is decreased, so that, in effect, the ripple moves particle load from the divertor to the wall. Even with the ripple, the co-injected ions contribute only a load comparable to shine-through. For the heat load the results are similar, except that particles hitting the wall are more energetic than those destined to the divertor. This suggests that the particles hitting the wall are mostly ripple-trapped. The particle flux results are summarized in Table 3-1.

Table 3-1: The breakdown of the particle fluence between the wall and the divertor. The fraction of the total neutral beam source rate shown in parenthesis.

	fluence/10^{19} s^{-1} (co-inj.)		fluence/10^{19} s^{-1} (counter-inj.)	
simulation	wall	divertor	wall	divertor
No ripple, no E_r	—	1.8 (2.4 %)	6.0 (8.3 %)	7.6 (10 %)
Only E_r	0.1 (0.1 %)	2.0 (2.8 %)	8.1 (11 %)	7.4 (10 %)
Only ripple	3.1 (4.3 %)	1.9 (2.6 %)	13 (18 %)	4.0 (5.5 5)
Both ripple & E_r	3.0 (4.1 %)	2.7 (3.7 %)	11 (14 %)	6.2 (8.5 %)

The effect of radial electric field on wall and divertor loads is small, but it has much more pronounced an effect on the edge parallel velocity distribution in the counter-injection simulation. Comparing Figure 3.10 and Figure 3.11 shown that, in addition to removing particles and thereby decreasing the phase space density everywhere in the (ρ, $v_{\parallel}$)-space, the radial electric field skews the distribution. In the co-injection simulation the effect was less pronounced but still observable.

Benchmarking ASCOT against experimental NPA data: In standard H-mode, the edge MHD behaviour leads to individual ELM events which can lead to unacceptable loads to plasma-facing component. In the so-called Quiescent H-mode (QH-mode) a more continuous, benign MHD behaviour called Edge Harmonic Oscillations (EHO) is observed. This MHD activity has all the desired properties: it facilitates density control and impurity exhaust while leaving the good core confinement intact. It is not clear what is the trigger for EHO, nor is it understood what suppresses the ELMs, but the fact that QH-mode is obtained only with counter-injection of neutral beams suggests that the fast ions play a role. Moreover, there is a growing consensus that fast ions contribute significantly to the edge stability in high performance discharges, but until now their contribution has not been routinely included in the stability considerations.

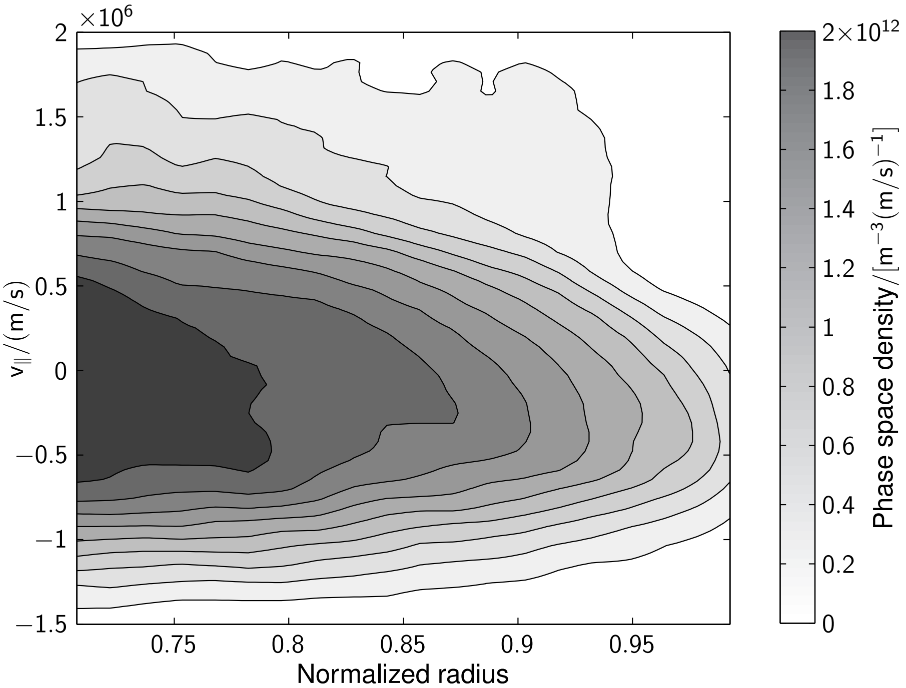


Figure 3.10: Parallel velocity distribution in the counter-injection (QH-mode) simulation with zero radial electric field and axisymmetric magnetic field.

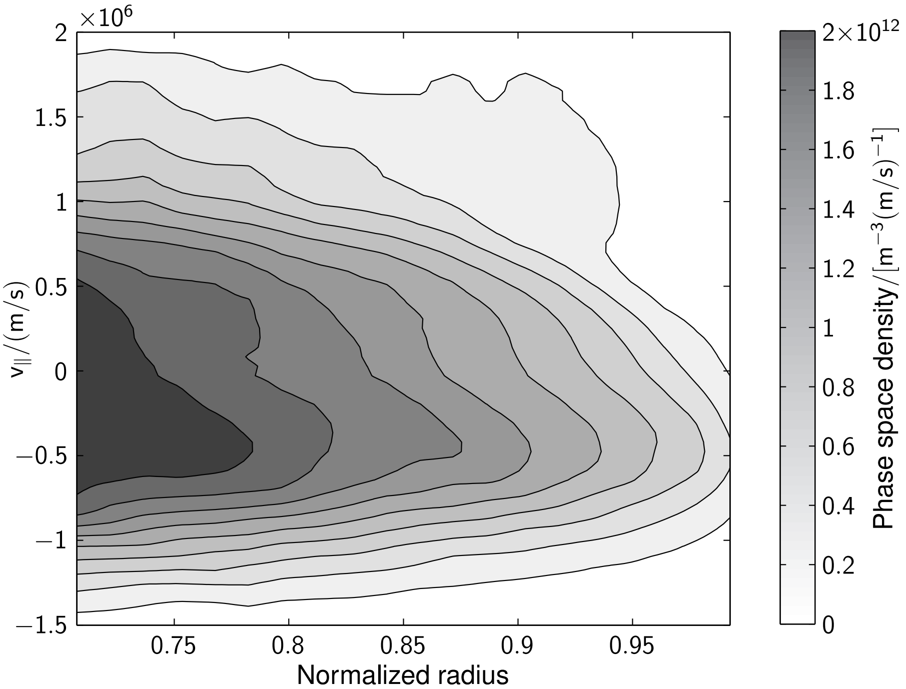


Figure 3.11: Parallel velocity distribution in the counter-injection (QH-mode) simulation with finite ripple and zero radial electric field.

ASCOT can evaluate the fast ion population in a tokamak plasma in the presence of collisions, magnetic ripple and radial electric field. But to have confidence in numerically obtained results, the ASCOT-calculated distributions have to be quantitatively benchmarked against experimentally accessible data. ASCOT includes a model for the Neutral Particle Analyzer (NPA), and in February 2005 six ASDEX

Upgrade discharges were dedicated to benchmarking ASCOT against NPA measurements. In these discharges both the beams and the detector sightlines were varied from shot to shot. Afterwards, all the different combinations of various beams and sightlines were simulated with ASCOT.

Tritium surface distribution: The surface distribution of 1 MeV tritium ions born from deuterium–deuterium reactions between neutral beam and bulk plasma was simulated. The simulation was carried out for a representative ASDEX Upgrade discharge including the effects of toroidal ripple. The first results are in relatively good agreement with the measurements. However, the measured tritium had accumulated over the whole experimental campaign. Therefore a simulation of only one discharge can not be expected to reproduce the measured profile exactly. Thus more discharges will be simulated in order to evaluate the surface distribution time-integrated over the whole experimental campaign. Using the coordinates illustrated in Figure 3.12, the simulated tritium flux onto the divertor and the wall is shown in Figure 3.13.

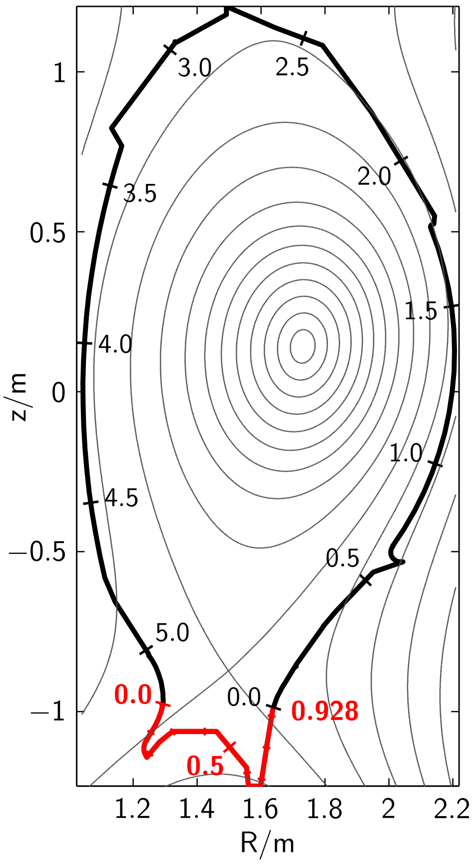


Figure 3.12: Illustration of the divertor and wall coordinates used in Figure 3.13.

3.2.2 Kinetic Electrons in the Scrape-off Layer of AUG

The 2D SOLPS fluid code package is used to model the edge plasma of ASDEX Upgrade tokamak (AUG). It provides the AUG background data for ASCOT and ERO codes. The SOL plasma has typically medium collisionality in H mode, in which case kinetic effects become considerable. One aspect related to this is the overestimation of various fluxes occurring in fluid calculations. As in all major edge plasma codes, the kinetic effects are taken into account in SOLPS by introducing flux limits that keep the flux values within realistic bounds. The evaluation of these limits can be based on theoretical calculations and experimental results, but sometimes one has to rely simply on trial-and-error method.

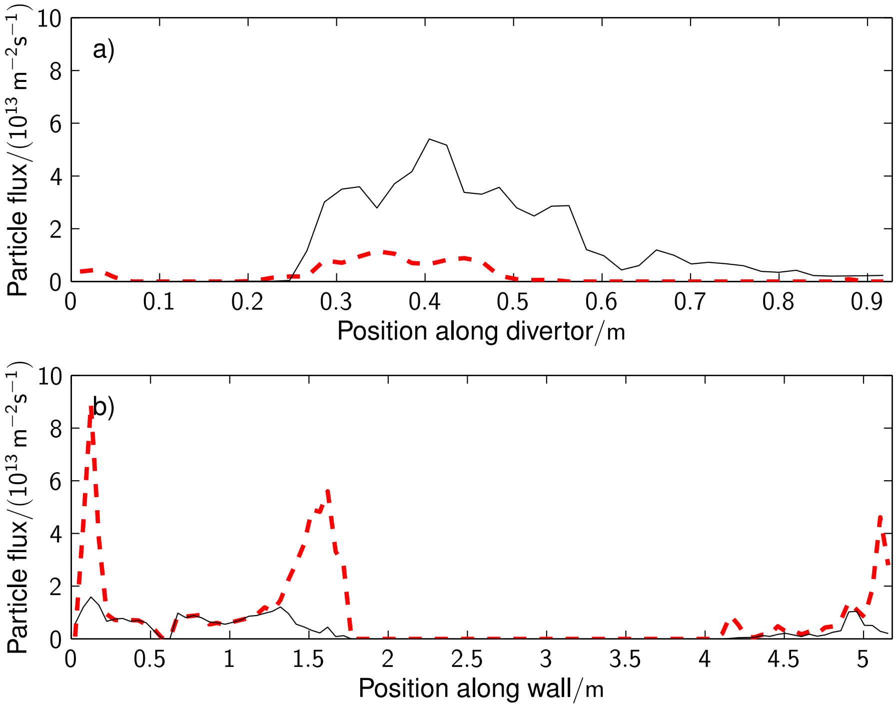


Figure 3.13: The simulated tritium flux onto the divertor and the wall with (red dashed line) and without ripple (black solid line). The divertor and wall coordinates are shown in Figure 3.12.

The flux limits are very likely an insufficient way to account for all kinetic effects. An example of this are the kinetic electrons, which are fast electrons whose interaction with the background particles via collisions is negligible. Kinetic electrons are most likely to reach the SOL at the midplane in the low field side, where the flux surfaces are very close together and the hot electrons can easily diffuse from the inner flux surfaces with high T_e to the flux surfaces outside the separatrix with low T_e. Kinetic electrons have a strong influence on the formation of the Debye sheath next to the divertor plates and in heat conduction to material surfaces. These aspects can not be accommodated by the proper evaluation of flux limits, and so it is of crucial importance to know how significant the contribution of fast electrons is.

The study of kinetic electrons with the single particle code ASCOT was initialized in summer 2005. It began with the evaluation of an interpolation program that adapts the SOLPS background to the ASCOT grid. By the end of the year 2005, the first results showing strongly non-Maxwellian electron energy distributions near the divertor were obtained. However, electron simulations with non-analytic magnetic background were found to be very susceptible for numerical errors. The work continues with adjustments to the simulation calculation parameters.

3.2.3 Kinetic ions and divertor load asymmetries in AUG

In high performance tokamaks the power load to the divertor plates can get very high. In the present machines the parallel heat flux is in the order of few hundred MW/m^2 and in ITER it will be a few GW/m^2. Technologically acceptable limit for average power load would be approximately 10 MW/m^2. The heat flux can be reduced by constructing the divertor so that the tiles are not perpendicular to the particle beam, but there is a large inclination. However, this is not enough but achieving the 10 MW/m^2 limit from the plasma point of view requires understanding and controlling the power and particle transport and deposition processes.

The contribution of fast pedestal ions to the divertor power loads in ASDEX Upgrade tokamak was simulated using the ASCOT code. The fast ions from the top of the pedestal can not be described by fluid model, but a kinetic approach must be used. This is because the fast ions are affected by collisions less than slow particles. Therefore the energy distribution of the ions in the divertor can be non-Maxwellian even if the initial energy distribution was Maxwellian.

Analysis of the unperturbed orbits indicates that inner and outer divertor plates should receive approximately equal particle flux. When the effect of collisions is included, the outer plate should receive more particles. This change in the flux asymmetry is mainly caused by pitch collisions and the experimental fact that there are more particles near the colder divertor plate which, in our case, was the inner one.

Several ensemble simulations were carried for ions from the top of the pedestal. Simulations with a simple plasma background, where there was no SOL background nor electric field, and T_i=T_e was assumed, showed that the power load asymmetry between the divertor plates can be reversed by modifying the collisionality. The results were consistent with the previous ASCOT simulations done for JET.

Simulations, where a background with realistic ion temperature distribution was used, also showed the above mentioned reversal in the power load asymmetry. Furthermore, an interesting "cup" like behaviour (see Figure 3.14) was observed in the load asymmetry as a function of collisionality when either the initial temperature or the ion density was varied, see Figure 3.14.

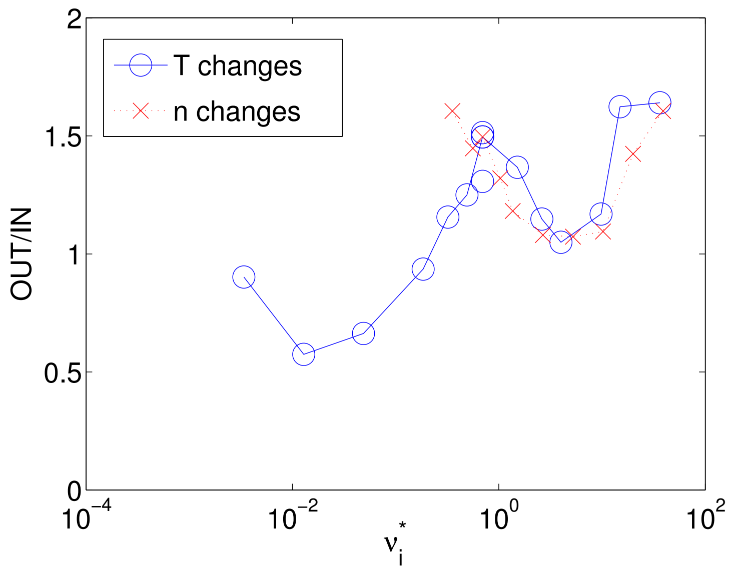


Figure 3.14: The effect of collisionality on power load asymmetry. The collisionality was varied by scaling the temperature and the density separately. The effect of different random number seeds was studied for the real discharge parameters in the middle.

Simulations with 2D SOL background showed that the collisions in the SOL have two roles. First, the collisions bring more ions to the divertor, and secondly, they cause the ion flux to shift away from the inner divertor plate. Simulations with constant radial electric field did not show any noticeable effect on the power loads.

3.2.4 Simulation of Material Transport in SOL

The present project has been initiated in spring 2004 as the ERO code was obtained from Forschungszentrum Jülich. ERO was installed into the computers of CSC, the Finnish IT Centre for Science, and later to a workstation at TKK, which was found to be a more convenient environment for code maintenance and simulations. In 2004, solving incompatibility problems required much effort, as the code had been only used at the FZJ server and, therefore, not easily portable.

Besides technical modifications, the TKK team has so far implemented a few significant improvements in ERO, two of which are described below. First, the missing toroidal dimension has been added into the 2D divertor version. Second, merging the limiter and divertor versions into a single fully functional code has been initiated.

Earlier it has been possible to obtain only toroidally integrated erosion and deposition profiles from divertor simulations and many experiments are indeed nearly toroidally symmetric. In local puffing experiments, however, toroidal symmetry can not be assumed, and a truly 3-dimensional simulation is called for. For instance, the 2002/2003 tracer methane puffing experiment of ASDEX Upgrade requires simulations with the 3D version, providing at the same time an excellent opportunity for benchmarking the code. Here the deposition pattern of carbon originating from the gas outlet at the divertor was measured over a 2-dimensional area. The 3D version was completed in spring 2005 and first applied to this experiment. We found that the simulated toroidal decay length of carbon agrees with the measured value, but the cross-field width and shape of the measured distribution can not be reproduced at present. These simulations are continued once a more realistic SOLPS background plasma is available.

Over the years, several subversions of ERO have been born as a result of minor modifications for different purposes. For example, there have been separate codes for each simulated divertor tokamak: JET, ASDEX Upgrade and ITER. Moreover, limiter tokamaks and linear plasma machines have their own code branches. This poses a problem of development, since all branches have in common a large part of the code, which undergoes continuous development and is currently stored as several copies that are not linked to each other. To facilitate more efficient maintenance, a long-term goal is to have a single code capable of simulating all different cases. In summer 2005 implementing all divertor-relevant functionality as an option in the up-to-date limiter version started the work towards such unified ERO. At the moment this project is in testing phase.

ERO will be extended to global simulation during the next few years, but so far the simulations are spatially too limited for investigations of global carbon migration. For this task, the impurity transport code DIVIMP was imported to TKK during the summer 2005. DIVIMP is another Monte Carlo code that follows the migration and deposition of impurity neutrals and ions. It is capable of modelling the entire SOL of a tokamak, but it cannot resolve the gyro motion and lacks the description of chemistry. To combine the strengths of both codes we consider using ERO to describe the break-up of the puffed methane and providing the distribution of ion species as an input to DIVIMP.

Thus far, DIVIMP has been used to simulate the $^{13}CH_4$ puffing experiments performed in the end of ASDEX Upgrade 2004/2005 campaign. Because the current version of DIVIMP was unable to simulate the methane break-up, the puffing was modelled by placing a $^{13}C^+$ source in the midplane of the vacuum vessel. The resulting poloidal carbon deposition profiles will be compared with experimental profiles obtained at VTT Technical Research Center of Finland as soon as they are available. An image of a typical deposition profile is presented in Figure 3.15. It clearly shows that most deposition occurs on the wall close to the puffing location, but substantial amounts of carbon also end up on the divertor.

There have been and will be several visits abroad in connection to ERO work (FZJ, IPP and JET, a total of 18 weeks between March 2004 and March 2006), mostly funded through the Euratom Mobility Agreement and by JET.

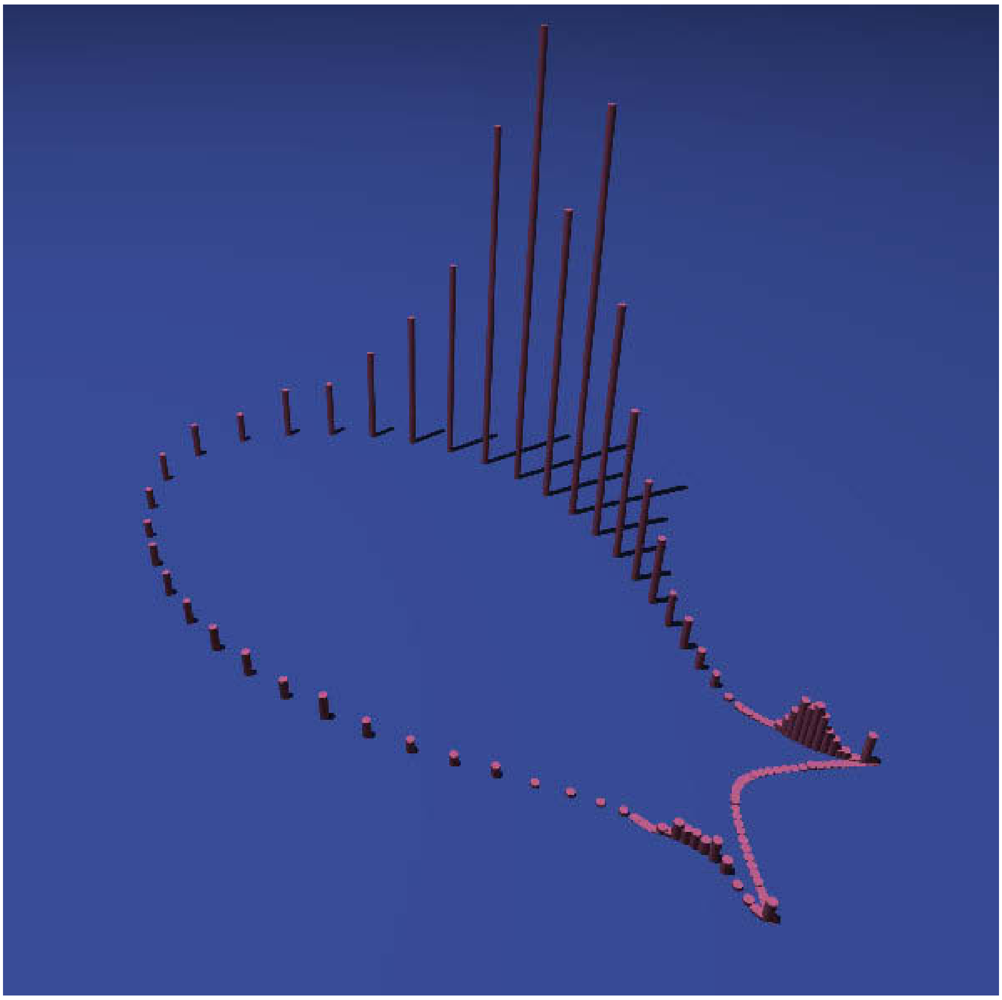


Figure 3.15: Poloidal deposition profile of carbon puffed into the reactor from the outer midplane. The heights of the columns represent the relative amounts of deposited carbon.

3.2.5 Stochastization of magnetic fields and magnetic reconnection

Project objectives are studies of ergodization and magnetic reconnection in ASDEX Upgrade tokamak. In particular, in the real geometry of this tokamak and for realistic plasma parameters in this machine the following problems will be studied.

- The topology of the magnetic fields will be investigated, such as the strength of the ergodization, the "diffusion" of the magnetic field lines, particle orbits in the ergodic zone etc.
- The numerical model is expected to show the penetration of the magnetic perturbations into the plasma, followed by magnetic reconnection. Are these processes delayed when the perturbations have very high frequencies?
- Will they proceed when the perturbations have spun-up the plasma to a matching rotation frequency?
- How important is the presence of a magnetic resonance in the plasma depending on parameters such as viscosity and perturbation amplitude?
- When a velocity-shear driven instability is important?

The role of stochastization of magnetic field lines in fast reconnection phenomena occurring in a magnetised fusion plasma has been analyzed. A mapping technique for the field lines of a toroidally confined plasma where the perturbation parameter is expressed in terms of experimental perturbation amplitudes determined from the ASDEX Upgrade tokamak has been used. It is found that fast reconnection observed during amplitude drops of the neoclassical tearing mode instability in the frequently interrupted regime can be related to stochastization. Quantitative criteria of stochasticity in reconnection have been found.

3.2.6 Gyrotron studies

The development work on a coaxial cavity gyrotron with an output power of 2 MW at 170 GHz as could be used for ITER has been continued. A new theory has been proposed for calculating azimuthal instability of radiation in gyrotrons with overmoded resonators. In a series of papers the Hamiltonian approach for calculated electron dynamics in gyrotron resonators has been developed. In particular a symplectic map has been proposed. Natural limits of power generated by a gyrotron have been established.

3.3 Strategic basic plasma research

3.3.1 Code development

Year 2005 has been time of active code development. In guiding-centre orbit following code ASCOT there have been improvements both in diagnostics and models in order to model the upcoming experiments at ASDEX Upgrade and JET more realistically and comprehensively. In gyrokinetic full *f* code ELMFIRE, major improvements have been done or are ongoing to save memory requirements thus making it possible to simulate larger devices in future. Development of plasma surface interaction code ERO has during this year included addision of third dimension, revision of numerical grid and some technical changes.

3.3.2 ASCOT code development

Improved Portability: After an intensive programming and testing effort, all mathematical library dependencies have been eliminated from ASCOT. As a result the code is now easily portable. Installation of ASCOT to the JET Analysis Cluster (JAC) at Culham, UK, has been completed, making the code available to numerous new users.

MDSPlus Compatibility: The data input and output of ASCOT are being standardized making use of the MDSplus data management platform. A test server for storing of ASCOT simulation results has been set up and a set of programs to transfer the results to and from the server has been programmed. In the future the server may be used for storing ELMFIRE results as well.

Output Distributions: Until recently, many output distributions of ASCOT have been in arbitrary units. The most important distributions have now been upgraded to physical units.

Importing SOL/Divertor Background Data of ASDEX Upgrade: Related to simulations of electron distributions in the scrape-off layer (SOL) and divertor plasma on ASDEX Upgrade, a preprocessor for interpolating two-dimensional AUG SOL backgrounds to a grid used in ASCOT has been programmed.

Stellarator Version: More benchmark testing of the stellarator version has been conducted in collaboration with Kiev Institute for Nuclear Research (KINR). Orbit topologies simulated by ASCOT and by the ORBIS code of KINR (using different coordinate systems) are in good qualitative agreement. Exactly matching background data for the two codes is required to achieve quantitative agreement of the results.

3.3.3 ELMFIRE code development

ELMFIRE is a massively parallelized (MPI) global full *f* gyrokinetic 5D particle-in-cell plasma turbulence code under development at TKK and VTT. So far ELMFIRE parallel runs have been done on the IBM eServer 1600 supercomputer with 32-128 processors at CSC. During 2006 version running on CSC's new HP ProLiant DL145 cluster "Sepeli" has been developed. After code optimization performed at Åbo Academi (graduate thesis supported by CSC) with AVL trees using hashed caches for efficient memory usage, ELMFIRE can be run with 500 processors in Sepeli for a 100x600x32 grid using 500 particles per cell, acceptable for turbulence saturation studies in reasonably sized annular volumes inside the ASDEX Upgrade toroidal plasma. With this optimized version of the AVL-tree, the program runs up to 30 % faster than the original matrix representation and scales to larger problems. The capabilities will further increase with the introduction of the new supercomputing resources at CSC which are being acquired. Programming of domain decomposition further decreasing memory requirements is in progress.

Benchmarking ELMFIRE to various edge and core plasma conditions with the myriad of other codes is in progress. Code has been benchmarked against the other gyrokinetic code predictions for so-called "Cyclone base case" linear mode characteristics and nonlinear saturation level of heat diffusivity in adiabatic and kinetic electron simulations. Moreover, detailed predictions of neoclassical properties, including the radial electric field, in the presence of turbulence have been performed. One major effort has been the inclusion of impurities (Oxygen) into simulation including the changes needed to the Monte Carlo binary collision model. Also, the probability distribution function (PDF) of density perturbations in the code has been compared to experimental probe measurements at FT-2 tokamak. The ELMFIRE code is included in IMP #4 which is one of the five EFDA Integrated Modeling Projects (IMP) and, also, it has been agreed with the Lawrence Livermore National Laboratory (LLNL) California, to test ELMFIRE (with the upgrades) against the plasma boundary codes under development there.

Figure 3.16: *ProLiant DL145 cluster is used in ELMFIRE simulations (© Petri Penttinen, CSC).*

3.3.4 Code upgrades and development of graphical user interfaces for the JAMS suite of codes

In 2004, a major code integration project was launched at JET with the aim of providing graphical user interfaces and common standards for input and output for a number of mainly transport and MHD stability codes used by the JET modelling community. This integrated suite of codes is known as JAMS (JET Application Management System). The code integration project has continued throughout 2005 with the installation of a number of new modules and upgrades, most notably graphical user interfaces for the 2D edge transport code EDGE2D and the linear MHD stability code ELITE, and a considerable extension of the capabilities of the 1.5D core transport code JETTO.

The Laboratory of Advanced Energy Systems has been heavily involved in developing the graphical user interfaces for ELITE and in developing the JETTO transport code. The recently released ELITE graphical user interface is similar in design and philosophy to an earlier launched interface for the MISHKA-1 ideal linear

MHD stability code. Specifically, the ELITE design is built to process output from transport simulations with codes in the JAMS suite, with the MHD equilibrium solver HELENA used as an intermediate stage. Scans in the α-s parameter space are supported by the interface. Here, α is the normalised pressure gradient and s is the magnetic shear.

The upgrades to the JETTO transport code include a restructuring of all the presently available ELM models, the addition of new theory-motivated linear ELM models, the inclusion of an energy sink model for modelling of ripple-induced convective thermal ion losses and the incorporation of an implementation of the so-called Porcelli sawtooth model. All these features have been implemented into the default version. The restructuring of the ELM models was necessary for the facilitation of new models and due to inconsistencies in the old design. The added theory-motivated ELM models are based on the concept that the mode amplitude of an unstable mode is calculated self-consistently from linear MHD models of instability. They include pure ballooning, pure peeling and combined ballooning-peeling models previously used in modelling work at JET. The option to enhance neo-classical resistivity during the ELMs as a way to model the dynamo β effect is also incorporated. In the model for convective thermal ion ripple losses, an energy sink term is included in the continuity equation for the ion pressure. In the Porcelli model, sawtooth stability is calculated self-consistently at each time step of the transport simulation. Upon violation of stability, a sawtooth crash is triggered. In addition to the publicly released models, various test version of JETTO have been created for internal use. Among other things, modules for the modelling of neo-classical tearing modes, of diffusive ripple transport and of an ergodic limiter have been written but not yet released for public use.

3.3.5 Particle-in-cell code development

Particle-in-cell (PIC) codes are presently developed and used for a variety of plasma applications in fusion and beyond. Extended PIC methods are applied at VTT for rf antenna coupling studies for ITER and present experiments like JET.

As a spin-off from fusion, the PIC codes will be applied to plasma chemical processing of materials including the oxidisation and decomposition of waste. This work will be done in close co-operation with the Finnish industry.

3.4 Plasma-wall interaction

3.4.1 The application of a 1D PIC code has been extended to industrial application. Deuterium irradiation induced defect concentration in W

Institutes: **University of Helsinki**, Accelerator Laboratory
VTT Processes

Researchers: T. Ahlgren, K. Heinola, J. Keinonen, E. Vainonen-Ahlgren, J. Likonen

Tungsten (W) has been proposed for first-wall material in thermonuclear reactors, where its behaviour in the presence of hydrogen (H) containing plasma irradiation at elevated temperatures is of key interest. Retention and re-emission of H isotopes, i.e. deuterium (D) and tritium (T), in tungsten are of great importance, especially tritium recycling and inventory.

Deuterium induced defects in polycrystalline tungsten were studied. Deuterium was implanted at room temperature into mirror-polished tungsten samples with energy of 30 keV per D^+ ion and dose of $5.8\cdot10^{16}$ cm^{-2} in order to make the depth profile measurements for implantation induced hydrogen traps possible. Fairly low implantation dose was used to avoid blister formation. Due to the high mobility of D atoms, only trapped D is left in the W sample after the implantation. Implanted W samples were annealed at elevated temperatures and the thermodesorption spectrum for the out-gassed D_2 molecules was recorded *in-situ* with a quadrupole mass spectrometer (QMS). Retained D-concentrations were analyzed with nuclear reaction analysis (NRA) and secondary ion mass spectrometry (SIMS). The SIMS results were normalised with nuclear analysis. In this way, the good depth resolution of SIMS was combined with the quantitative NRA method. For the D concentration determination, the NRA method with non-elastic nuclear $^3He(D,p)^4He$ reaction was implemented for the first time at the Accelerator Laboratory. SIMS measurements were carried out at Technical Research Centre of Finland.

As results, we observed four different defect types in tungsten that trap deuterium. In Figure 3.17 are presented the QMS results for the out-gassed D_2 molecules. Annealing of the W samples releases D atoms gradually from the traps, resulting in D_2 molecule desorption from the sample surface. Thermodesorption spectra and annealing temperatures are presented in the figure as a function of annealing time for removing D atoms from different traps. Release temperatures for different traps according to numerical analysis were 455, 560, 663 and 801 K. The corresponding release temperatures are pointed by dashed lines in the figure.

In Figure 3.18 are presented the D depth profiles from the as-implanted and annealed W samples measured by SIMS. The colored areas refer to trapped D atom concentrations, which are released during annealing (see Figure 3.17).

Table 3-2: Parameters for the four different traps resulting from 5.8×10^{16} cm^{-2} 30-keV D atom implantation.

	Defects/cm^2	Temperature where D_2 release starts (K)
1st	1.5×10^{16}	455±5
2nd	9.1×10^{15}	560±2
3rd	4.8×10^{15}	663±2
4th	3.3×10^{15}	801±20

Total number of each defect type produced by 30-keV D implantation was obtained to be 0.260, 0.156, 0.082 and 0.056 traps cm^{-2}/implanted D atom. When the sample temperature reaches about 1100 K, no D is left in the vicinity of the surface. The observed similarity of the first D trap profile with the simulated SRIM-03 vacancy profile, makes it tempting to associate the first trap with irradiation induced vacancies. The traps which can be distributed far deeper than expected from D implantation range, are associated with lattice distortion due to implantation induced extended defects of interstitial type.

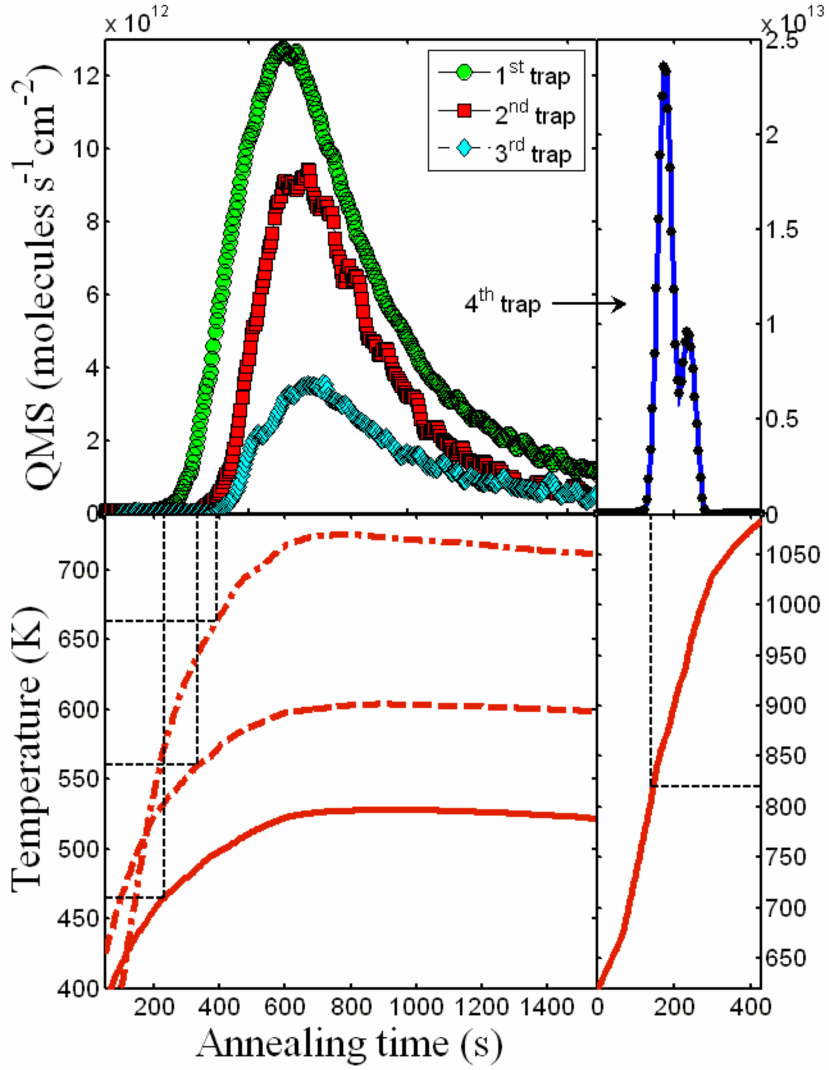


Figure 3.17: Ahlgren01.bmp:D2 molecule thermodesorption spectra and annealing temperatures as a function of annealing time for removing D atoms from four different traps in W. D release starts at sample temperatures 455, 560, 663 and 801 K from traps 1 to 4, respectively.

3.4.2 Blistering of W surfaces by low energy hydrogen ion bombardment

Institutes: **University of Helsinki**, Accelerator Laboratory

Researchers: K. O. E. Henriksson, A. V. Krasheninnikov, K. Nordlund

While the sputtering of tungsten by light ions is negligible, there exists another problematic irradiation-induced effect, which needs to be understood. Under intense hydrogen irradiation gas bubbles are formed under the surface. Depending on the implantation energy and dose, this zone can extend to a depth of several microns. When the bubble pressure exceeds a certain limit, the surface is fractured. Large pieces of material are then ejected to the boundary plasma. In 2004 we examined the blistering under He and very high-flux He conditions. In 2005 we extended the modelling to the use of object kinetic Monte Carlo, to enable modelling of the long-range H migration in the reactor, and subsequent bubble formation. The bubble formation is illustrated in Figure 3.19.

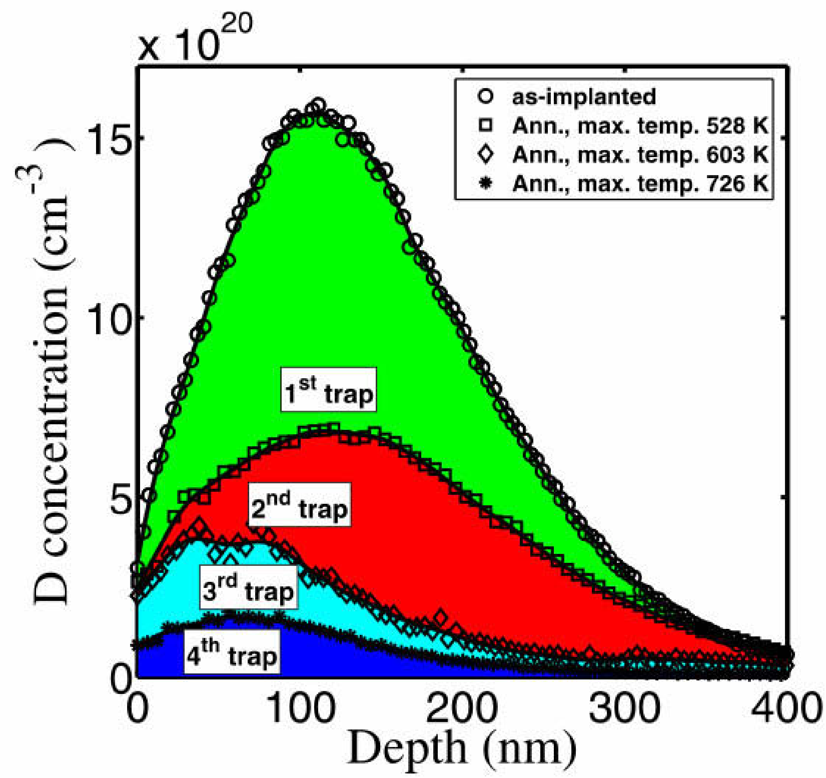


Figure 3.18: As-implanted and annealed D depth profiles measured by SIMS and normalised with NRA. The different coloured regions show profiles of each trap type.

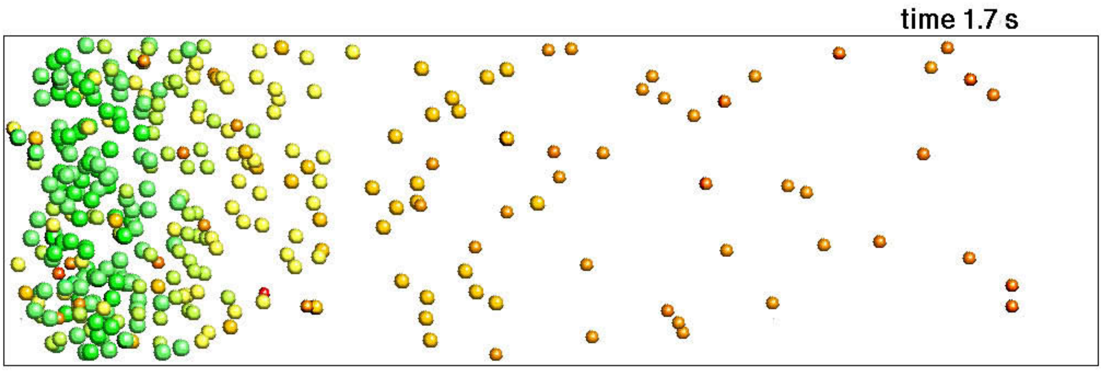


Figure 3.19: Object Kinetic Monte Carlo simulation of the buildup of He in W during prolonged He bombardment. Each ball shows a He cluster, with the ball sizes corresponding to the cluster size. The figure shows how a dense region of He clusters has formed in the near-surface region, preventing additional He atoms from entering deep into the sample. Such a He bubble layer may eventually lead to blistering and bubble rupture.

The results provided an explanation for the vastly different depths of hydrogen and helium bubbles found in experiments on tungsten after non-damaging ion irradiation. The fundamental reason is the different self-trapping behavior of the gas ions in the target. Our density functional theory calculations and molecular dynamics simulations showed that two hydrogen atoms trap each other weakly with a binding energy of less than 0.3 eV, whereas helium atoms form strong pairs with a binding energy of about 1 eV. Due to this, larger helium bubbles can form spontaneously close to the projected range, whereas hydrogen atoms migrate deep down in the tungsten target before becoming trapped by defects, as verified by the present results of kinetic Monte Carlo simulations. These findings are also relevant to blistering, as they help to explain why the lid thicknesses of hydrogen blisters are so much larger than the projected range of the ions.

3.4.3 Mixed material formation and erosion

Institutes: **University of Helsinki**, Accelerator Laboratory

Researchers: P. Träskelin, N. Juslin, K. O. E. Henriksson, A. V. Krasheninnikov, K. Nordlund

In ITER the transport of hydrocarbons in the reactor will lead to the formation of both C and WC thin films in the surface layers of the divertor. Using the interaction model for WCH developed in 2004 we can now model both the formation of WC thin films as well as the erosion of the WC material by H bombardment. A particularly important question is whether the WC material can erode by the swift chemical sputtering mechanism active in C. If this is true, the WC may erode much faster than the original W material.

Using a recently developed potential model for the W-C-H system, we examined the modification and erosion of tungsten carbide by incoming D ions. Simulations of the sputtering of pristine crystalline WC show only physical sputtering, with yields which agree well with experiments within the statistical uncertainties. Simulations of high-dose irradiation, where the change in the sample structure induced by each ion is taken into account, indicate that WC amorphizes during low-temperature ion irradiation. This leads to the formation of loosely bound carbon chains at the surface which can erode by chemical sputtering. On the other hand, during prolonged irradiation preferential sputtering (physical and chemical) will lead to a strong W enrichment at the surface, and then naturally the chemically enhanced sputtering of C will be much reduced or even completely cease.

An integrated particle-in-cell simulation model for incineration of organic radioactive or toxic waste in high-pressure low-temperature RF capacitive oxygen plasma is developed. Charged-particle kinetics, discharge evolution by RF power between the electrodes, neutral particle kinetics and chemistry in waste interaction by excited oxygen molecules are incorporated for the first time in the same model. With this model, a high waste incineration rate is found at large pressure.

3.5 ITER Equatorial Port Shielding

Institute: **VTT Processes**

Research Scientists: P. Kotiluoto, F. Wasastjerna

In ITER the shutdown dose rate is the critical quantity from the viewpoint of shielding design for the equatorial ports. If this can be kept below the required limit of 100 μSv/h, it seems that the other quantities affected by the port shielding design will also be acceptable. The shutdown dose rate is dominated by activation caused by neutrons streaming along the gap between the port plug and the port walls. This streaming accounts for something like 90% of the total dose rate, and, to keep it acceptable, a dogleg in the gap is essential.

Thus the design details in the dogleg area may have a significant influence on the shutdown dose rate and consequently on the whole adequacy of the port design. In a shielding calculation for a helium cooled pebble bed test blanket module (HCPB TBM), the authors found the design detail information available to them at the time somewhat inadequate. Therefore, we carried out a closer investigation of the design details and their effects on the shutdown dose rate. This study is relevant not only to the port containing the HCPB TBM. Presumably the other equatorial ports, except the NBI ports, will use the same design, so the results apply to them too.

The emphasis in this work was not so much on assessing the adequacy of any one design as on determining how various aspects of the design affected the streaming. Thus it was not considered necessary to calculate the actual shutdown dose rate for all designs. Instead the tallied quantity was the outgoing fast neutron current (above 0.1 MeV) at surface which intersects the port at right angles to its axis, 964.2 cm from the axis of the whole machine. This is about 40 cm beyond the outermost part of the dogleg in the model.

The calculations were done stepwise in the sense that first a calculation with the original geometry model (model gs01) was performed, then successive details of the geometry were changed with a calculation after each step (models gs02, gs03,…,gs07), until a geometry resembling that specified by the latest drawings was achieved (model gs08).

The fast neutron current, average neutron energy and average cosine angle of the current calculated for models gs02…gs08 were compared to the original model gs01. The results are described in more detail elsewhere. However, significant changes in the neutron current were observed. Especially, in the top and bottom gap the average energy and current increased notably when the filler shield tongues were deleted. At maximum, the neutron current was increased almost by a factor of 10 in the top and bottom gap (model gs08 compared to gs01).

The latest geometry model (model gs08) is shown in Figure 3.20. While there are still differences between the drawings and the model, we think that gs08 is reasonably similar to the drawings to be considered realistic.

We also performed a full neutron-photon calculation for model gs08, using the direct one-step (D1S) method to estimate the shutdown dose rate and to compare it to the shutdown dose rate by the original model. The methodology was similar to that used in our original calculations for the gs01 geometry.

The shutdown dose rate in the port extension exceeds 100 μSv/h in every cell in the port extension. The average (volume-weighted) dose rate in the port extension is 160 μSv/h, higher by a factor of 4.75 than the gs01 result.

Thus we have to conclude that the port and plug design embodied in the present drawings is unsatisfactory from a neutronics viewpoint. This does not, of course, mean that the design embodied in model gs01 is optimal. In particular, the narrower gap in front between the plug and the surrounding filler shields and manifolds is welcome. However, the dogleg length (the measurement marked with red two-headed arrows in Figure 3.20) is excessive and needs to be reduced to as small a value as practicable. This could be done either by shifting the shoulder of the plug forward (i.e., towards the plasma) or by adding material in front of the dogleg. Of these two approaches, the latter is likely to be far more effective.

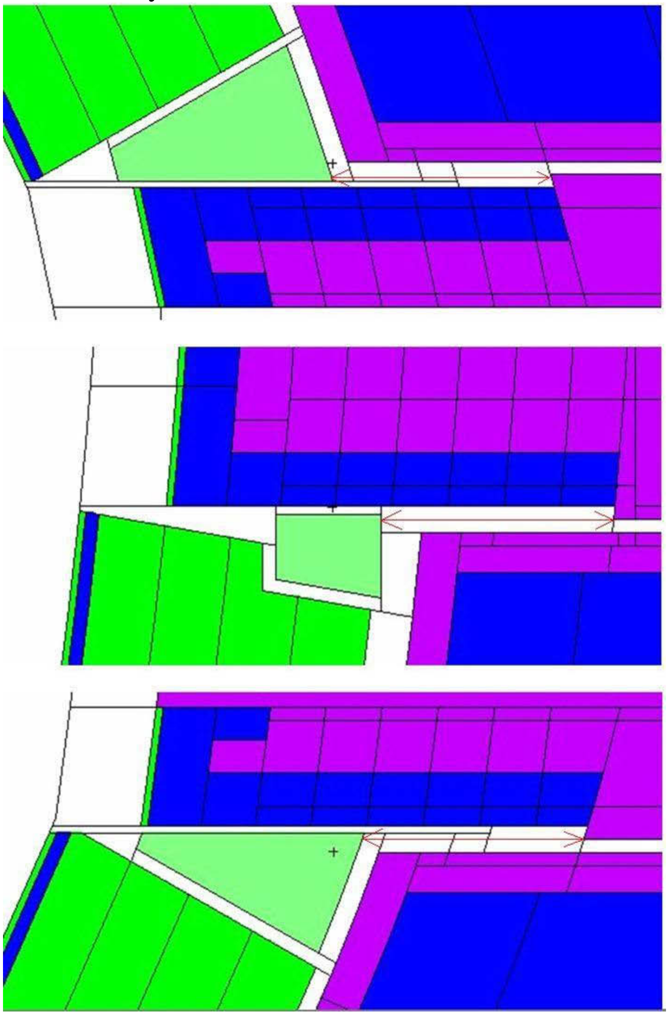


Figure 3.20: Top, side and bottom parts of the dogleg in model gs08.

4 EFDA TECHNOLOGY WORKPROGRAMME: PHYSICS INTEGRATION

4.1 Characterisation of AUG Wall Tiles and Plasma Facing Components with Surface Analytical Techniques

Institute:	**VTT Processes** (VTT PRO) **IPP/ASDEX Upgrade Team** (AUG)
Research Scientists:	Dr. E. Vainonen-Ahlgren (VTT PRO, Project Manager), Dr. J. Likonen (VTT PRO), MSc. T. Renvall (VTT PRO), Dr. V. Rohde (AUG), Dr. M. Mayer (AUG), MSc. J. Kolehmainen (Diarc), MSc. S. Tervakangas (Diarc).
Company:	**DIARC Technology Inc.**
Volume in 2005:	12 person-months

4.1.1 Introduction

Erosion and re-deposition are issues of major importance for ITER in that the rate of erosion of the divertor targets and building up of deposited films (and the T retained therein) may ultimately limit the choice of divertor materials and the operational space for ITER. Moreover, carbon deposition in nowadays tokamaks has been observed to be much higher at the inner than outer divertor. The reason for this asymmetry is unclear. A possible explanation might be a drift in the scrape-off layer (SOL) from outboard to inboard side. This work is a close collaboration with IPP under the Task Force IV (Divertor Physics and First Wall Materials), and focuses on experimental investigation of material transport in SOL and the study of in/out asymmetries in divertor deposition.

Before the 2004 campaign the AUG upper divertor marker tiles were prepared by arc discharge technique. 3 µm thick carbon layer on a 100 nm Re interlayer and 0.5 µm thick W layer were deposited for this purpose (Figure 4.1). The stripes were deposited by the company DIARC Technology Inc. using the DIARC plasma method. Carbon, tungsten and rhenium beams were created by plasma generators, and masked tiles were exposed to it while being rotated in a rack. The deposition was carried out in vacuum at room temperature.

$^{13}CH_4$ was puffed at the end of 2004 experimental campaign into the torus from the outer mid-plane during 5 identical top single null H-mode (ELM type I) discharges in hydrogen. The total amount of ^{13}C injected during the experiment was 1.48×10^{22} atoms. ^{13}C serves as a marker because it can be distinguished from the ^{12}C isotope with surface analytical methods.

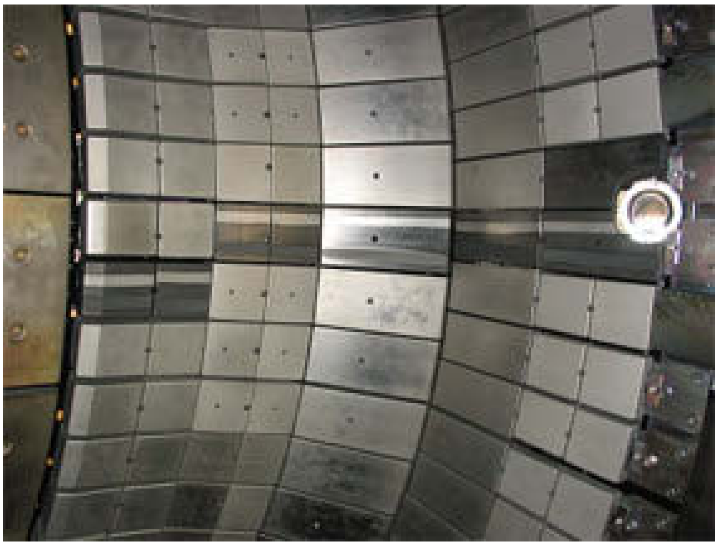


Figure 4.1: Upper divertor tiles. The bright and dark stripes correspond to W and C markers, respectively.

4.1.2 Results and Discussion

For the depth profiling secondary ion mass spectrometry (SIMS) was utilised. A set of samples of 10 mm in diameter was cut from the divertor tiles on IPP site. Area with a marker stripe and original graphite tile were present on the same sample (Figure 4.2).

Assuming that the sticking coefficient on impurities is different for W and C, SIMS measurements were performed on C marker stripe and graphite nearby W marker stripe. For the quantitative interpretation of the SIMS results, calibration sample with the same matrix is needed. Unfortunately no W sample for quantitative calibration was available, so measurements were performed on graphite nearby the W stripe. Total amount of ^{13}C found on DLC stripe and graphite nearby W was 12 and 6%, respectively.

Figure 4.2: Samples cutting.

Long term erosion/deposition data were also measured. Depth profiles obtained on the limiter tile PSL (s=19.4 m) on C and W marker stripes are presented in Figure 4.3 (a) and (b), respectively. In the case of C stripe (Figure 4.3 (a)), layer deposited by plasma during the campaign can be observed near the surface. Then the original C film with following Re interlayer can be seen. For the W stripe (Figure 4.3 (b)), layer deposited by plasma and original W film can be seen. In both cases films deposited by plasma are enriched with hydrogen, deuterium and boron. Furthermore, presence of carbon was observed in the layer deposited by plasma on the W stripe.

Also as a part of the task following marker stripes were deposited on 3 low inner divertor and 4 limiter tiles by the arc discharge technique. Carbon and tungsten films 3 and 2 µm thick, respectively, were grown on the all tiles. Additionally, 2 µm thick Ni and Al films were deposited on the limiter tiles. Re interlayer was deposited under all stripes.

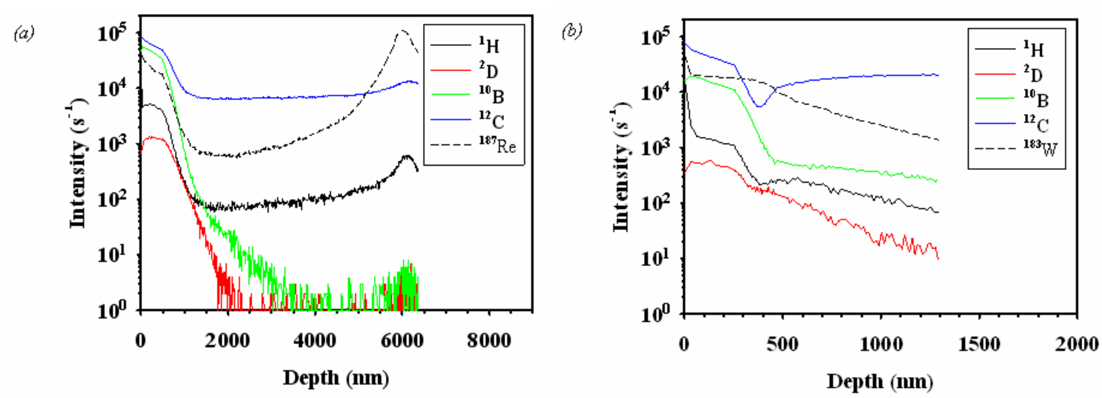


Figure 4.3: Long term depth profiles measure on C (a) and W (b) marker stripes. Presented spectra measured on the limiter tile PSL s=19.4m

5 EFDA TECHNOLOGY WORKPROGRAMME: VESSEL/IN-VESSEL

5.1 Further development of e-beam welding process with filler wire and through beam control

Institute: **VTT Industrial Systems**

Reseach Scientists: Miikka Karhu, Tommi Jokinen, Veli Kujanpää

EFDA Task: TW3-TVV-EBEAMS

5.1.1 Introduction

In spite of the fact that the title of the task refers to e-beam welding with filler wire, this research report presents the results of conduction limited laser-arc hybrid welding experiments carried out at VTT Industrial Systems. Above results from the fact that the direction and the content of the experimental program of the task changed by the proposal of EFDA from e-beam welding with filler wire into conduction limited hybrid welding.

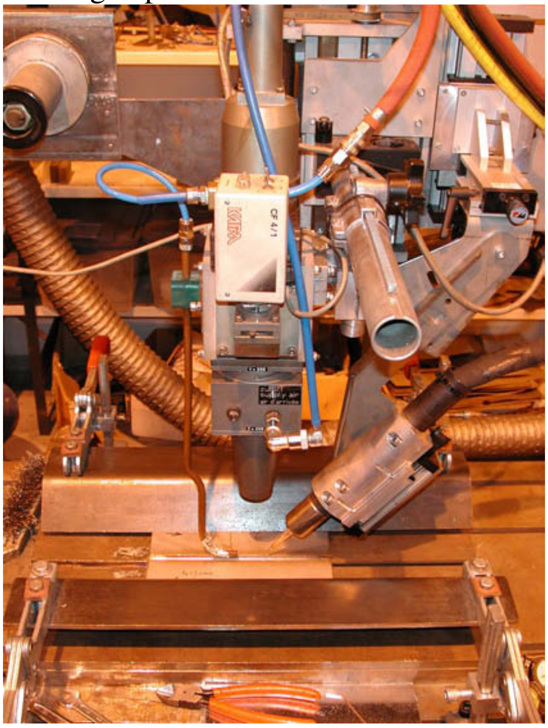
5.1.2 Objectives

It has been demonstrated in VTT's previous EFDA tasks, that hybrid welding with keyhole mode offers very efficient way to produce multi pass welds in very narrow gap thick section joints. However, in this new task, the main object was directed to a different approach on hybrid welding. In this approach, a power density of a laser beam spot was purposely dispersed by using defocusing. In the groove filling experiments, a power density of laser beam was kept in the range (~10^4 W/cm^2), which led the hybrid process towards to conduction limited regime. The experimental program included Nd:YAG + MIG conduction limited hybrid welding experiments, where the object was to study the feasibility to fill and bridge larger groove gaps than what can be welded with keyhole -mode hybrid welding.

5.1.3 Main results

In this new hybrid welding approach, the used groove configurations (especially width of the groove gaps) were resembled like those used in conventional narrow gap welding processes (e.g. NG TIG). When successful groove filling in above mentioned groove configurations is aimed, a keyhole mode hybrid welding can not be used in filling runs, because gap widths are too wide to be bridged. Under the circumstances, a laser part (in terms of power density of the beam at the work piece) as well as filler wire feeding rate in GMAW and welding speed has to be tuned in hybrid process such that much wider weld beads can be produced. Therefore, it was decided to disperse a power density of a laser beam spot purposely by using defocusing and in that way bring the welding process inside the conduction limited regime.

During the experimental program of this study, series of conduction limited hybrid welding tests for austenitic stainless steel test joints were carried out using a combination of 3 kW Nd:YAG laser and MIG welding process. In Figure 5.1 it is shown the set-up and an example of beam/wire interaction arrangement used in welding experiments.



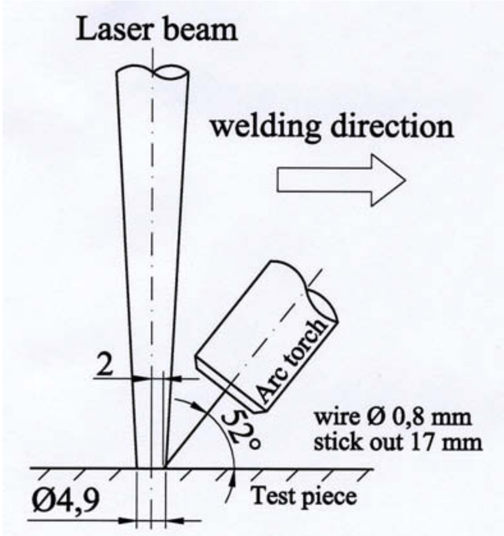


Figure 5.1: General view from the experimental set-up. On the right: An example of the beam/wire set-up used in hybrid process.

Constant parameters used in welding experiments can be seen from Table 5-1. Concerning variable process parameters, the main interest were laid on the following variables: Focal position together with the diameter of the impinging beam spot size, welding speed and filler wire feeding rate.

Table 5-1: The constant welding parameters.

The constant parameters	
Laser power	3 kW
Focal length	200 mm
Diameter of filler wire	0.8 mm
Stick out (electrode extension)	17 mm
Orientation and the angle of arc torch	leading, 52^{o}
Shielding gas and flow rate: - Via arc torch's nozzle - Via extra nozzle	 Argon, 12 l/min Argon, 12 l/min

Welding experiments started with preliminary study, in which a basic knowledge concerning the influence of parameter variations on process behaviour was studied. Parameter knowledge gained from the preliminary study was applied in the further experiments. In those experiments the feasibility to fill and bridge larger groove gaps than what can be welded with keyhole-mode hybrid welding was studied. Figure 5.2 presents a cross-sectional illustration of the groove configurations used in the further experiments with 10 mm thick test pieces. The experiments showed that with using conduction limited hybrid method in welding of filling passes even a gap width of near 11 mm can be bridged and filled, Figure 5.3 and Figure 5.4. This improved gap bridging ability concerning welding of filling passes based on the feasibility of conduction limited hybrid method to make filling passes wider than what can be produced by using a keyhole-mode hybrid welding alone.

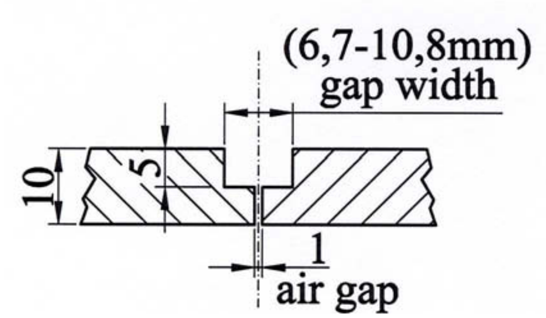
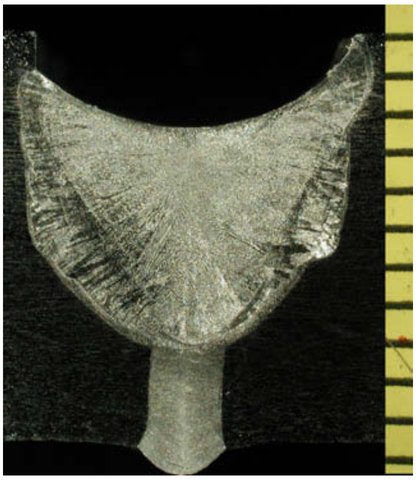


Figure 5.2: The groove geometry used in the further experiments. At first, the root pass was welded using keyhole hybrid welding and after that, one filling pass was welded using conduction limited mode. Varied gap widths were between 6,7 mm and 10,8 mm.

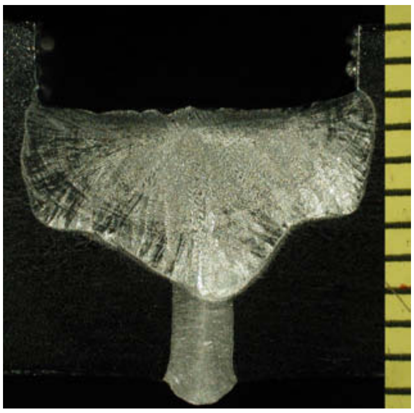


- Focal position:
(Defined from the surface of the root pass)
+ 50 mm / spot size ~Ø 5.9 mm

- Filler wire feeding rate:
17 m/min (25.3V/137A)

- Welding speed:
0.3 m/min

Figure 5.3: Macro cross-section of test weld and used variable parameters. The used gap width: 9.0 mm.



- Focal position:
(Defined from the surface of the root pass)
+ 60 mm / spot size ~Ø 7.0 mm

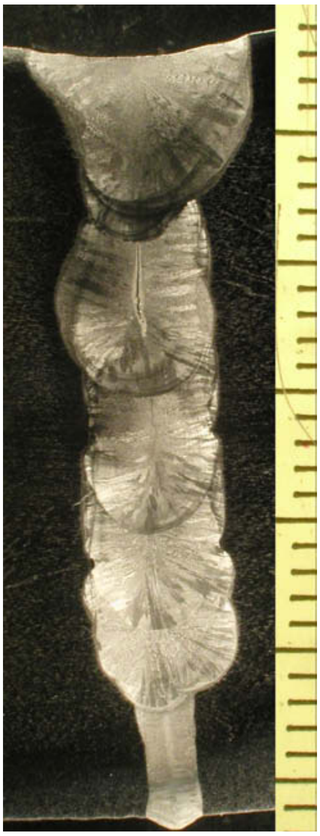
- Filler wire feeding rate:
18 m/min (25.9V/142A)

- Welding speed:
0.3 m/min

Figure 5.4: Macro cross-section of test weld and used variable parameters. The used gap width: 10.8 mm.

In the multi pass welding experiments of 30 mm thick test pieces approx. 8–10 mm wide, partially chamfered narrow gap grooves could be effectively filled; Using one pass per layer technique, approx. 24 mm deep groove geometries could be filled with

5 to 6 filling passes. That means on average a vertical fill-up of approx. 4–8 mm per filling pass, Figure 5.5 and Figure 5.6.



Vertical groove fill-ups for the each filling pass:
5. 7.0 mm
4. 5.0 mm
3. 4.3 mm
2. 3.8 mm
1. 4.5 mm
Defined by measuring the groove depth before and after welding.

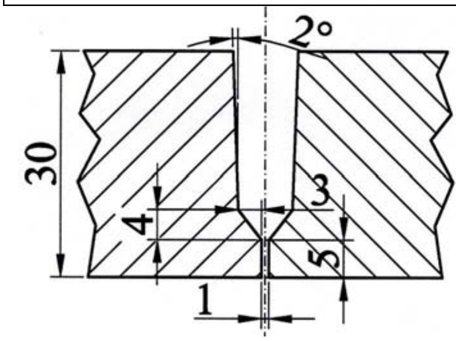


Figure 5.5: Macro cross-section of multi pass test weld and used groove geometry. On the right: Obtained vertical groove fill-ups for the each filling pass. Used variable parameters: welding speed 0.3–0.4 m/min, focal position +25–+40 mm, wire feeding rate 13.5–16.5 m/min.

The results from the distortion measurements confirmed the pre-assumption that distortions in conduction limited hybrid welding will be increased compared to the ones generally occur in keyhole hybrid welding method. This arises from the fact that used groove volumes were larger than the ones normally applied in keyhole mode hybrid welding. Larger/wider groove geometries need more melted filler material and more dispersed welding energy which in turn results in increased welding energy input and increased distortions.

During the multi pass welding experiments of 30 mm thick test pieces, it was realised that hot cracking had been occurred in certain upper filling passes of every multi pass weld. By large, the hot cracking had been occurred in the filling passes which depth to width ratio was in the range of 1.3–1.7. It is known that e.g. too large a depth to width ratio of produced weld pass combined to high rigidity of welded structure could cause increased hot cracking sensibility. Too large a depth to width ratio of a produced filling pass could be avoided by controlling the process parameters, but the effect of high overall rigidity of welded structure on the sensibility of hot cracking is a less known factor. In a certain degree, it could be advisable to put further efforts to the research work, which will deal with the hot cracking issue more thoroughly at the above mentioned point of view.

Vertical groove fill-ups for the each filling pass:
6. 2.6 mm
5. 4.2 mm
4. 3.8 mm
3. 4.1 mm
2. 4.4 mm
1. 4.4 mm
Defined by measuring groove the depth before and after welding.

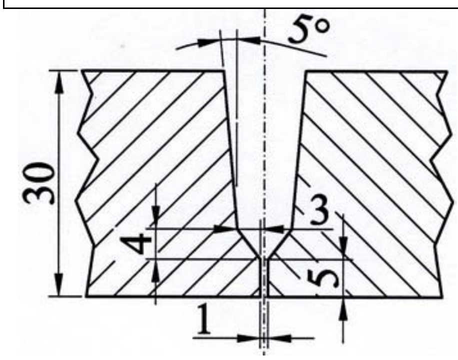


Figure 5.6: Macro cross-section of multi pass test weld and used groove geometry. On the right: Obtained vertical groove fill-ups for the each filling pass. Used variable parameters: welding speed 0.3–0.35 m/min, focal position +40–+45 mm, wire feeding rate 13–16.5 m/min.

5.2 DTP2 facility design update and integration studies

Institute: **VTT Industrial Systems**

Research Scientists: Jorma Järvenpää, Arto Timperi

EFDA Contract: EFDA/04-1192 (TW4-TVR-DTP2.2)

Objective

To adapt DTP2 platform and the cassette mock-up designs to meet the VTT facilities and assuring operator safety system design.

Brief description of the task deliverables

- Report of DTP2 platform design adaptation and quality assurance system development
- Adaptation of the DTP2 platform and the Cassette mock-up design for the VTT facilities
- Defining QA-procedure for precise delivery
- Preparing call-for-tender –documentation with EFDA
- Participation on procurement control and acceptance testing
- Report of the procurement project

Main Results in 2005

The first version of the DTP2 construction was designed by ENEA in Italy. VTT has checked the construction and redesign it to meet the VTT facilities. All modifications in the ITER reactor construction was dated also in DTP2 drawings (Figure 5.7).

In the ITER reactor was changed the locking system of the divertor cassette to the toroidal rails. The weight and shape was also changed. The new divertor mock-up cassette was designed according these modifications (Figure 5.8).

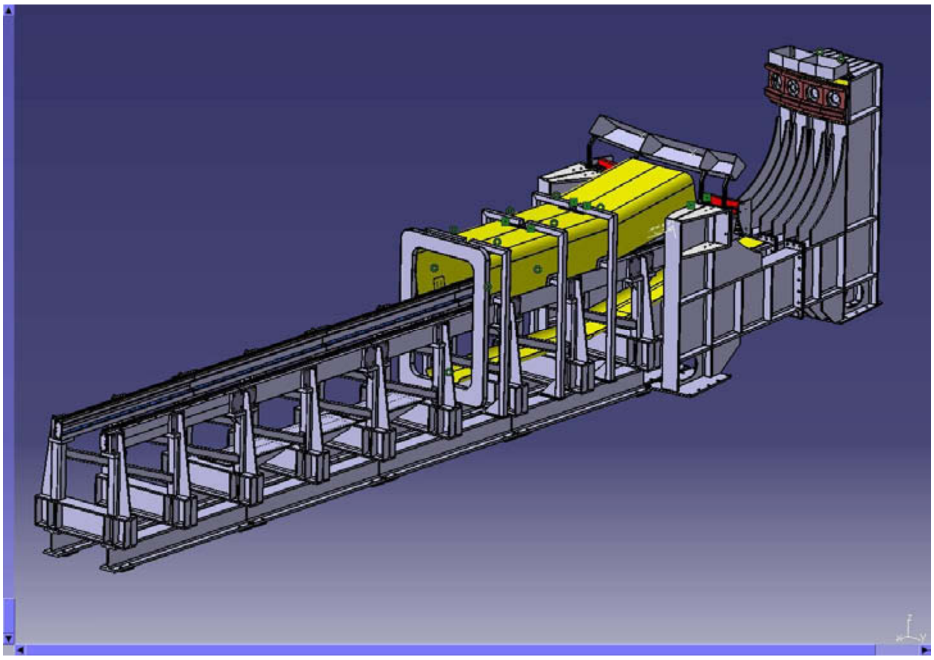


Figure 5.7: Figure 1. The DTP2 structure.

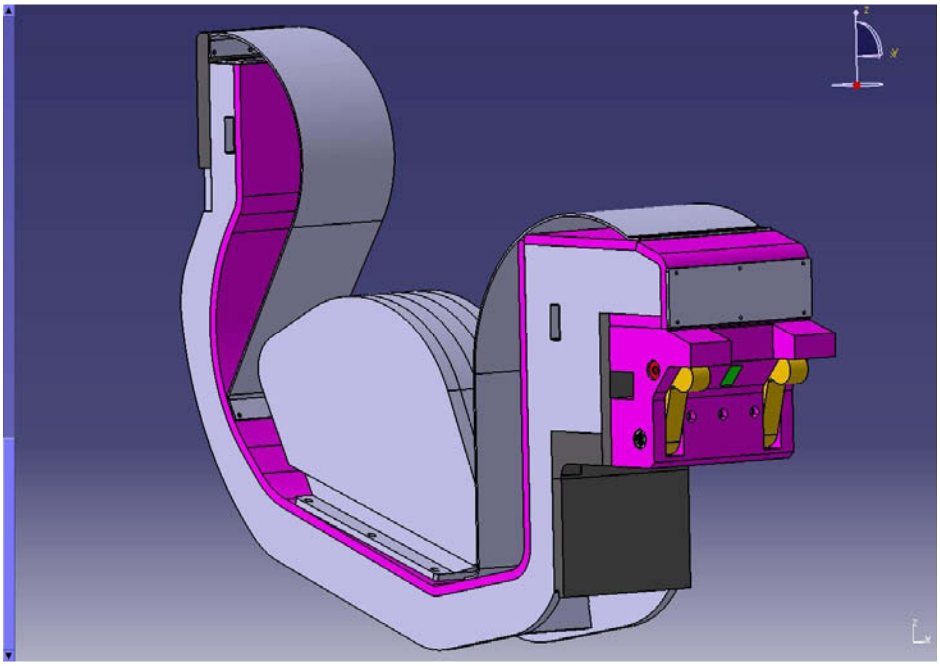


Figure 5.8: Divertor cassette mock-up.

For the DTP2 structure and the divertor casette mock-up was made technical specifications together with EFDA. EFDA asked tenders from manufacturers in the spring 2005 according these specifications. The manufacture of the DTP2 steel construction has been chosen TP-Konepaja Oy in Tampere and the structure should be at VTT in the end of the year 2006.

The final report "DTP2 facility design and integration studies" was published. In this report is described layout of the DTP2 and the divertor cassette mock-up construction and their design principles.

5.3 Provision of test models and software for the CMM system

Institute:	**Tampere University of Technology** Institute of Hydraulics and automation (TUT/IHA)
Research Scientists:	M. Siuko, M. Vilenius, J. Mattila, A. Muhammad, O. Linna, A. Sainio, A. Mäkelä, J. Poutanen, S. Verho, H Mäkinen, M Luomaranta and H. Saarinen
EFDA Task:	TW5-TVR-CMMHLC

5.3.1 Introduction

A Single Axis Model (SAM) representing a typical CMM/SCEE water-hydraulic axis was designed and built to be used as a general testing system for fusion reactor maintenance robot. SAM is used to prove the general applicability of a Moog Servo Controller (MSC) M3000 to control a large water-hydraulics actuator. In practise this means not only checking of reached positional accuracy and trajectory following accuracy of movements of a single axis but also testing of programming and communication methods of the MSC. This is done by integrating the MSC into a Higher Level Controller (HLC), which is performing multi-axis joint interpolation for controlling all axes of CMM/SCEE.

To support development of HLC and User Interface (UI), a virtual Mover (CMM + SCEE) has been created. Virtual Mover consists of both 3D model and dynamic simulation model of mechanics and hydraulics of CMM/SCEE. Virtual Mover, MSC, HLC are used to perform a hardware-in-the-loop (HIL) system.

5.3.2 Single Axis Model (SAM)

The general setup of the Single Axis Mock-up is presented in Figure 5.9. The body of the mock-up is based on a seesaw – principle, which allows testing of various loading situations to demonstrate realistic loading conditions of CMM/SCEE joints. When loads are balanced, the system is working with purely inertial load, which is the case with the SCEE when moving a cassette sideways. When loads are not balanced, the system is working with gravity and inertial loads, which is the case when the CMM is moving the cassette vertically.

Furthermore the Single Axis Model is designed to meet the loads, stiffness and hydraulics characteristics such that it represents both the CMM and SCEE axes.

Components (like cylinders, valves etc.) and solutions (like fixing of the column) are selected to be of the type to be used on the CMM/SCEE.

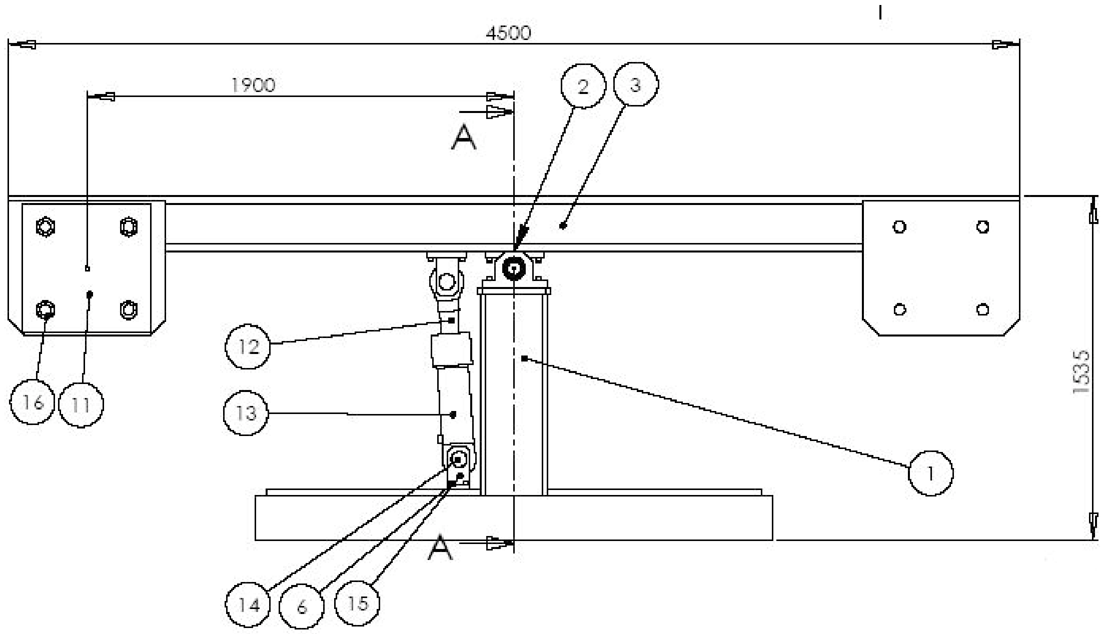


Figure 5.9: Main dimensions of the Single Axis Model.

Main components:

Base (1)	
Bearing housing (2)	
Rectangular beam (3)	250 x 150 x12 – 4500
Test weight (11)	247 kg per piece
Hydraulic cylinder (12) & (13)	32/25 or 125/80
Fastening fork for cylinder (15)	

Motion range of the test bench is ± 22.5 degrees, when the stroke of the hydraulic cylinder is 230 mm. The position of the hydraulic cylinder is adjustable.

Taken measurements proved that dynamic properties of the test system are similar to theoretical values of CMM and SCEE.

5.3.3 Development of User Interface and Higher Level Controller

Higher Level Controller consists of two desktop PC's, Figure 5.10. The other one is used to offer a interface for the operator to control the device on the task level (for example to select a file name which includes a set of trajectory points and to give a command to start moving through this trajectory) and the other one is used to perform time critical actions for the device, like joint interpolation for the previously selected trajectory. Physically, a real-time desktop PC is used to perform these time-critical tasks. Communication between these two PCs is done by using TCP/IP.

Moog Servo Controller programming environment is compatible with IEC61131-3 standard. Programming of the MSC has proven to be flexible and the environment offers wide range of function libraries. Expandability of the MSC is good when using external modules through a CAN-bus. For communication both CAN and Ethernet (UDP/IP-protocol) can be used. These both will be tested to select the best solution to change information between HLC and MSC.

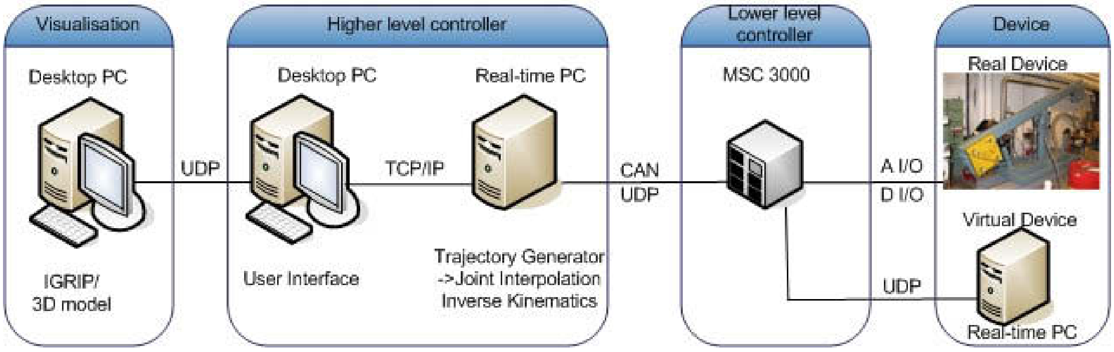


Figure 5.10: Main components of control system.

The project will continue until August 06. The developed HLC system is capable of driving the Mover in Cartesian coordinate system by utilizing verified trajectory control points saved in a file. On the other hand it is also possible to drive Mover in joint coordinate system with the aid of a joystick data. Joystick control methods in Cartesian frame will be added to HLC. In practice this means that it is possible to move Mover in coordinate systems familiar from industrial robots: moving relative to the base frame of the device, in the Tool Center Point frame (TCP) or in the Work Object frame. Further integration of system will be continued: adding watchdog functionality between real-time PC and MSC, further integration of HIL system to include controlling of the hydraulic on/off valves.

5.4 CMM Design Finalization

Institutes:	**Tampere University of Technology**, Institute of Hydraulics and automation (TUT/IHA) **Arelmek Oy**
Research Scientists:	M. Siuko, M. Vilenius, J. Mattila, S. Verho, H Mäkinen, H. Saarinen and Kyösti Keto
EFDA Contract:	EFDA/04-1191 (TW4-TVR-DTP2.1)

5.4.1 Objectives

Design changes to CMM and SCEE were made, because the cassette locking system was changed and the interface between the cassette and the SCEE was defined accurately. Some changes were also made to the structure of the main parts of the CMM and SCEE. These changes were made to improve manufacturability and mechanical properties (mainly strength) of the CMM and SCEE. The total weight of CMM and SCEE was reduced about 1000 kg due to the modifications.

Objectives were to update documentation of CMM design and 3D models of the simulation model according to the latest design changes of the prototype CMM and SCEE.

Checking of 3D models and updating of 3D simulations were made focusing into following areas: new cassette locking system, interface between the hook plate and the cassette and WHMAN task simulation dealing with second cassette un/locking.

Structural or dimensional changes were made to following CMM/SCEE parts:

-CMM Side wheel positions
-CMM Lift arm structure
-SCEE Adapter structure
-SCEE Cantilever arm structure
-SCEE Hook plate structure and dimensions

Because of the changes in dimensions and structures of the above mentioned parts also the stress analyses were reviewed, Figure 5.11.

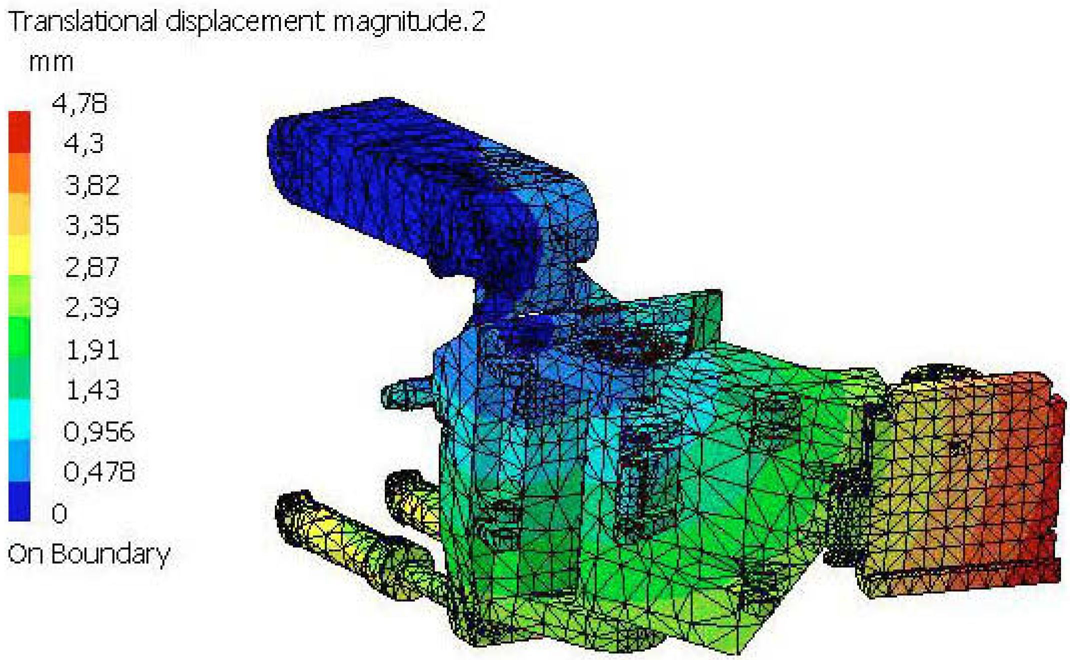


Figure 5.11: The CMM-SCEE lifting assembly reaching the furthermost left, with the cassette load.

5.5 Development of water hydraulic manipulator

Institute: **Tampere University of Technology**
Institute of Hydraulics and Automation (TUT/IHA)

Research Scientists: J. Mattila, M. Siuko, M. Vilenius, A. Muhammad, O. Linna, A. Sainio, A. Mäkelä, J. Poutanen and H. Saarinen.

EDFA Task: TW5-TVR-WHMAN

5.5.1 Introduction

The longer term aim of this task is to build the first teleoperated prototype Water Hydraulic Manipulator (WHMAN) for DTP2-platform (Divertor Test Platform 2). At

DTP2-platform ITER divertor replacement operations are studied, developed and demonstrated. WHMAN will be installed on the top CMM and SCEE and is needed for assisting work such as bolting/unbolting, pipe cutting and welding. WHMAN tasks are expected mostly to be teleoperated with force feedback (human-in-the-loop) due to complex nature of tasks and limited viewing system.

TW5-TVR-WHMAN-project target was to extend existing purpose build water hydraulic arm design with 3 DOF wrist mechanisms. Designed 3 DOF wrist will be installed to existing multi-joint mock-up arm and master-slave teleoperation methods are studied with developed water hydraulic manipulator using Phantom desktop 6 DOF force/torque feedback master arm.

TW5-TVR-WHMAN task consisted of two deliverables which were:

Deliverable 1: For wrist assembly actuators, improved water hydraulic vane actuator design, manufacturing and testing will be carried out. Also different fluid power transmission line (pipes, hoses and borings) designs are studied for wrist mechanism. The most failsafe, leakage free design which does not add too much joint friction or weight is chosen. Virtual prototyping is used extensively for improving wrist designs and for controller development prior to actual real prototype building. The 3 degree-of-freedom wrist assembly design is presented in final report.

Deliverable 2: The designed 3 DOF wrist will be manufactured, tested and installed to existing multi-joint mock-up arm and teleoperation methods will be studied with multi-joint water hydraulic manipulator using Phantom desktop 6 DOF force/torque feedback master arm. The results are presented in final report.

5.5.2 Deliverable 1: Improved water hydraulic vane actuator design

Water hydraulic vane actuator has superior performance when light and compact in volume but still high torque actuator is needed to drive a robotic joint. The downside of the vane actuator is the difficulty in finding an optimal balance between actuator seal leakage and seal friction. Also, load holding functionality is not as good that it is with hydraulic cylinder actuators.

The improved water hydraulic vane actuator prototype was designed with traditional engineering analysis rising from mechanical and fluid power engineering. Then the initial actuator chamber and shaft designs and material selections were optimized with the aid of FEM-analyses. Next figures (Figure 5.12 and Figure 5.13) show results of an example FEM-analyses.

In Figure 5.14 assembled water hydraulic vane actuator in test bed is shown. In test bed actuator leakage, friction and controllability issues were investigated.

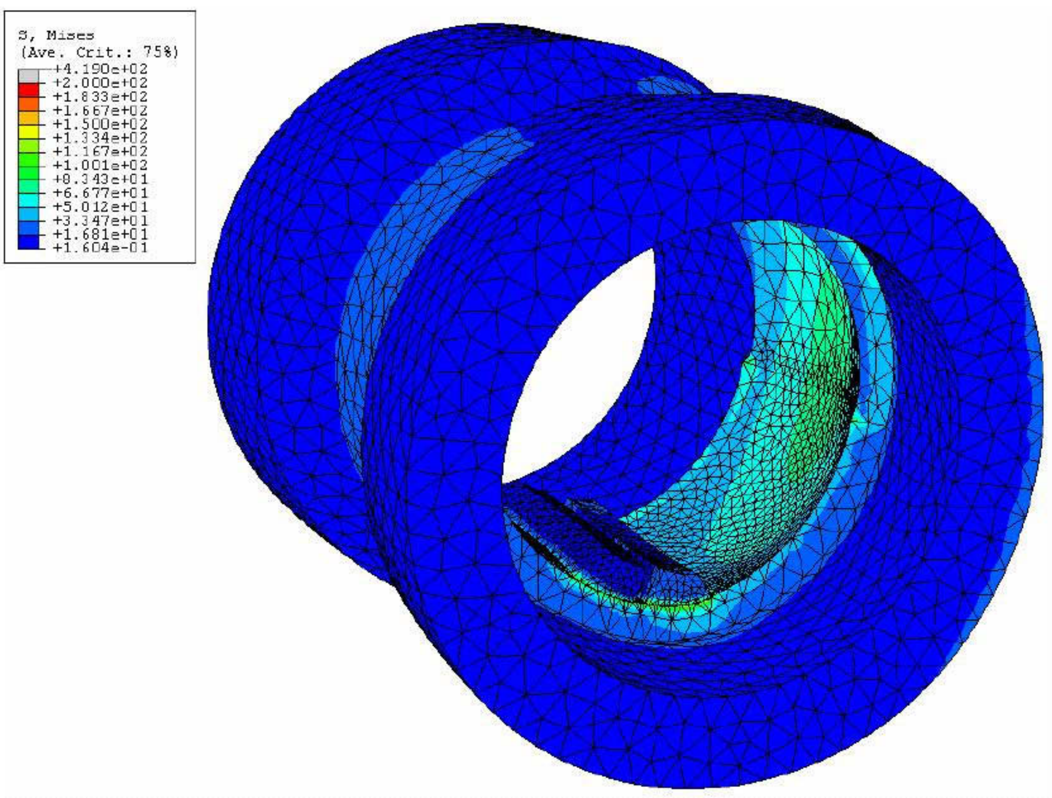


Figure 5.12: Tension field of vane actuator chamber.

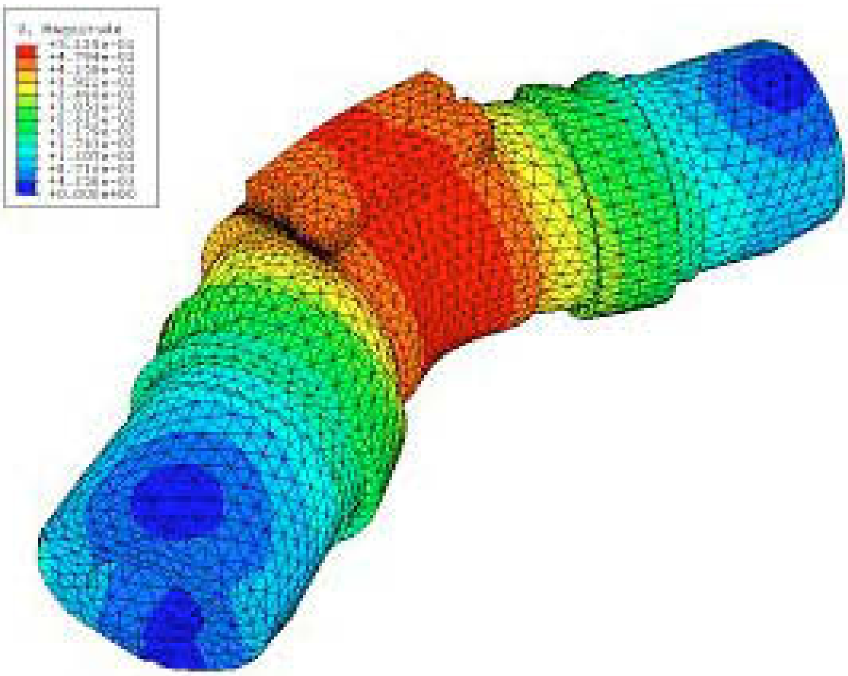


Figure 5.13: Tension Field of vane actuator axle.

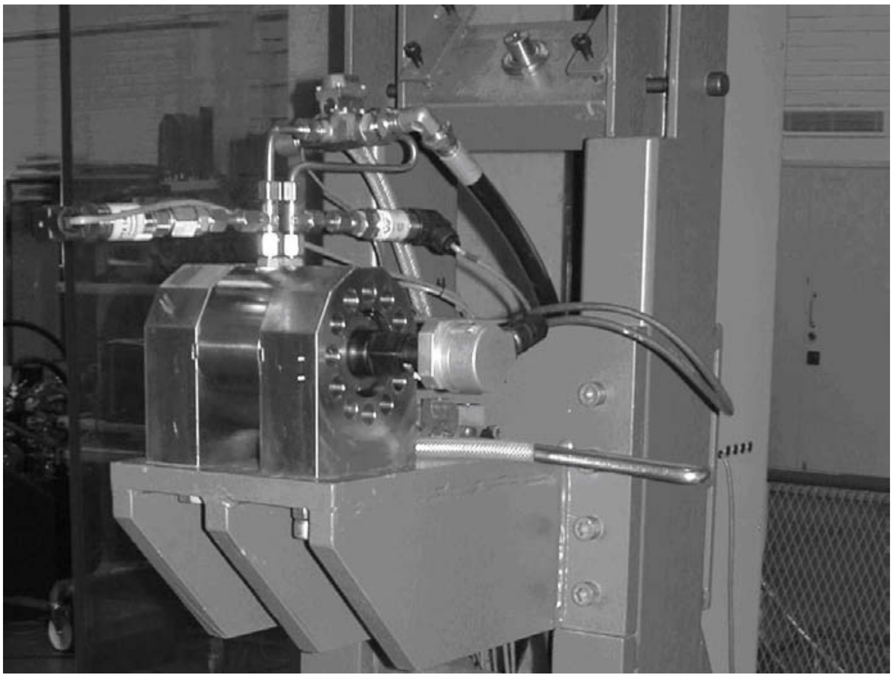


Figure 5.14: Water hydraulic vane actuator test bed.

The purpose of this sub-task was to build a design, analysing, manufacturing and testing procedure for building water hydraulic vane actuators of different nominal torque output. With the developed procedure remaining two water hydraulic vane actuators for the 3 DOF wrist were build.

5.5.3 Deliverable 2: Water hydraulic 3 DOF wrist design

Second task of TW5-TVR-WHMAN consist of two major parts which were:

a) Development of 3 DOF water hydraulic wrist for WHMAN
b) Teleoperation control system development for WHMAN with new 3 DOF wrist

Development of 3 DOF water hydraulic wrist for WHMAN

The final design of the 3 DOF WHMAN wrist is shown in Figure 5.15.

3 DOF WHMAN wrist consist of three water hydraulic vane actuators connected in series. Fluid power transmissions lines are fitted inside of manipulator structural linkages. After last joint 6 DOF force/torque sensor is installed for research and controller development purposes. Later manipulator gripper (not shown) will be installed to finalize the design. In Figure 5.16 water hydraulic arm developed in TW4-TVR-WHMAN with new 3 DOF wrist is shown. Water hydraulic arm developed earlier has rotation joints driven by water hydraulic cylinder (linear) actuators which allow limited joint range (typically below 120 Degrees). Therefore, in the

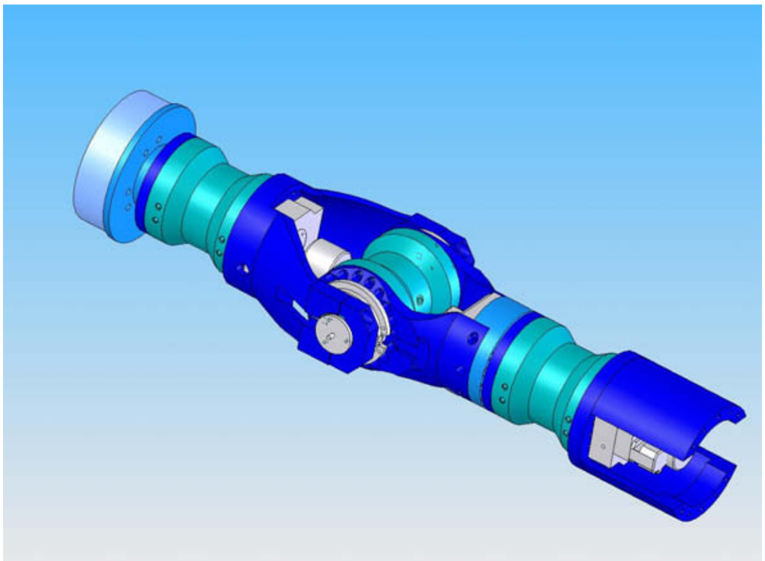


Figure 5.15: DOF WHMAN wrist final 3D model.

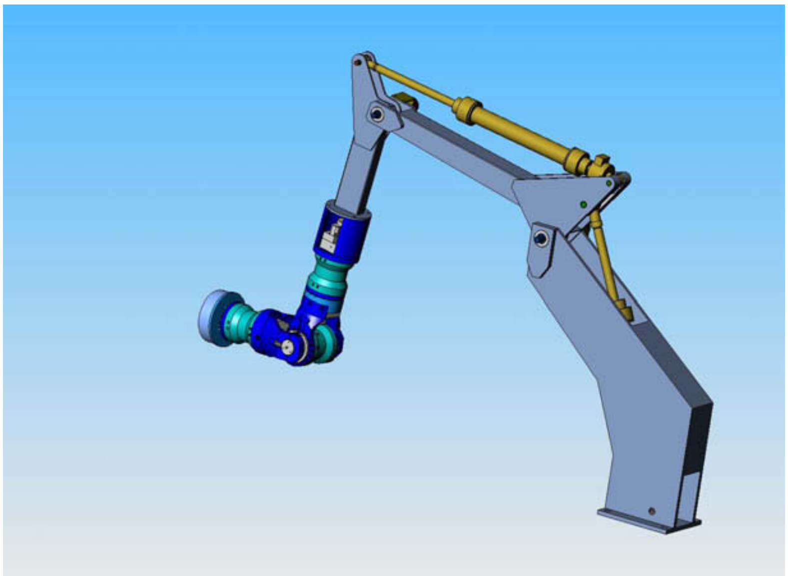


Figure 5.16: New 3 DOF wrist attached to existing water hydraulic arm.

Teleoperation control system development for new WHMAN with 3 DOF wrist

A block diagram of the IHA teleoperation control system architecture is shown in Figure 5.17. Teleoperation control system is of master-slave type. Master device is a Phantom Premium 6 DOF force/torque feedback device and slave device is either virtual prototype of WHMAN or actual WHMAN prototype.

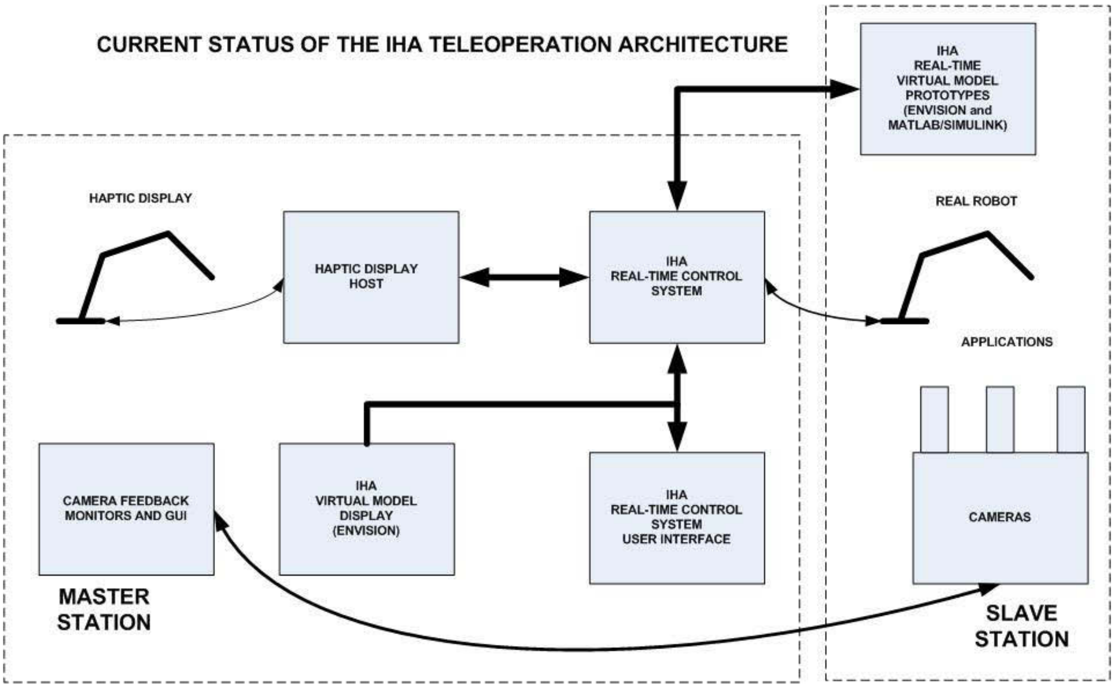


Figure 5.17: A block diagram of the IHA teleoperation control system.

5.6 Design and development towards a parallel water hydraulic intersector weld/cut robot IWR

Institute: **Lappeenranta University of Technology**
Institute of Mechatronics and Virtual Engineering (IMVE)

Researchers: Heikki Handroos, Huapeng Wu, Pekka Pessi, Yong Liu, Eeva Tenkanen, Hassan Yousefi, Jussi Hopia

EFDA Tasks: TW3-TVV-ROBASS and TW5-TVV-IWRFS

5.6.1 Introduction

Within the two on-going task TW3-TVV-ROBASS; Upgrade Robot to Include Water Hydraulics and a Linear Track and TW5-TVV-IWRFS; Demonstration of IWR Operational Feasibility the parallel Intersector Weld/Cut Robot has been develop Since the track support system of IWR was significantly modified by EFDA, the robot design had to be upgraded. The complete design of the robot trolley was modified to meet the new requirements. The robot kinematics was converted into 6-DOF Stewart type kinematics and three additional motion axes were introduced to meet the new work envelope requirement. In total the new robot version includes 10-degrees of mobility. The robot has now to be able to operate on the both sides of the track. The design is now being finalised with a manufacturing company and the manufactory of the mechanical parts are started. The design of water hydraulic system is been finalized. In addition to the actual design and optimisation of the hydromechanical system of IWR an emergency shutdown program for the robot controller to protect the robot and its parts against failures in cases of loss of control caused by electronic problems was developed within TW4-TVV-ROBOT; Further Development of Stability, Safety and Accuracy of IWR. Also a vibration damping method based on pietzo-electric actuators for IWR to reduce the vibration amplitude while machining was studied and demonstrated by a simplified mock-up. Furhermore, the applicability of laser measurement in measuring piston position of a water-hydraulic cylinder was studied by theoretical analysis and literature survey.

5.6.2 The new IWR concept providing 10 degrees of mobility

Figure 5.18 shows the assembly of the upgraded version of IWR. The waterhydraulic 6-DOF robot 1 is mounted on a steel carriage 2 which is driven by two separate servo motors including cyclo gears 4. The carriage moves along the track by employing rack and pinion drive. A waterhydraulic bearing force compensation system 3 is maintaining constant contact force between bearing wheels and rails. Since the radius of the track is varying the distance between the upper and lower wheel must vary in order to prevent damage. To be able to operate in the both sides of the track slewing bearing 8 with rotational drive unit 5 is introduced. To enlarge the work envelope of IWR an additional linear drive unit with bearings 6 will be used. Hydraulically driven tilting mechanism of the hexapod frame 9 is also needed to reach the lower areas of the work space.

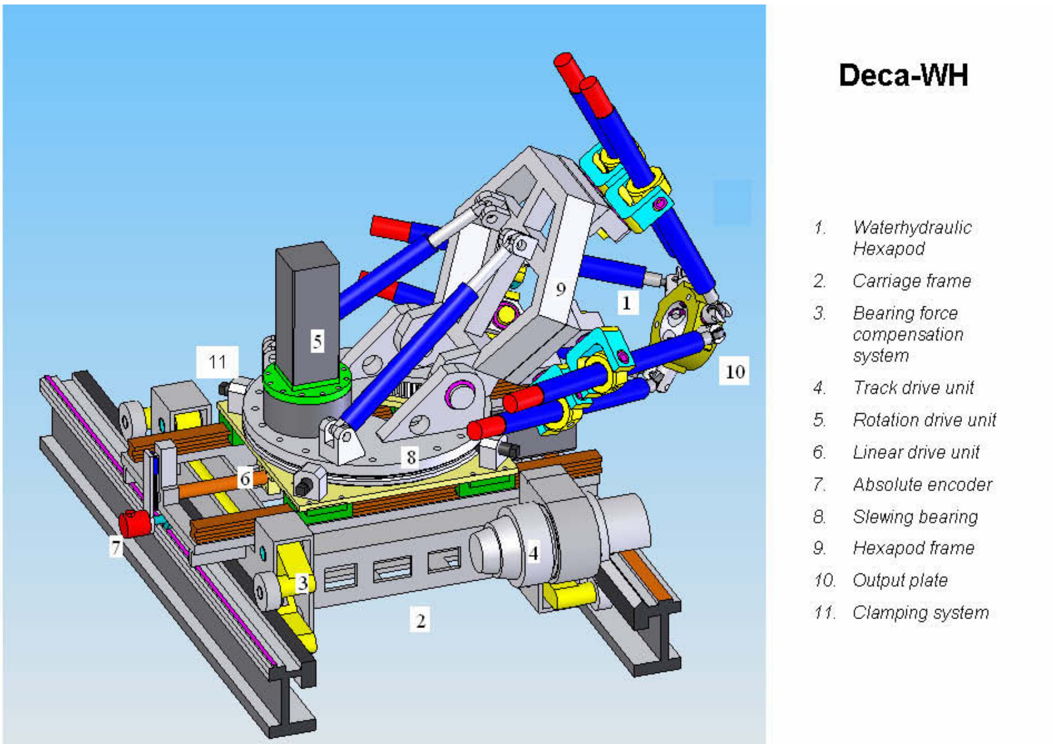


Figure 5.18: Upgraded IWR construction.

All ten degrees of mobility cannot be active during the operations of the robot. They are mainly used for moving the hexapod robot in appropriate positions to avoid workspace boundaries. The rotation table 8 is equipped by waterhydraulic clamping system 11 in order to improve the stiffness while machining.

The mechanical parts of the robot are currently manufactured by Imatran Kone Oy. The component selections for the waterhydraulic system of the robot is completed. The waterhydraulic servocylinders are currently designed and manufacturer by Hytar Oy.

5.6.3 Vibration suppression of IWR by piezo-hydraulic hybrid actuators

To demonstrate the idea of suppressing machining vibrations in IWR by piezo-hydraulic actuators was demonstrated by a simple test rig. A 30mm long piezo actuator with 300 micron stroke was purchased. A test rig including the piezo, a single hydraulic position servo and a hydraulic loading force servo was built and instrumented as shown in Figure 5.19. Load compensation tests were carried out by giving sinusoidal force into the position control system. The error caused by the flexibility of the hydraulic position servo was successfully compensated as shown in Figure 5.20.

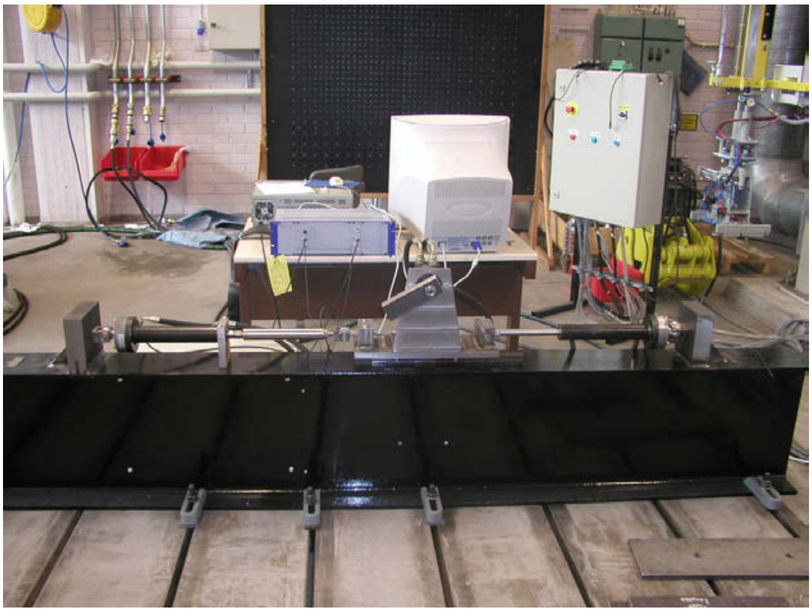


Figure 5.19: Test rig for piezo-hydraulic hybrid actuator.

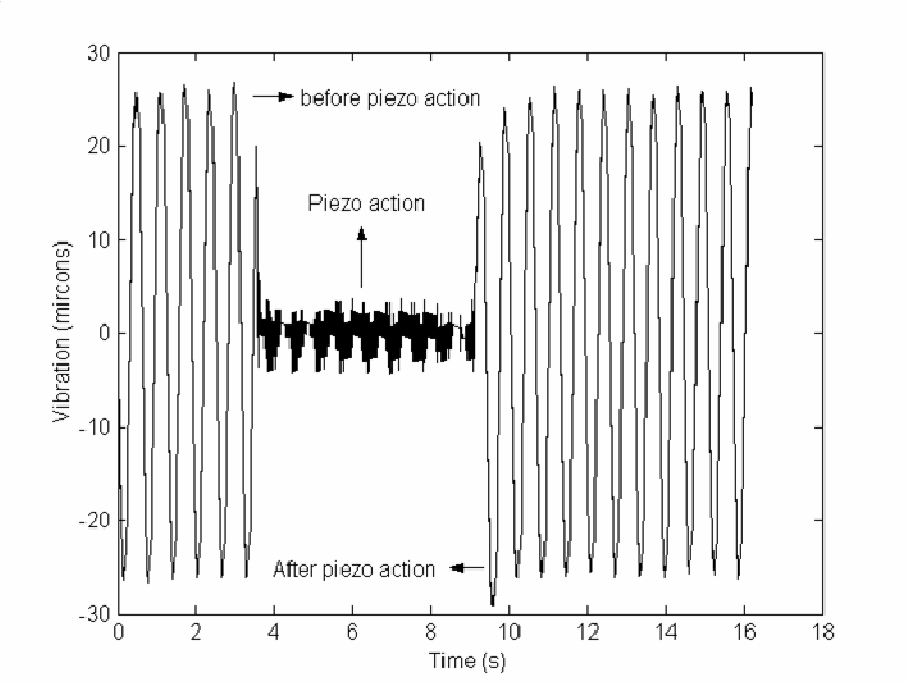


Figure 5.20: Experimental vibration suppression test result.

5.6.4 Laser interferometer position measurement of waterhydraulic cylinder

A novel method for measuring the position of a waterhydraulic cylinder was proposed. The method based on laser interferometer measurement was developed by using theoretical analysis and literature survey. According to theoretical study the method is applicable but expensive and it requires precise temperature measurement because the measurement principle is extremely sensitive to temperature variations. Also variations in water pressure must be taken into account. The experimental test system based on commercially available components was carefully designed. Because of unexpectedly high equipment costs and requirement for extreme temperature measurement accuracy the experimental testing plan was given up. Despite of this, the proposed method in conjunction with fiber optics might be promising solution in ITER environment because of its insensitivity to radiation.

5.6.5 Condition monitoring program for IWR

Before fault situations can be detected, all the possible faults, which could occur conceivably during the execution of task, must be defined.

Defined fault situations for the IWR robot are:

- Loss of sensor signal
- Defective sensor signal
- Jam of the valve spool
- Unexpected contact with environment
- Oversized machining force

After defining all fault situations, the detection methods were developed. To detect defective encoder signal or jammed spool, the speeds of cylinders are approximated from pressures and valve currents using hydraulic flow equations. To detect the overloading situations a Jacobian matrix based estimation of Cartesian forces from hydraulic forces is carried out in the developed program. The fault detection executes command to the robot to go to pre-defined safe-state. The safe-state must be chosen such that the robot can freely be driven into a position in which in can be transported into such track location in which it can be taken out from vacuum vessel for repair. Also when entering the safe-state, the robot is not allowed to collide with any obstacle. A monitoring algorithm was designed and demonstrated in Simulink program successfully.

5.7 In Reactor Fatigue Testing of Copper Alloys

Principal investigator: Seppo Tähtinen, VTT Industrial Systems

EFDA Task: TW3-TVM-COFAT2

Brief description of the task deliverables

The copper alloys used in the first wall and divertor components of ITER are likely to experience concurrently both static and cyclic stresses (thermal and mechanical) and an intense flux of 14 MeV neutrons. In addition, because of the nature of the plasma burn cycles, the materials used in these components will undergo creep deformation which will also be cyclic in nature. It is important, therefore, to simulate the impact of this creep-fatigue interaction on materials performance and lifetime under the dynamic conditions of neutron irradiation and applied cyclic stress with a certain hold time. This is the main objective of the present task.

Main results 2005

In order to investigate creep-fatigue interaction, fatigue modules were designed. The first versions of the fatigue modules were designed so that they could be used several times. After the in-reactor tests, the tested specimens were supposed to be changed with a manipulator. However, there were some problems with the clearances of the modules and also with the controlling program. Since usable results were not obtained in the preliminary tests, it was decided that a new fatigue module version should be designed. The new fatigue module version was designed to be non-reusable. The non-reusable design allowed firm fixing of the specimen which enabled the removal of the clearances. Two fatigue modules were manufactured according to the new design. One fatigue module is shown in Figure 5.21. The modules were tested and calibrated at VTT by using conventional PID-control program. The results were satisfactory and permission for in-reactor test was given.

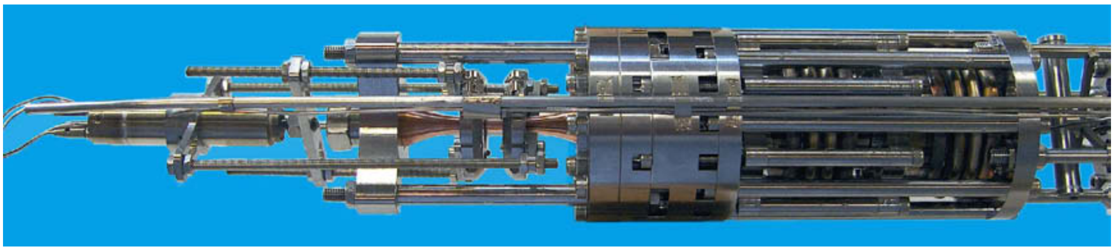


Figure 5.21: In-reactor fatigue module with a specimen and a displacement transformer.

The creep-fatigue experiments of unirradiated CuCrZr alloy have shown that the impact of creep-fatigue interaction is most damaging at low strain amplitudes. Therefore it was decided to use low (0.5 %) strain amplitude in the first in-reactor tests. In December 2005, two strain controlled creep-fatigue tests were conducted in the core of the BR-2 test reactor in Mol using VTT's pneumatic control system and fatigue loading modules. The creep effect was introduced by applying 10s and 100s holding times at maximum compression and tension. The temperature of the modules was approximately 70 and 90 ºC. Specimens lasted approximately 2500 cycles which took 4 days with 10s hold time and 10 days with 100s hold time. After the test, the rig was transported to a hot cell where the specimens were cut loose from the modules. The microstructure of the specimens will be examined later.

5.8 Testing of irradiated CuCrZr/SS joints produced under different blanket manufacturing conditions

Principal investigator: Seppo Tähtinen, VTT Industrial Systems

EFDA Task: TW4-TVM-CUSSPIT

Objectives

The aim of this task is to measure the effect of irradiation on the joint strength and base material strength. The irradiation will influence the joint strength but it will also affect the base material strength, the dependence of the joint strength on the base material strength, if any, shall be assessed.

An approximately 90 specimens have been irradiated, 30 each at doses of 0.001, 0.01 and 0.1 dpa with an irradiation temperature of 150°C, in inert gas. There will be for each dose 12 mini charpy specimens (27 mm x 4 mm x 3 mm) and 18 tensile specimens (39mm x 5mm x 1mm) (TBD). The irradiated samples will be provided by SCK-CEN.

The main objective of the present task is to carry out fracture toughness tests on post-irradiated Cu/SS joint specimens produced under different blanket manufacturing conditions. Data will be entered in the database and also post irradiated Cu-alloy mechanical properties will be assessed.
Fracture toughness testing and reporting of post-irradiation fracture toughness results
Data collection and assessing of post-irradiated Cu-alloy mechanical properties
Earlier studies have shown that mechanical and fracture toughness properties of copper alloy to stainless steel joint specimens are lower than those of the corresponding copper alloys and that those properties are dominated by the properties of the copper alloys, and particularly, by the strength mismatch and mismatch in strain hardening capacities between copper alloys and stainless steel.

Fracture toughness test results on unirradiated HIP joint specimens indicate that the strength mismatch between the base alloys is dominating the fracture toughness behaviour of HIP joint specimens. Those specimens with relatively high strength CuCrZr alloy, e.g., yield strength of copper alloy is higher or close to that of stainless steel, show relatively low fracture toughness values. On the other hand, those specimens which have relatively low strength CuCrZr alloy, e.g., yield strength of copper alloy is clearly lower than that of stainless steel, show relatively high or comparable fracture toughness values with those of CuCrZr alloy.

The test temperature and neutron irradiation primarily affects the copper alloy to stainless steel joint properties through changing the strength mismatch between the base alloys. Work will continue by testing the neutron irradiated samples during 2006.

6 EFDA TECHNOLOGY WORKPROGRAMME: TRITIUM BREEDING AND MATERIALS

6.1 Radiation damage in EUROFER: FeCr thermo-dynamics

Institute: **University of Helsinki**, Accelerator Laboratory

Research Scientists: N. Juslin, C. Björkas, K. Nordlund

EFDA Task: TW5-TTMS-007-13A

High-Cr, reduced-activation ferritic/martensitic steels, such as EUROFER, are receiving special attention as potential structural materials for future fusion reactors, thanks to their higher swelling resistance, higher thermal conductivity, lower thermal expansion and better liquid-metal compatibility than austenitic steels. For similar reasons, similar steels had been considered in the past for fast breeder reactors and are now being proposed for accelerator-driven systems and Gen-IV reactors. Their behaviour under irradiation has been extensively studied for the last 30 years. These studies have overall confirmed the suitability of high-Cr steels to withstand prolonged exposition to neutron irradiation. However, no experimental facility capable of reproducing the hard neutron spectra at high doses of a fusion environment currently exists. Thus, the in-service behaviour of these steels must be in fact *extrapolated* to the real conditions. In order to do so in a correct way, it is important not only to test the material under ever more realistic conditions, but also to reach a reasonable level of understanding of the physical phenomena which drive the material response to irradiation as a function of dose and temperature.

The first step in obtaining such understanding is to be able to model irradiation effects in the FeCr system. This system presents a particular challenge to computer simulations, since the heat of mixing of the alloy changes sign twice with increasing Cr content. We have, in close collaboration with the Swedish and Belgian fusion research associations, developed an interatomic potential for FeCr which describes correctly the thermodynamics of the alloy for all Cr compositions.

The thermodynamic testing of this potential has shown that the FeCr phase diagram is at least qualitatively correctly described in the temperature range of interest in fusion reactors. We have used the potential to assess how the presence of Cr affects the primary state of damage in FeCr. We simulated how atoms which have received recoil energies of 0.5–50 keV from reactor neutrons produce damage in FeCr and analyzed the distribution of damage in point defects and clusters. A typical damage distribution is illustrated in Figure 6.1. The results show that while the overall amount of damage is only slightly increased by the presence of Cr, the amount of Cr is enhanced in slightly enhanced interstitial defects. During long time scale evolution, the Cr content should increase further because it is energetically favourable for mobile interstitials to bind Cr atoms. This in turn is likely to affect the overall thermal aging behaviour of fusion reactor pressure vessel steels during the neutron irradiation it is subject to during the reactor operation.

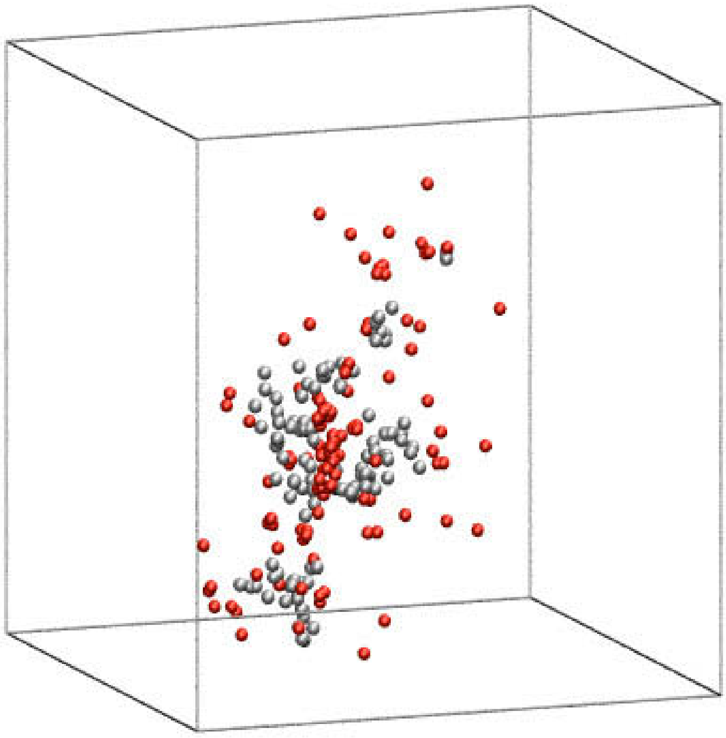


Figure 6.1: Distribution of damage in FeCr produced by an atom which has received a 20 keV kinetic recoil energy from a reactor neutron. The red balls show interstitial and the white balls vacancy atom positions.

6.2 IFMIF Test Facilities: Update of the MCNP model and shielding analysis of the ceiling

Institute: **VTT Processes**

Research Scientists: P. Kotiluoto, F. Wasastjerna

EFDA Task: TW5-TTMI-004-D3

The objective of the work was to model the "horseshoe shield" of IFMIF, to update the geometry in other ways, and to carry out a shielding calculation for the cover (ceiling) of the test cell.

The earlier IFMIF model, prepared in 2004, was first modified to include a horseshoe shield enclosing the low and very low flux test modules, with their associated graphite reflector, in front, on the sides and at the top and bottom, see figures (Figure 6.2) and (Figure 6.3). The shield was modelled as consisting of 8 layers of 5 cm each, except at the top where only 7 layers were used but a total thickness of 40 cm was retained. The purpose of this was to provide sufficient resolution for importances or weight windows, even if the shield thickness is increased, and to add flexibility to the model.

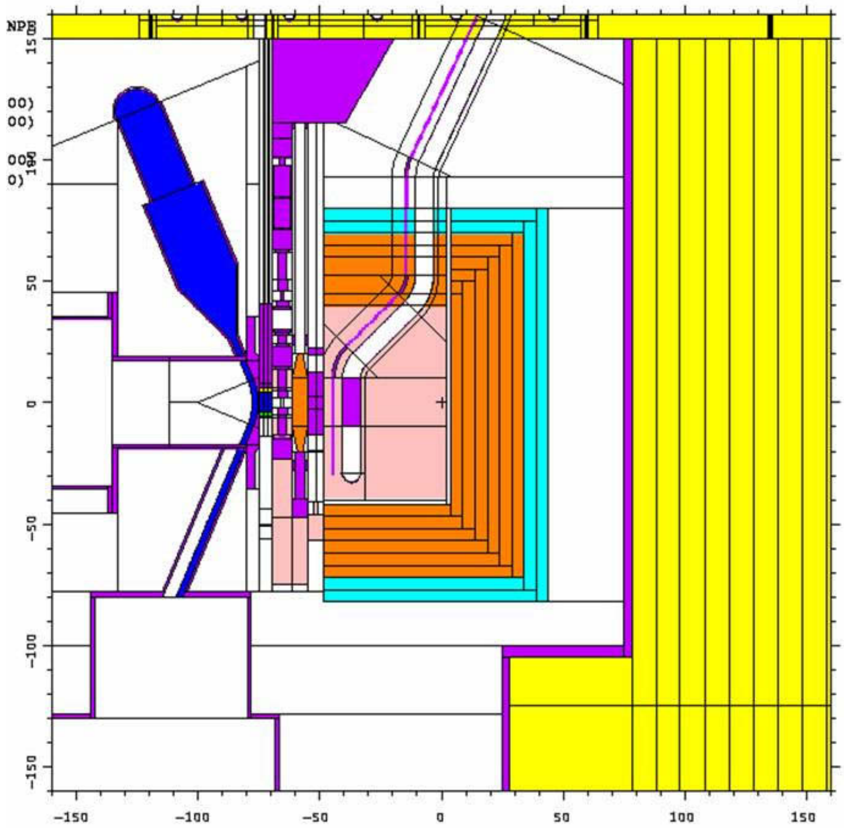


Figure 6.2: Elevation (px) view of geometry model md34 with the horseshoe shield.

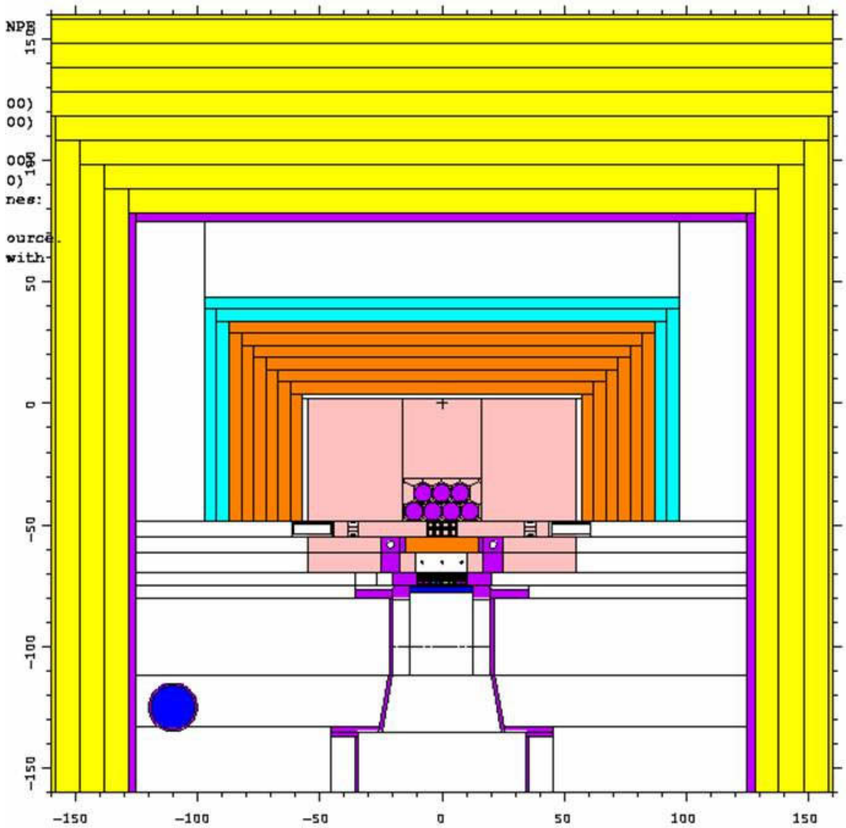


Figure 6.3: Plan (py) view of geometry model md34 with the horseshoe shield.

In addition, the latest drawings indicated that the dimensions of the VTA1, central, VTA2 and VIT shielding blocks had changed compared with earlier design. These changes were included in the updated model. Also the cell subdivision was changed to optimize it for cover shielding calculations.

The cover of the IFMIF test cell, as modelled, consists of blocks of concrete with circular and rectangular, almost planar, penetrations, with bends or offsets to limit streaming and with steel linings, usually assumed to have a thickness of 3 mm. The total thickness is 240 cm at the sides and 270 cm in the middle, where the VTA1, central, VTA2 and VIT shield blocks rise 30 cm above the rest of the cover.

Calculating neutron and gamma transport through such a shield and the dose rate above it is a challenging task. Only Monte Carlo programs can model streaming in the penetrations really adequately. We used McDeLicious, a variant of MCNP4C modified to cope with the D-Li reaction and the neutrons emerging from it with energies up to 55 MeV. McDeLicious, like any Monte Carlo program, needs very effective variance reduction to deliver usable results for a problem like this in practicable time.

A full n,p (neutron-photon) calculation is very demanding. Thus we preferred to start with a calculation covering only fast neutrons (above 0.1 MeV). This requires an order of magnitude less computer time.

However, even the fast neutron calculations proved troublesome. Using importances as a variance reduction method, we encountered some long histories, which may indicate that some streaming path has not been sampled by the preceding histories and that the calculated quantities might be underestimated. Therefore we were reluctant to undertake time-consuming n,p calculations. The reason for long histories should first be investigated. Nevertheless, the fast neutron calculations themselves gave a lower bound for the dose rate above the cover with IFMIF operating.

Calculations were carried out for three cases. In the first, the horseshoe shield was present. In the second, it was absent (replaced by void). The third was otherwise like the second case, but the linings of the ducts and shield blocks were made of aluminium instead of steel, since certain other calculations had hinted that light atoms might be preferable in lining materials for shield penetrations.

This shielding calculation revealed that, as it stands, the test cell cover is quite inadequate to permit a human presence in the access cell during operation, except possibly for very short periods. In all modelled cases the fast neutron dose rate alone exceeded the preliminary planned upper limit of 100 μSv/h at the top of the cover. If a human presence in the access cell is required only when IFMIF is shutdown, the shielding properties of the cover may be adequate, though that needs to be verified.

Based on calculations, replacing the steel linings with aluminium turned out not to help. On the contrary, it increased the dose rates, nearly by a factor of 2 in some locations.

Full n,p calculations of the shutdown dose rate are planned to be done in 2006.

6.3 European Blanket Programme: Tritium Breeding and Materials (EBP)

Principal investigator: Seppo Tähtinen, VTT Industrial Systems

EFDA Task: TW4-TTMS-005B

Objectives:

The objective of the Task is to study the properties of low activation ferritic stainless steel Eurofer. Deformation and irradiation behaviour of pure Fe and FeCr alloy will be studied during in-reactor tensile testing.

The performance of the in-reactor tensile tests will be based on earlier developed technology and will be carried out in collaboration with Risø and SCK-CEN. Manufacturing of tensile modules and modification of tensile specimens have been done. In-reactor experiments have been scheduled to be carried out during summer 2006.

7 EFDA TECHNOLOGY WORKPROGRAMME: MAGNETS

7.1 Cross-checking of the strand acceptance tests

Institute: **Tampere University of Technology/**Laboratory of Electromagnetics

Research Scientists: Risto Mikkonen, Iiro Hiltunen

7.1.1 Background

The conductor material for the different superconducting magnets in ITER will be either NbTi or Nb_3Sn. The primary objective of this task is to check and verify if the performance of the delivered Nb_3Sn strand from the advanced strand procurement action is according to the new strand specification.

Laboratory of Electromagnetics in TUT has facilities to characterize Nb_3Sn superconductors designed according to the specifications settled by EFDA. The temperature of the samples can be set with VTI device which enables an operating temperature between 1.6 K – 300 K. The external magnetic field is generated with a new hybrid solenoid providing the magnetic flux density up to 16 T. A new sample insert (including the current leads and instrumentation) was made at TUT. During the measurements the ITER type sample holder was used. For the benchmarking EFDA provided samples of Nb_3Sn strand. The heat treatment of the sample was made in Outokumpu Poricopper. The benchmarking tests were made twice in accordance to the specifications made by EFDA.

7.1.2 Objectives

According to the original plan TUT had to characterise five samples delivered by EFDA during 2005. The delivery of the samples was however delayed and at the same time it was verified that there was a leak in the VTI system delivered by Oxford. In addition in spring 2005 Outokumpu Italy constructed a new Nb_3Sn conductor. Because the repairing of the VTI system in Oxford takes a long time it was agreed that the final tests will be done to this new wire. Figure 7.1 shows a cross-section of a Nb_3Sn strand.

7.1.3 Results

The measuring program included the following items: The thickness of the Cr-plating around the conductor was determined by EDS studies. The Cu:non-Cu ratio was measured with a SEM image of the strand cross section with an optical microscope. The twist pitch of the conductor was determined by etching a section of the strand with HNO_3 and the pitch angle was estimated from the optical image of the strand. The residual resistance ratio of the copper in the conductor was measured at room temperature and at 20 K utilising a cryocooler. The critical current of the sample was measured at liquid helium temperature with an external magnetic field of 12 T with a high precision data acuisition system with a nanovolt meter. From the measured voltage current curve the conductor n-value (express the transition behaviour from superconducting to normal state) was also determined. The final report was delivered to EFDA in August 2005. Figure 7.2 shows a measuring set-up. The high field magnet is immersed in liquid helium inside the cryostat.

The benchmarking tests indicated that TUT has the facilities and know-how for carrying out the measurements specified in the contract and TUT has now a full readiness for continuing the co-operation which is scheduled to 2007.

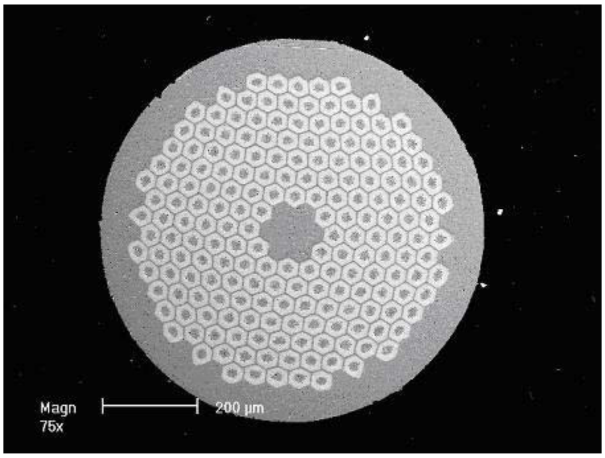


Figure 7.1: Cross-section of a Nb_3Sn superconductor.

Figure 7.2: The sample holder (left) with measuring cryostat.

8 EFDA TECHNOLOGY WORKPROGRAMME: SYSTEM STUDIES

8.1 Evaluation of fusion and other future energy production alternatives

Principal investigator: Antti Lehtilä, VTT Processes

EFDA Task: TW5-TRE-FESO-2

8.1.1 Introduction

The project has been carried out under the EFDA programme for Socio-economic Research on Fusion (SERF). The objectives of the project were to analyse the role of fusion in the future global energy system, focusing on interactions between global developments and the Finnish national energy system. As fusion is a very promising energy supply option for the future, it is important to evaluate the possible role of fusion during the latter part of this century, in view of the great global energy challenges ahead. One key area of analysis is the assessment of long-term scenarios of climate change mitigation, which can have great implications concerning the future competitive position of different energy technologies.

The project was carried out in collaboration with five other research groups, all using a common tool in their analyses, the EFDA Global TIMES model. The core component of the tool is the TIMES model generator, which has been developed as a joint effort under the ETSAP programme. VTT has been actively participating in this work, also during the EFDA project. The development and use of the model, however, requires also a sophisticated user shell. The use of the Global TIMES model is based on the VEDA modeling software developed by a Canadian consultant company Kanors, Inc., in close association with the ETSAP programme.

The first version of the database for the Global TIMES model was prepared in 2004 by a consortium of consultancies. This first version was used as a starting point for the work to be carried out in 2005 by the six research teams. During the year, a number of serious problems were, however, identified both in the modeling software and model database. Consequently, several major updates had to be made both to the software and database. However, no systematic documentation has been provided on the various changes, and this has posed an additional obstacle for carrying out the work as originally planned. Due to the many technical problems involved, on EFDA's initiative the due date of the projects was postponed to end of March, 2006.

Despite these difficulties, the model has been fully taken into use at VTT, and a series of preliminary scenario analyses have been made with it. Nonetheless, all members of the collaborating research teams continue to further improve the model database.

8.1.2 Research methods

The primary tool for making the energy system analyses is the Global TIMES model developed under the EFDA SERF programme. The model is a large and complex multi-regional partial equilibrium model. The basic version of the model consists of 15 world regions, with an additional distinction of the resources and production of primary energy in OPEC and non-OPEC countries within each region. International trade of crude oil, natural gas and LNG is included. During the project the model was augmented at VTT by an additional region describing the Finnish energy system.

Within each of the regions the structure of the model is organized in the form of a Reference Energy System (RES), as depicted in Figure 8.1. The RES describes all the relevant energy, material and emission flows in the energy system, from primary production to the demand of energy services. On a more detailed level, each of the sectoral blocks in the RES includes characterizations of various energy and process technologies relevant to the sector. In total, each sector includes the description of about 1500 different existing and new technologies. Energy services have been grouped into 43 distinct categories, each of which have its own baseline demand scenario. The maintenance and updating of such a large technology database is a demanding task. Within the project, parts of the database were reviewed and updated at VTT, under the coordination by EFDA.

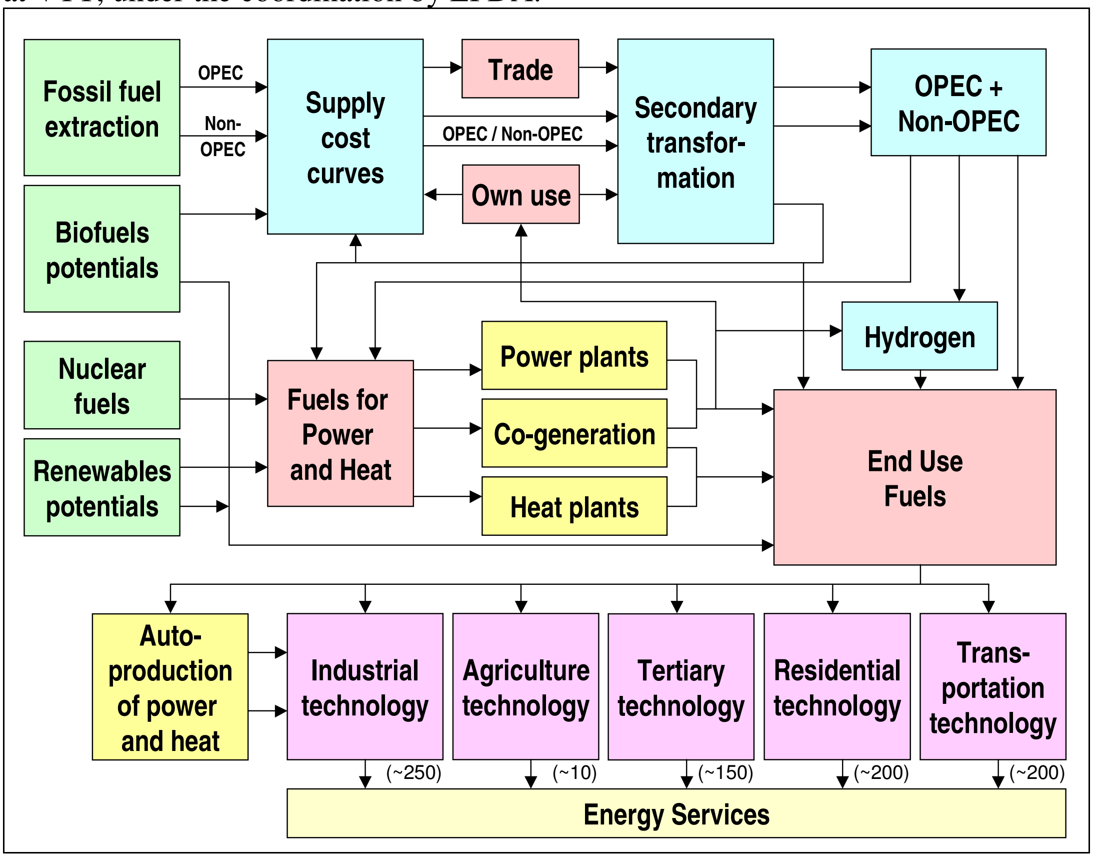


Figure 8.1: Simplified structure of the Reference Energy System (RES) in each region.

For the analyses of the possible role of fusion power, estimates are needed about the development of fusion power plant technology until 2100. The timing of the introduction of the first commercial plants, the development of the investment costs, and the potential maximum pace of implementation are among the key issues to be addressed.

Figure 8.2 illustrates tentative scenario alternatives constructed for the investment costs (based on e.g. the EFDA PPCS study) and maximum penetration. In the high penetration scenario, the share of fusion power could reach about 20% of total world electricity generation by 2100. In the base scenario the introduction of the first commercial plants is assumed to occur about a decade later, which would lead to a considerably lower maximum potential. In both cases the constraints imposed by tritium supply and total power plant construction capacity have been taken into account. Note that in the case of fission reactors the current total capacity is only about 400 GW, and the maximum construction rate has been about 30 GW/y.

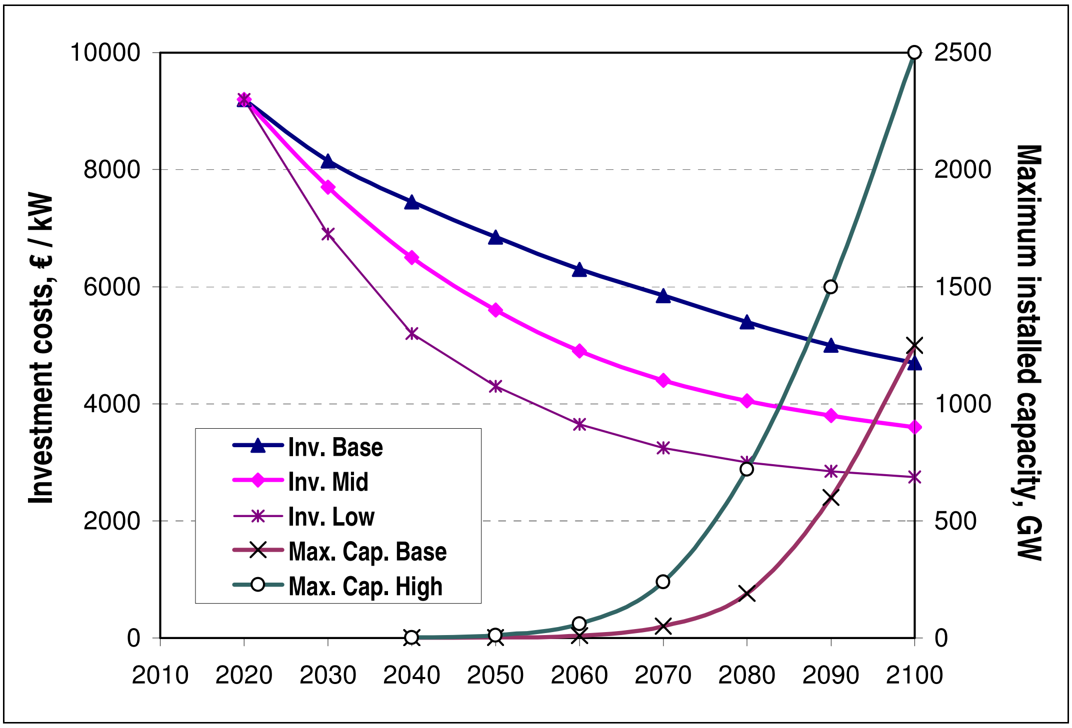


Figure 8.2: Baseline, Mid and Low scenarios for the investment costs, and Baseline and High scenarios for the maximum potential of fusion power by 2100.

Concerning the most important factors that will influence the penetration of fusion power, the following assessments can be made:

- Climate change policies have a key impact on the position of fusion;
- The availability of fossil fuel resources (oil and gas) can have a significant impact;
- The development of CCS technologies can have a notable impact;
- The availability of uranium resources, the acceptability of fission reactors, and the development of generation IV fast breeder reactors have a strong impact on the position of fusion;
- Assumptions on the potentials of solar and wind power can also have a notable impact on fusion.

Due to the relatively high costs of fusion, future constraints on greenhouse emissions appear to be the most important incentive for the rapid introduction of fusion power. However, as there are many competing low-emission technology options, the model should include a very balanced database for all future power production options and energy resources. Moreover, due to the uncertainties involved, the analysis should be based on several different scenarios for all important influencing factors.

In view of the high importance of climate change policies on the future energy system, a full climate model was implemented into the model at VTT. The climate model includes the calculation of the atmospheric concentration of all major greenhouse gases, and the radiative forcing and global temperature changes caused by them. In addition, the TIMES model generator was also extended with tools for describing the external costs caused by the environmental impacts.

8.1.3 Main results

The main results from the project can be summarized as follows:

- Development of a full Climate Module into the modeling environment, which calculates the atmospheric concentrations and the induced radiative forcing of all greenhouse gases
- Completion of a new sub-model for Finland into the Global model
- Review of the model database as part of the collaborative tasks
- Performing preliminary runs with the model for baseline and climate change mitigation scenarios.

Due to the technical difficulties related to the modeling software and documentation, most of the work in the project had to be focused on model testing and development work. During the work it became evident that the model database still needs quite extensive improvements before the model results can be considered sufficiently balanced and reliable. Therefore, at this phase one cannot put much weight yet on the results from the preliminary model runs. During the second phase of the work in 2006, the database will be thoroughly updated and the model will be subsequently applied for balanced and versatile scenario analyses. One of the focuses in the analyses will be the assessment of the value of fusion with respect to climate change mitigation.

A technical report describing the details of the model development work, findings from the database review, together with recommendations for improvements, and the results from the preliminary scenario runs is under preparation, and the report will be publicly available in April, 2006.

9 UNDERLYING TECHNOLOGY

The Underlying Technology program encompasses activities to ensure continuity in the field of specialised staff and strengthen competence in the area of structural integrity of fusion structures and components. These studies are conducted at VTT.

The evaluation of reliable material properties is a prerequisite to structural integrity and safety analysis of fusion structures and components. This area consists of theoretical expertise and novel mechanical testing capabilities concentrating on in-reactor mechanical testing, the validation of miniature size specimens used for characterisation of post-irradiated materials, on the generation of post-irradiated material property data in simulated coolant environments and on the development of sophisticated structural and surface analysis methods.

TUT/IHA has been working for ITER divertor remote maintenance tasks since 1994. IHAs special area is water hydraulics, where IHA is the leading research institute in Europe. Development work is carried out and prototypic applications have been designed and manufactured under various tasks for DRP- test platform, Italy.
During earlier projects further development needs have been found, which do not directly meet the strategy of the EFDA program, but still are very important for final ITER remote handling devices.

9.1 Scientific objectives and milestones for the period

9.1.1 Further development of novel methods and studies on radiation effects verification of specimen size effects

Principal investigator: Seppo Tähtinen, VTT Industrial Systems

Development of loading units for in-situ dynamic loading experiments during neutron irradiation is a challenging long-term objective, which will provide unique basic data for both modelling and assessment of structural integrity. Another objective is to generate fracture mechanical data on ITER candidate materials and HIP joints by using pre-cracked single edge notched bend SEN(B) specimens, and to verify the effect of the specimen size on fracture mechanical properties in order to fully utilize the miniature specimens used in irradiation test programs. An important aspect of the method development is to test small post-irradiated specimens in simulated coolant water environment.

The present SCC models utilize electrochemical data gathered from so called "fast-straining" electrode measurements, which measure the repassivation kinetics of films ruptured in specific environments upon rapid, dynamic straining. Peak current densities of copper and brass measured in this manner are orders of magnitude higher than those recorded for unstrained surfaces. Crack growth rates calculated using such "fast straining" current density data in a slip-step dissolution model correlate well with the crack growth rates measured in slow strain rate (SSRT) experiments at relatively fast strain rate (>10-6s-1). However, the crack growth rate data obtained with very slow SSRT of copper demonstrate that EAC crack growth rate does not approach zero, even if the number of film rupture events becomes very small. Thus, assuming that dissolution is an essential part of EAC, there must be additional sources of dissolution current than slip-step dissolution in order to account for the crack growth rates measured in very slow strain rate or static conditions.

The effects of strain rate on interactions between copper and its oxide films have been studied. The electrochemical oxidation process of copper is accompanied by the generation of vacancies in the copper substrate. Diffusion of vacancies from the oxide/metal interface or annihilation of vacancies by dislocation reactions is essential for oxidation to continue. Sufficiently slow straining without breaking the passive film on copper leads to a re-arrangement of the dislocation sub-structure at the interface, which helps to consume the oxidation-generated vacancies. The balance of slow straining and oxidation/dissolution produces environmentally-enhanced plasticity in the copper substrate, and simultaneously, accelerated corrosion. These conditions occur as long as the strain rate is low, i.e., on order of less than 10^{-8} s^{-1}.

10 OTHER ACTIVITIES

10.1 Conferences, Workshops and Meetings

P. Kåll participated in the ENEN Nuclear Reactor Theory course, SCK-CEN, Mol, Belgium, 12 – 29 January, 2005.

S. Tähtinen participated in the project meeting on Long Term Programme, EFDA Close Support Unit, Garching, Germany, 13 January 2005.

M. Airila, J. Likonen and J. Lönnroth participated in the Task Force E Workshop, Lausanne, Switzerland, 18 – 21 January, 2005.

S. Tähtinen participated in the project meeting on Cu/SS joints, CEA Grenoble, France, 19 January 2005.

S. Tähtinen and P. Moilanen participated in the project meeting on in-situ fatigue testing, SCK-CEN Mol, Belgium, 20 – 21st January and 24 May 2005.

T. Kiviniemi and J. Lönnroth, M. Nora, K. Rantamäki and T. Tala participated in the Task Force T Workshop, JET, UK, 24 – 28 January, 2005.

S. Karttunen and J. Likonen participated in the JET first-wall project meeting, JET, UK, 24 January, 2005.

S. Karttunen participated in the Dinner Debate "Energy for the coming generations; the role of fusion in the future energy mix" at the European Parliament, 25 January, 2005. The debate was organised by the FOM Association and sponsored by Dorette Corbey (MEP).

T. Tala participated in the Task Force Leader preparation meeting for JET 2005 Campaigns, JET, UK, 31st January – 4 February, 2005.

25 participants participated in the Diagnostics Workshop, Espoo, Finland, 31 January, 2005.

O. Dumbrajs participated in the 13th Saratov Winter School-Seminar on Microwave Electronics and Radiophysics and gave an invited plenary lecture, Saratov, Russia, 31 January – 5 February 2005..

A. Timperi, J. Järvenpää participated in the DTP2 project meeting, EFDA-CSU Garching, Germany, 9 – 10 February, 2005.

M. Airila, T. Ahlgren, K. Heinola, K. Henriksson, V. Hynönen, T. Ikonen, N. Juslin, J. Likonen, S. Karttunen, T. Kurki-Suonio, P. Kåll, K. Nordlun, T. Renvall, R. Salomaa P. Träskelin and E. Vainonen-Ahlgren participated in the The First German-Finnish Workshop on Carbon Migration in Fusion Devices: Measurements and Modelling, Tervaniemi, Finland, 14 – 15 February, 2005.

K. Rantamäki participated in the Task Force H Workshop, Cadarache, France, 20 – 23 February, 2005.

S. Janhunen participated in the SciDAC Workshop on Plasma Turbulence, Laguna Beach, California, USA, 21st – 23rd February, 2005.

S. Karttunen participated in the 5th Public Information Group meeting, Cadarahce, France, 2nd – 3rd March, 2005.

A. Timperi participated in the VR Workshop, Halden Norway, 2nd – 3rd March, 2005.

A. Lehtilä participated in the Energy Modelling Experts Workshop, EFDA Model Steering Group Meeting, Madrid, Spain, 8 – 11 March, 2005.

N. Juslin participated in the 7th Workshop on Multiscale Modelling of Fe-Cr Alloys for Nuclear Applications, SCK-CEN Headquarters, Brussels, Belgium, 14 – 15 March, 2005.

S. Karttunen participated in the ITER-like Wall Project Board meeting, UKAEA / JET, UK, 16 March, 2005.

T. Ahlgren, M. Airila, J. Heikkinen, K. Henriksson, V. Hynönen, S. Janhunen, S. Karttunen, T. Kiviniemi, T. Kurki-Suonio, P. Kåll, J. Likonen, P. Moilanen, M. Nora, K. Rantamäki, S. Saarela, A. Salmi, R. Salomaa, S. Sipilä, T. Tala and E. Vainonen-Ahlgren participated in the XXXIX Annual Conference of the Finnish Physical Society, Espoo, Finland, 17 – 19 March, 2005. For the first time a Fusion and Plasma Physics Session (Oral and Posters) was organised. In addition to 20 presentations in the posters session, oral presentations were given by S. Karttunen, J. Heikkinen, K. Henriksson, P. Kåll, S. Sipilä and T. Tala.

K. Nordlund participated in the Plasma-Surface Interactions meeting in Oak Ridge, Tennessee, USA, 18 – 24 March, 2005.

K. Rantamäki participated in the WIN Global Annual meeting, Prague, 4 – 8 April, 2005.

J. Lönnroth participated in the 10th US-European Transport Task Force Workshop, Napa, California, USA, 6 – 9 April 2005.

J. Heikkinen participated in the Magnum-psi Ad Hoc Group Meeting at FOM-Institute for Plasma Physics, Rijnhuizen, Netherlands, 11 – 12 April, 2005.

V. Kujanpää and T.Jokinen participated in the European Laser Applications Network, Lappeenranta, 9 – 10 May 2005.

A. Timperi participated in the Tampere Crossing, Tampere, Finland, 17 – 18 May, 2005.

V. Kujanpää participated in the Fibre Laser seminar, BIAS, Bremen, Germany, 24 May 2005.

Over 70 participants participated in the FUSION 2005 - Annual Seminar of the Association Euratom-Tekes, Paasitorni, Helsinki, Finland, 30 – 31 May, 2005. Invited speakers were A. Peacock and Lawrence Jones from EFDA CSU Garching.

L. Annila, S. Karttunen, J. Mäkinen and A. Timperi participated in the Fusion Industry Workshop, Madrid, Spain, 1st – 2nd June 2005.

M. Siuko and J. Järvenpää participated in the Working meeting in EFDA, Garching, Germany,, 8 – 9 June 2005.

V. Kujanpää and T. Jokinen participated in the Conference on Lasers in Maufacturing and Laser Fair, München, Germany, 13 – 16 June 2005.

K. Nordlund participated in the EFDA TW5-TTMS-007 task meeting, Garching, Germany, 27 – 30 June 2005.

S. Henriksson, T. Kiviniemi, T. Kurki-Suonio, K. Rantamäki, S. Sipilä, T. Tala participated in the 32nd EPS Conference on Controlled Fusion and Plasma Physics, Tarragona, Spain, 27 June – 1st July, 2005.

A. Timperi participated in the Power-Gen, Milano, Italy, 28 – 30 June, 2005.

V. Kujanpää and T. Jokinen participated in the International Institute of Welding, Prague, Check Republic, 10 – 14 July 2005.

F. Wasastjerna participated in the ITER neutronics Meeting with EU, 14 July, 2005.

F. Wasastjerna participated in the Discussion on Implementation of ITER/EFDA Neutronics Task ITA 73-07, 15 July, 2005.

S. Henriksson participated in the 42nd Culham Plasma Physics Summer School, Culham, UK, 18 – 29 July, 2005.

K. Nordlund participated in the International Symposium on Plasma Chemistry, Toronto, Canada, 8 – 12 August, 2005.

V. Kujanpää participated in the Nordic Laser Materials Processing Conference (NOLAMP 10), Piteå, Sweden, 17 – 19 August 2005.

R. Salomaa participated in the 12th International Conference on Emerging Nuclear Energy Systems, Brussels, Belgium, 21st – 26 August, 2005.

S. Janhunen, F. Ogando and V. Hynönen participated in the Carolus Magnus Summer school, Mechelen, The Netherlands, 4 – 16 September, 2005.

S. Karttunen participated in the Follow-Up Meeting for the EU Integration of New and Recent Partners in the Euratom Fusion Programme in Garching, Germany, 12 – 13 September, 2005.

V. Kujanpää participated in the Welding and Cutting Fair, Essen, Germany, 12 – 14 September, 2005.

N. Juslin participated in the 8th Workshop on Multiscale Modelling of Fe-Cr Alloys for Nuclear Applications, KTH, Stockholm, Sweden, 15 – 16 September 2005.

T. Tala participated in the Workshop on real-time control, JET, UK, 19 – 20 September, 2005.

H. Mäkinen, M.Siuko and S. Verho participated in the Delmia Worldwide user conference 2005, Fellbach, Germany, 19 – 20 September, 2005.

K. Rantamäki, M. Santala and T. Tala participated in the Task Force S2 Workshop, JET, UK, 19 – 23 September, 2005.

T. Kurki-Suonio participated in the "Säteilevät Naiset" seminar, Helsinki, Finland, 22 September, 2005.

J. Heikkinen participated in the European Fusion Theory Conference, Aix-en-Provence, France, 25 – 29 September, 2005.

P. Kåll participated in the IPP Summer university for plasma physics, Greifswald, Germany, 25 – 30 September, 2005.

A. Timperi participated in the Meeting in the European Parliament, Strasbourg, France, 27 September, 2005.

T. Kiviniemi participated in the 10th IAEA Technical meeting on H-mode Physics and Transport Barriers, St. Petersburg, Russia, 28 – 30 September, 2005.

S. Karttunen participated in the Millenium New Materials Seminar, Mikkeli, 29 – 30 September, 2005.

A. Lehtilä participated in the Energy Modelling Experts Workshop, EFDA Model Steering Group Meeting, Garching, Germany, 4 – 7 October, 2005.

V. Kujanpää participated in the Vacuum Vessel Meeting, Magdeburg, Germany, 4 – 6 October, 2005.

S. Tähtinen and P. Moilanen participated in the project meeting on Cu research, Risø National Laboratory, Roskilde, Denmark, 7 October 2005.

T. Kurki-Suonio, E. Vainonen-Ahlgren participated in the AUG Rinberg seminar, Rinberg, Germany, 9 – 14 October, 2005.

J. Likonen participated in the ITER-like Wall Project Board meeting, UKAEA / JET, UK, 12-13 October, 2005.

J. Likonen participated in the EU-PWI TF Task Force meeting, Cadarache, France, 17 – 19 October, 2005.

J. Heikkinen participated in the 10th International Workshop on Plasma Edge Theory in Fusion Devices, Jülich, Germany, 17 – 19 October, 2005.

J. Lönnroth participated in the 10th International Workshop on Plasma Edge Theory in Fusion Devices, Jülich, Germany, 17 - 19 October 2005 and gave an invited talk.

M. Siuko and J. Mattila participated in the Collaboration meeting in JET, Culham, UK, 21st October 2005.

V. Kujanpää, A. Jansson and H. Pantsar participated in the International Conference on Lasers and Electro-Optics, ICALEO'05, Miami, Florida, USA, 31st October – 3rd November, 2005.

S. Sipilä participated in the Kickoff Meeting for ITER ICH Design Tasks ITA–51-03 and ITA-51-05, Garching, Germany, 13 December 2005.

M. Siuko participated in the ITER - Opportunities for European Industry Workshop – Barcelona, 13 – 14 December, 2005.

J. Lönnroth participated in the 13th European Fusion Physics Workshop, Montegrotto Terme, Italy, 14 - 16 December 2005.

J. Poutanen participated in the Multilink Sleeved Pin meeting, Garching, Germany, 15 December, 2005.

10.2 Visit

J. Lönnroth was seconded to EFDA-JET under JOC, Culham, UK, on 1st January – 31st December, 2005.

M. Santala was seconded to EFDA-JET under JOC, Culham, UK, on 1st January – 31st December, 2005.

T. Kiviniemi visited EFDA-JET under S/T Order, on 16 January – 12 February, 2005.

T. Tala visited EFDA-JET under S/T Order, on 15 May, 2005 – 5 March, 2006.

J. Likonen visited EFDA-JET under S/T Order, on 2nd – 24 July, 2005.

J. Likonen visited EFDA-JET under S/T Order, on 24 – 28 October, 2005.

K. Rantamäki visited EFDA-JET under S/T Order, on 7 November – 9 December, 2005.

M. Nora visited EFDA-JET under S/T Order, on 5 – 16 December, 2005.

T. Kurki-Suonio visited IPP-MPG,Garching, Germany, on 17 February – 2nd May, 2005.

M. Airila visited IPP-Garching, Germany, on 25 April – 1st June, 2005.

F. Wasastjerna visited FZK Karlsruhe, Germany under the Staff Mobility agreement, on 19 June – 30 July and 17 October – 27 November, 2005.

T. Kurki-Suonio visited IPP-MPG,Garching Germany, on 11 July – 2nd August, 2005.

V. Hynönen visited IPP-MPG, Garching Germany, on 11 July – 9 August, 2005.

T. Kurki-Suonio visited IPP-MPG,Garching Germany, on 5 – 18 October, 2005.

J. Lönnroth visited General Atomics, San Diego, California, USA, on 7 - 11 November 2005.

V. Kujanpää visited Fraunhofer Institute ILT, on 16 – 17 March 2005.

L. Annila, M. Siuko, A. Timperi visited Olkiluoto Nuclear Power Plant, on 6 April, 2005.

J. Likonen and E. Vainonen-Ahlgren visited IPP-Garching, Germany, on 18 May, 2005.

J. Ihamuotila (Finnish Academies of Technology), I. Andersson (Prizztech), L. Annila (Prizztech), K. Asikainen (Ekono), K. Happonen (Aamulehti), M. af Heurlin (Tekes), S. Immonen (Ministry of Trade and Industry), B. Karlemo (Outokumpu), S. Karttunen (VTT) visited CEA Cadarche and the ITER site, on 23 May, 2005.

V. Kujanpää visited BIAS, Bremen, Germany, on 24 May 2005.

V. Kujanpää visited SLV Rostock, Germany, on 25 May 2005.

N. Juslin visited SCK-CEN in Mol, Belgium, on 6 – 10 June, 2005.

J. Järvenpää and M. Siuko visited EFDA Garching, Germany, on 8 – 9 June, 2005.

V. Kujanpää visited Force Institute, Brondby, Denmark, on 8 June 2005.

J. Lönnroth visited University of Innsbruck, Innsbruck, Austria (on behalf of JOC), on 14 – 22nd June 2005.

V. Kujanpää and T. Jokinen visited EFDA, Garching, Germany, on 15 June 2005.

M. Airila visited FZJ Jülich, Germany, on 4 – 8 July, 2005.

E. Vainonen-Ahlgren visited IPP- Garching, Germany, on 29 – 30 July, 2005.

M. Siuko, J. Mattila, H. Koivisto, A. Timperi visited Oxford Technologies, on 22 – 23 September, 2005.

A. Timperi visited Moog Control Systems, Stuttgart, Germany, “Controls for DTP2 + other issues”, on 28 September, 2005.

J. Lönnroth visited Japan Atomic Energy Agency, Naka, Japan (on behalf of JOC), on 24 October - 4 November 2005.

V. Kujanpää visited Edison Welding Institute and Ohio State University, Columbus, Ohio, USA, on 28 October 2005.

M. Siuko visited JET, Culham, UK, on 10 November 2005.

10.3 Visitors

Keith Barnes from SPI and Patrick Brandelind from BFi OPTiLAS visited VTT Industrial Systems on 20 January 2005.

David Campbell and Christian Ingesson from EFDA CSU, Garching visited VTT Processes on 31st anuary 2005.

Dietmar Wagner from Lasag GmbH visited VTT Industrial Systems on 31st January – 1st February 2005.

Lawrence Jones and Michael Irving from EFDA CSU, Garching visited VTT and Lappeenranta University of Technology on 3 – 4 February 2005.

Prof. William Steen from University of Liverpool visited Lappeenranta University of Technology on 15 February 2005.

Michael Pick, Jim Palmer from EFDA CSU, Garching visited VTT Industrial Systems and Tampere University of Technology on 22nd – 23rd February 2005.

Matej Mayer from Max-Planck-Institute für PlasmaPhysik in Garching visited Accelerator Laboratory, University of Helsinki, on March – May, 2005.

Lawrence Jones from EFDA CSU, Garching visited Lappeenranta University of Technology on 12 April, 2005.

Paul Coad from JET/UKAEA visited VTT Processes on 18 – 29 April, 2005.

Udo von Touissaint from Max-Planck-Institute für PlasmaPhysik in Garching visited Accelerator Laboratory, University of Helsinki, on 2nd – 18 May, 2005.

Lawrence Jones and Alan Peacock from EFDA CSU, Garching visited VTT on 30 – 31st May, 2005..

Michael Irving from EFDA CSU, Garching visited VTT Industrial Systems and Tampere University of Technology on 22nd June 2005.

B. N. Singh, (Risø), J. Dekeyser, P. Jacquet and G. Engelen from SCK-CEN visited VTT Industrial Systems and participated in project meeting TW4-TVM-COFAT2 In Reactor Fatigue of Copper Alloys on 17 August 2005.

K. Sugiyama from Kyushu University visited VTT Processes on 24 August – 2nd September 2005.

Dr. Tech. Alexander Andreev from S.I. Vavilov State Optical Institute visited TKK on 1st September – 30 November, 2005.

Michael Irving and Jim Palmer from EFDA CSU, Garching visited VTT Industrial Systems and Tampere University of Technology on 1st – 2nd September, 2005.

Arkadi Kreter from FZJ Jülich visited VTT Processes on 4 – 10 September.

I.G. Tigelis from University of Athens visited TKK on 1 – 15 October, 2005.

Z. Ioannidis from University of Athens visited TKK on 1 – 15 October, 2005.

Henri Weisen from CRPP-Lausanne visited VTT Processes on 10 – 11 October 2005.

G. Engelen from SCK-CEN visited VTT Industrial Systems on 11 October 2005.

Alan Rolfe from Oxford Technologies Ltd, England visited TUT/IHA on 14 October 2005.

Paul Coad from JET/UKAEA visited VTT Processes on 7 – 18 November 2005.

V. Lutsenko from Kiev Institute for Nuclear Research visited Helsinki University of Technology on 21 November - 5 December 2005.

10.4 Staff Mobility Reports

10.4.1 Simulations of methane puffing experiments of ASDEX Upgrade with ERO code

Name of seconded person:	Markus Airila.
Sending Institution:	TKK, Helsinki University of Technology Association Euratom-Tekes
Host Institution:	Max-Planck-Institut für Plasmaphysik, Garching
Dates of secondment / Mission:	24 April – 3 June, 2005

Work Plan / Milestones

The Monte Carlo code ERO has been developed at IPP and FZJ for simulation of impurity transport in the scrape-off layer of magnetically confined plasmas. Since 2004, TKK has participated in the development of ERO in collaboration with FZJ. While FZJ takes the main responsibility of JET-related ERO simulations, TKK focuses on simulating ASDEX Upgrade. In particular, we have been developing a code version which can simulate divertor experiments in 3D for the first time.

ERO was originally developed for limiter experiments and is intrinsically a 3D code, but in all divertor simulations so far a version reduced to 2D has been used. We have modified the code and it is now possible to simulate also divertor-type experiments in three dimensions.

Our present interest lies in a methane puffing experiment performed in ASDEX Upgrade which have been analyzed using nuclear reaction analysis techniques. This experiment included several identical H-mode discharges at the end of a campaign with $^{13}CH_4$ puffing in the divertor region. The deposition pattern of the marker ^{13}C on five tiles in the vicinity of the puffing valve was found after the tiles were removed from the tokamak. Because the gas puff occurs in a single position, the deposition varies both in the toroidal and poloidal direction over the divertor surface. Therefore, full 3D simulation is required. The simulations will be of great value for benchmarking the ERO code with experiments.

Another set of simulations is related to our experimental proposal which was accepted in the ongoing programme. We attempt to identify which plasma parameters are important for the impurity transport. To this end, a set of experiments with varying parameters will be performed. The proposed experiments also serve as benchmarking of the 3D divertor version of ERO.

If time permits, discussion about kinetic modelling of electrons in the scrape-off layer of ASDEX Upgrade plasmas will be integrated in the present visit. There has been an informal proposal from IPP to TKK to carry ACSOT simulations on the subject. We have started the work but only very preliminary results have been obtained. The use of realistic ASDEX Upgrade plasma backgrounds and cross-field particle fluxes requires more work.

Report

The testing phase of the 3D divertor version of ERO code was completed and first simulations with proper numerical resolution were carried out. They focused on the methane puffing experiments from which an NRA measured 2D deposition pattern is available for benchmarking. On the basis of these simulations the following conclusions can be drawn:

- 3D simulations with good statistics can be run in reasonable time (1 day).
- The distance travelled by carbon particles is rather well reproduced.
- E×B drift, which is supposed to be responsible for the experimentally observed shape of the deposition pattern, is not properly reproduced by ERO using the presently available plasma background, which has density and temperature very different from the actual experiment.

A plasma background from e.g. B2/Eirene codes, which matches well all avilable diagnostics in the divertor region, is necessary for useful simulations. A background which better matches the divertor diagnostics was elaborated with David Coster, but there are still additional data that need to be taken into account in the B2/Eirene simulation. Since the demand for background simulations is high, the use of DIVIMP code with its onion-skin model for generating plasma backgrounds for ERO independently in the future was discussed. Karl Krieger gave an introduction to DIVIMP, and the user permissions for M. Airila were set.

A seminar talk was given on the background, present status and future plans of ERO modelling work related to ASDEX Upgrade.

The issues related to development of ERO were discussed several times with Andreas Kirschner (FZ Jülich), and it was agreed that a one-week meeting will be arranged in July at FZJ to combine the enhancements to ERO completed over the past few months at TKK and FZJ.

ASCOT simulations of suprathermal electrons were discussed, and importing a suitable AUG SOL background to ASCOT was identified as the next step in the simulations.

The proposed experiments have been cancelled for two reasons:
1) The benchmarking of ERO with experiments is at the moment more dependent on suitable background plasmas than on new experimental data.
2) The experimental schedule on AUG is extremely tight due to several breaks in the present campaign.

10.4.2 Fast Particles in AUG Edge Region and Benchmarking ASCOT against NPA measurements

Name of seconded person: Taina Kurki-Suonio
Sending Institution: Helsinki University of Technology
Association Euratom-Tekes
Host Institution: IPP-MPG –Garching / Association Euratom-IPP
Dates of secondment / Mission 17/2 – 2/3/2005 and 11/7 – 2/8/2005

Work Plan / Milestones

1st task: benchmarking ASCOT results against the NPA (Neutral Particle Analyzer) and/or fast particle detector measurements.
2nd task: participate in the planning of the 2005 reversed I_p –campaign.
3rd task: time permitting, is to start testing the new 3-dimensional ripple model, recently implemented in ASCOT, against ASDEX Upgrade backgrounds.
4th task: integrate fast ion distributions to the relevant edge MHD stability codes

Report

1st task: reviewing last year's NPA measurements with U. Fahrbach revealed that these data missed the low energy channels needed for the neutral density. Therefore we designed a set of new shots, tailored for ASCOT benchmarking. These shots were carried out shortly after my visit (March 8). With M. Munoz we discussed ways to incorporate the fast particle detector in ASCOT, but due to difficulties in its geometry, this plan was set aside for now.

2nd task: Due to problems with the new control system it is not clear when the reversed I_p –campaign will take place this year → efforts centered in the shots mentioned above.

3rd task: the new 3-D ripple model is under testing for JET, which only includes ripple in B_T-component. AUG ripple is more realistic and demanding and needs more work.

4th task: Interfacing ASCOT with HAGIS was discussed with S. Pinches. It was decided that he will first investigate the possibility of extending HAGIS to the regions outside LCFS.

Additionally: A) test the validity of various assumptions in the SOLPS-calculations: ASCOT simulations of kinetic electrons were prepared (D. Coster and B. Scott), and
B) The study on the effect of a homogeneous radial electric field on NBI-ion orbits was finished.

10.4.3 IFMIF Test Facilities: Update of the MCNP model and shielding analysis of the ceiling

Name of seconded person: Frej Wasastjerna
Sending Institution: VTT Processes
Association Euratom-Tekes
Host Institution: Forschungszentrum Karlsruhe, Institut für Reaktorsicherheit
Dates of secondment / Mission 19 June – 30 July 2005, 6 weeks

Objectives
To perform shielding calculations on the test cell cover and determine fluxes and dose rates at selected locations in the access cell with an additional horseshoe-type neutron shielding block present inside the test cell.

Results
The horseshoe shield was incorporated in the model as planned. The actual calculations to determine the effectiveness of the horseshoe shield were done by Simakov.

Calculations of the dose rates in the access cell are ongoing at the time of writing. For more information, see section 3.2.14.

10.4.4 Study of the effect of magnetic ripple and radial electric field on the confinement of fast particles using ASCOT code

Name of seconded person:	Ville Hynönen.
Sending Institution:	Helsinki Univ. of Technology, Association Euratom-Tekes
Host Institution:	IPP-MPG / Association Euratom-IPP
Dates of secondment / Mission	11/7 – 9/8/2005

Work Plan / Milestones
The effect of a radial electric field is generally considered to arise only from its homogeneity, but even a constant electric field can affect NBI-born ions. Another important factor in the behaviour of fast ions is the finite toroidal ripple. Using the three-dimensional ripple model that has been recently implemented in ASCOT it is possible to study both effects separately or simultaneously.

During my visit, in collaboration with the experimental group, I will continue the analysis of the behaviour of fast ions in the presence of finite toroidal ripple and realistic radial electric field. Both counter- and co-injected neutral beams will be investigated. The ripple simulations also serve as a benchmark for the new model. Some of the time will be reserved for code development, specifically for the improvement of the ASCOT output distributions.

A secondary goal is to write better post-processing routines for the visualization of ASCOT results. Especially the relatively new four-dimensional particle distribution in spatial and velocity-space coordinates requires special care.

Report
Several simulations with and without ripple were carried out for co- and counter-injection discharges. For counter-injected particles the ripple has hardly any effect and even for co-injected particles there were no qualitative change in the results. The ripple model was also benchmarked against both other calculations and experimental measurements with Dr. Dux. However, including a realistic, non-uniform radial electric field in the simulation turned out to be more complicated than expected. The radial electric field is not available as a standard diagnostic, and to calculate it from other available data is a time-consuming process. With Dr. Conway the work on

importing the experimental radial electric field to ASCOT was initiated and is still ongoing. Considerable effort was put to improving the ASCOT output distributions and to other code maintenance work. This was fruitful, since as a result the most important output distributions are now in physical units.

There were more pressing matters than writing visualization routines, therefore the secondary goal had to be abandoned.

Additional tasks:
We agreed with Dr. Tardini that we would simulate ripple-induced fast ion losses for an internal transport barrier discharge, and some preliminary simulations were already carried out. Benchmarking of ASCOT against ASTRA, TRANSP, and FAFNER codes was also discussed.

With Drs. Coster and Chankin we solved a few problems related to the ASCOT simulations of kinetic electrons near the divertor targets, a project initiated at HUT earlier in the summer to facilitate interpreting and improving SOLPS calculations. ASCOT can now import the 2-dimensional SOL backgrounds and interpolate them at sufficient accuracy for reliable simulations.

The exporting of ASCOT simulation results to a MDSplus database, another project initiated at HUT earlier this summer, was discussed with Drs. Coster and Suttrop. Their advice were very useful, and presently we have a MDSplus server which can be used to store simulation results.

Documentation about the ASCOT output distributions and the new 3D ripple model was written.

10.4.5 ITER Equatorial Port Shielding

Name of seconded person:	Frej Wasastjerna
Sending Institution:	VTT Processes Association Euratom-Tekes
Host Institution:	Forschungszentrum Karlsruhe, Institut für Reaktorsicherheit
Dates of secondment / Mission	16 October – 26 November 2005, 6 weeks

Objectives
To complete the previous calculations for the HCPB-TBM and to clarify issues related to the effect of streaming in the gap between equatorial port plugs and port walls in ITER, especially how the detailed design in the vicinity of the dogleg in this gap affects the streaming. Generic questions about how to calculate streaming with the Monte Carlo method may also be addressed if the available time and funding permits.

Results
Previous HCPB-TBM calculations have been completed and the results have been described in the report "Shielding Calculations for a Helium Cooled Pebble Bed Test Blanket Module in ITER".

10.4.6 Erosion/deposition and material transport studies under JET Task Force Fusion Technology

Name of seconded person:	Jari Likonen
Sending Institution:	VTT Processes, Association Euratom-Tekes
Host Institution:	UKAEA / EFDA JET
Dates of secondment / Mission	4/7/05 – 22/7/05 and 24/10/05-28/10/05

Work Plan / Milestones

Erosion/deposition and material transport at JET have been investigated using various surface analytical techniques under Task Force Fusion Technology. This work has been done in close collaboration with JET/UKAEA. In 2001 Tekes coated a poloidal set of divertor tiles with W marker and these tiles were removed in 2004. Surface analysis of tiles is now underway at Tekes. The purposes of the visit are joint experiments under JET Task Force Fusion Technology task JW5-FT-3.24.

Visit: 4/7/05-22/7/05:

- evaluation and comparison of SIMS, TOF-ERDA and RBS results
- determination of erosion/deposition pattern at divertor tiles exposed in 2001-2004
- study ^{13}C migration in SOL
- calculation of ion fluxes at divertor for 2001-2004 campaigns

Visit: 24/10/05-28/10/05

- ion beam analyses of JET divertor tiles
- calculation of RBS spectra
- preparation of PSI2006 abstract

Report

Visit 4/7/05-22/7/07: Tungsten coated marker tiles made at Tekes in 2001 were removed in 2004 shutdown. During the staff mobility visit outer divertor tiles 7 and 8 were analysed with ion beam techniques at the Univ. of Brighton. RBS spectra were simulated using SIMNRA program and the coating thicknesses and compositions were calculated (Milestones 1 and 2).

Prior to the 2004 shutdown, an experiment was devised to provide specific information on material transport and SOL flows observed at JET. $^{13}CH_4$ and $B(CH_3)_3$ were injected into the plasma boundary from the outer divertor in the last day of discharges using one type of discharge only. The purpose of the experiment was to determine how the ^{13}C and boron are transported around the SOL, and where they are eventually deposited. The amount of puffed ^{13}C was about 3 times bigger than during the previous ^{13}C puffing experiment in 2001. The amount of boron was comparable to the amount of puffed Si in 2001. Boron was detected spectroscopicly at the outer divertor. The puffing experiment at JET was very successful. All milestones for this visit were achieved (Milestone 3).

Post-mortem surface analysis has shown that there is ^{13}C deposited on the collector probe and that there is more ^{13}C on the side facing the outer divertor. Divertor tiles 1, 3, 4, 7 and 8 have been analysed with RBS, NRA and SIMS. Surface analyses indicate that there is quite low amount of ^{13}C on the divertor tiles (23% of total input).

There is more ^{13}C on the outer divertor tiles than on the inner ones. Outer floor tile 6 has not yet been analysed. SIMS and RBS analyses agree well with each other (Milestone 3).

New marker coatings (C,W) for erosion/deposition studies will be installed at JET during 2004 shutdown. Some test samples and outer poloidal limiter tiles have been analysed with Rutherford backscattering technique (RBS) at the University of Sussex. RBS spectra were simulated using SIMNRA program and the coating thicknesses and compositions were calculated (milestone 1).

All surface analysis results (SIMS, RBS, TOF-ERDA) for JET tiles exposed in 1998-2001 and 1999-2001 were compiled during the visit and an overview paper on the results is preparation (milestone 1 and 2).

Other activities:
During 2004 staff mobility visit some of the coated tiles were measuredusing a surface profiler developed at JET. The purpose of the measurements is to measure the surface profile of the coated area before the tiles are installed at JET. After the tiles will be removed the measurements will be repeated. By comparing the results before and after plasma exposure, erosion/deposition pattern can be determined. The engineer who developed the system is retiring in August 2005 and at the moment there are no plans who will continue the work. Therefore some time was spent with the system and a test tile was measured with the surface profiler.

All the milestones for this visit were achieved.

Visit 24/10/05-28/10/05: During the staff mobility visit core samples from W coated inner divertor tiles 1, 3 and 6 were analysed with ion beam techniques at the Univ. of Brighton. Simulation of RBS spectra using SIMNRA program has been started during the staff mobility visit and will continue at VTT. As a result of the SIMNRA simulations the coating thicknesses and compositions will be obtained (Milestones 1 and 2).

All the milestones for this visit were achieved.

10.4.7 In-reactor fatigue investigations of copper alloys

Name of seconded person:	Sami Saarela
Sending Institution:	VTT Industrial Systems Association Euratom-Tekes
Host Institution:	SCK-CEN Mol, Belgium
Dates of secondment / Mission	20-28 October, 7-10 November and 21 November – 5 December 2005, 28 days

Background and objectives
The knowledge on the effects of irradiation on material behaviour is based on tests that have been conducted with materials which have first been irradiated and then tested. This kind of testing method does not correspond to the irradiation and loading environment that materials are exposed to in fission or fusion reactors. VTT's patented pneumatic material testing systems have been applied for in-reactor tensile testing in earlier tasks in collaboration with SCK-CEN and Risø National Laboratory. The in-reactor tensile results indicated that simultaneous irradiation and straining

strongly affects deformation behaviour of pure copper compared to unirradiated or post-irradiated conditions.

The objective of the present task was to obtain information on the effect of simultaneous irradiation and creep-fatigue type of loading on fatigue life of copper alloy. The studied material was prime aged CuCrZr alloy which is a candidate heat sink material for ITER first wall and divertor structures. Two strain controlled creep fatigue tests were conducted in the core of the BR-2 test reactor in Mol using VTT's pneumatic control system and fatigue loading modules. The creep effect was introduced by applying 10s and 100s holding times at maximum compression and tension.

First visit
The two pneumatic fatigue modules were tested and calibrated at VTT just before the first visit. During the first visit the modules were connected to the irradiation rig. After the leakage test the rig was transported to the BR-2 reactor building and connected to the computers and the helium gas supply loop. Different PID-values were tested and most suitable ones were selected. The control program was tested and modified. Out of reactor reference tests were started with both modules while the rig was hanging in the reactor building.

Second visit
When the reference tests were over the rig was transported out of the reactor building. During the second visit the two fatigue modules were completely disassembled in order to change the tested specimens to new ones. When the modules were ready, they were connected to the rig. After that, all instrumentation wires and pipes were spot welded to the modules to prevent problems caused by the cooling flow inside the reactor. Then the rig was transported back to the reactor building.

Third visit
During the third visit the modules, all the connections and the control and data acquisition programs were checked in the reactor building. The effect of the cooling water flow on the displacement transducers was tested and found negligible. During the start-up of the reactor, the rig was quickly inserted into the core to test its effect on the reactivity of the core. When the reactor was at full 60 MW power and the neutron flux had stabilised, the cooling water flow was started and the rig was slowly lowered into the core. The temperature of the modules stabilised to approximately 70 and 90 ºC in ten minutes and then the cyclic fatigue loading was started. The fatigue tests were conducted in strain controlled mode with a strain amplitude of 0.5% with 10 and 100s holding times. The cycle time was 120s and 300s for the modules 1 and 2, respectively. Specimens lasted approximately 2500 cycles which took 4 days with 10s hold time and 10 days with 100s hold time. After the test, the rig was transported to a hot cell where the specimens were cut loose from the modules. The microstructure of the specimens will be examined later.

11 PUBLICATIONS

11.1 Fusion Physics and Plasma Engineering

11.1.1 Publications in Scientific Journals – Fusion Plasma Physics

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10. T.M.J. Ikonen and O. Dumbrajs, "Search for deterministic chaos in ELM time series of ASDEX Upgrade tokamak", IEEE Transactions on Plasma Science **33** (2005) 1115.

11. L. Knuuttila, A. Lankinen, J. Likonen, H. Lipsanen, X. Lu, P. McNally, J. Riikonen, T. Tuomi, "Low Temperature Growth GaAs on Ge", Japanese Journal of Applied Physics **44** (2005) 7777-7784.

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APPENDIX A INTRODUCTION TO FUSION

A.1 Energy Demand Is Increasing

Most projections show world energy demand doubling or trebling in the next 50 years. This derives from fast population growth and rapid economic development. Energy sources that are not yet fully tapped include biomass, hydropower, geo-thermal, wind, solar, nuclear fission and fusion. All of them must be developed to meet future needs. Each alternative has its advantages and disadvantages regarding the availability of the resource, its distribution globally, environmental impact, and public acceptability. Fusion is a good candidate for supplying base load electricity on a large scale. Fusion has practically unlimited fuel resources, and it is safe and environmentally sound.

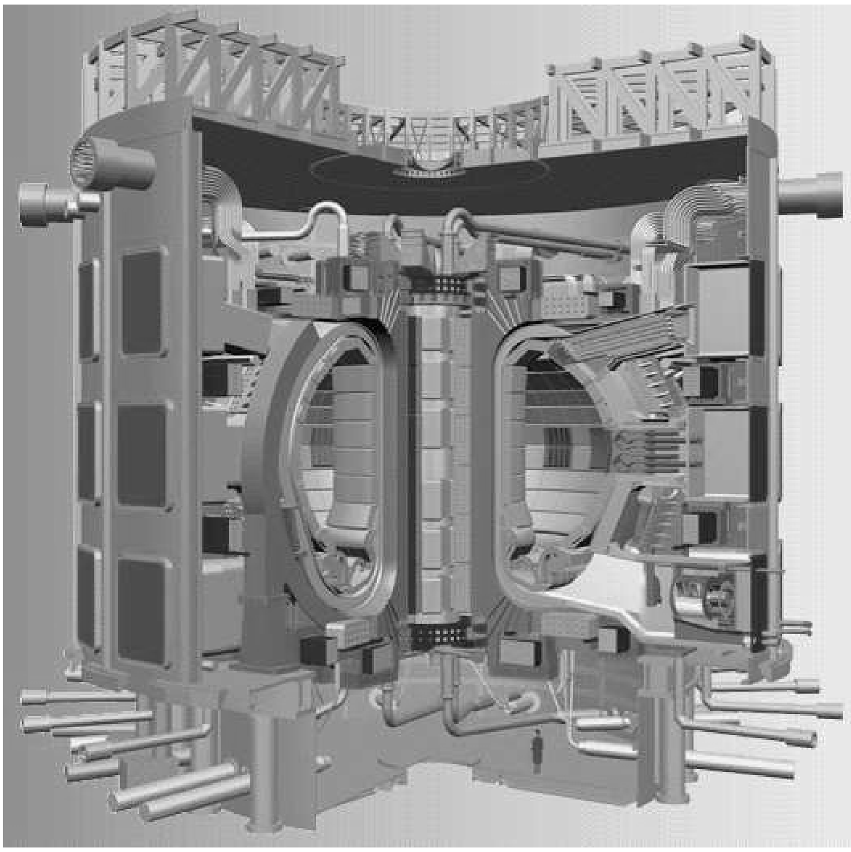

Figure 1: A design model for the experimental fusion reactor ITER.

A.2 What Is Fusion Energy?

Fusion is the energy source of the sun and other stars, and all life on Earth is based on fusion energy. The fuels burned in a fusion reactor are hydrogen isotopes, deuterium and tritium. Deuterium resources are practically unlimited, and tritium can be produced from lithium, which is abundant. The fusion reactions occur only at very high temperatures. For the deuterium-tritium reaction, temperatures over 100 million °C are required for sufficient fusion burn. At these temperatures, the fuel gas is fully ionised plasma. High temperatures can be achieved by injecting energetic particle beams or high power radio-frequency (RF) waves into the plasma. The hot plasma can be thermally isolated from the material walls by strong magnetic fields, which form a "magnetic bottle" to confine the fuel plasma. With a sufficiently large plasma volume, much more energy is released from fusion reactions than is required to heat and confine the fuel plasma, i.e, a large amount of net energy is produced.

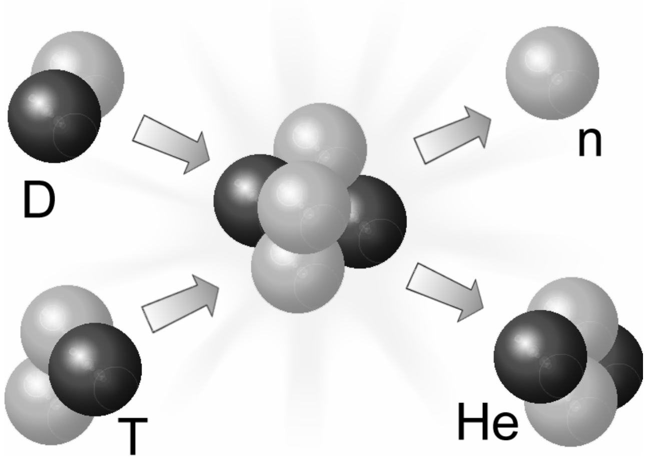

Figure 2: In a fusion reaction, deuterium (D) and tritium (T) fuse together forming a helium nucleus (^4He) and releasing a large amount of energy which is mostly carried by the neutron (n).

A.3 The European Fusion Programme

Harnessing fusion energy is the primary goal of the Euratom Fusion Progremme in the 6th Framework Programme. The reactor orientation of the programme has provided the drive and the cohesion that makes Europe the world leader in fusion research. The world record of 16 megawatts of fusion power is held by JET device, the Joint European Torus.

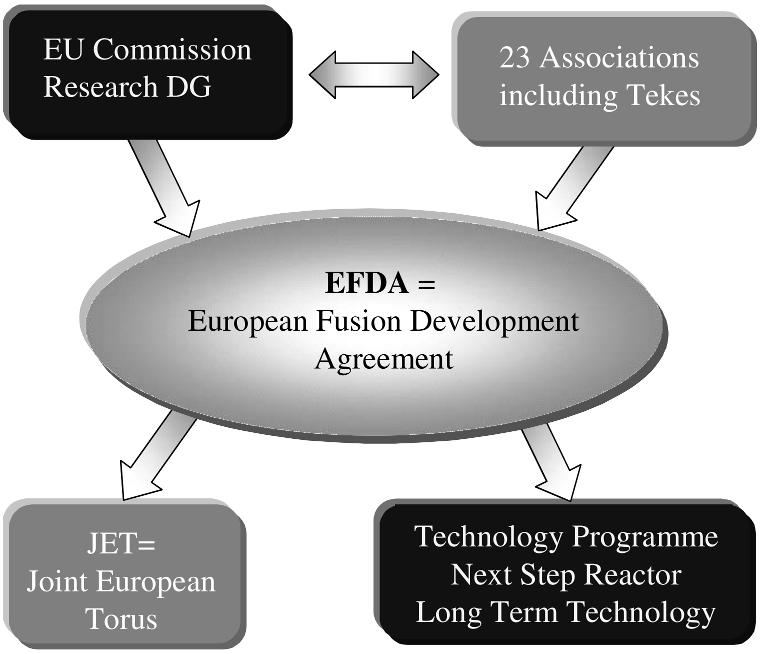

Figure 3: Multilateral EFDA covers JET research activities, fusion technology work for ITER and long term technology development. Tekes is one of the twenty three Euratom Associations.

Euratom Fusion Associations are the backbone of the European Fusion Programme. There are 23 Associations from the EU countries, Switzerland, Romania and Bulgaria. The multilateral European Fusion Development Agreement (EFDA) between all the Associations and Euratom facilitates the joint exploitation of the JET facilities and the fusion technology programme, which covers ITER related R&D and long-term technology.

A.4 ITER International Fusion Energy Organisation

To advance significantly beyond the present generation of fusion devices, a next step device, enabling the investigation of burning plasma in near-reactor conditions, is needed. This will be done in the global ITER project ("iter" is "way" in latin), which is the joint project of EU, Japan, Russian Federation, United States, China, India and South Korea. The ITER parties agreed in 2005 to site ITER in Europe (Cadarache, France). The detailed design and the extensive technical preparations have been completed and permitting to start the first procurements for ITER construction in 2006 - 2007. ITER parties have agreed in the most issues and the final decision to start the construction is expected to take place soon. ITER tokamak would be the smallest tokamak enabling an investigation of burning plasmas at fusion power levels of 400-500 megawatts with energy amplification exceeding 10. Latest results from various tokamaks indicate that even larger amplification factors could be attained in this device.

APPENDIX B INDUSTRIAL PARTICIPATION

Group: **The Finnish Blanket Group** consisting of Technip Offshore Finland, Diarc Technology Oy, Fortum Power and Heat Oy, Hollming Works Oy, High Speed Tech Oy, Kankaanpää Works Oy, Metso Powdermet Oy, Outokumpu Poricopper Oy and PI-Rauma Oy
Technology: Metal structures and plasma facing components
Contact: Jari Liimatainen jari.liimatainen@metso.com

Group: **The Finnish Remote Handling Group** consisting of Adwatec Oy, Fortum Power and Heat Oy, Hytar Oy, PI-Rauma Oy, Platom Oy, Creanex Oy, Rocla Oy and Delfoi Oy.
Technology: Remote handling, virtual reality, water hydraulics
Contact: Timo Mustonen timo.mustonen@creanex.com

Company: **ABB – Finland**
Technology: Power and automation
Contact: P.O. Box 661, 65101 Vaasa, Finland
Tel. +358-50-332-3515
Ralf Granholm ralf.granholm@fi.abb.com

Company: **Adwatec Oy**
Technology: Remote handling, water hydraulics, actuators and drives
Contact: Adwatec Oy, Polunmäenkatu 39 H 9, FIN-33720 Tampere,
Tel. +358 3 389 0860; fax. +358 3 389 0861
www.adwatec.com
Arto Verronen rto.verronen@adwatec.com

Company: **Aspocomp Oy**
Technology: Electronics manufacturing, thick film technology, component mounting (SMT), and mounting of chips (COB) in mechanical and electrical micro systems (MEMS) and multi-chip modules (MCM), PWB (or also called PCB), sheet metal manufacturing and assembly.
Contact: Aspocomp Oy, Yrittäjäntie 13, FIN-01800 Klaukkala,
Tel. +358 9 878 01244; Fax. +358 9 878 01200
www.aspocomp.com
Markku Palmu markku.palmu@aspocomp.com

Company: **Corrotech Oy**
Technology: Clean rooms, sheet metal production, mechanical engineering and surface treatment
Contact: Corrotech Oy, Teollisuuskatu 8, FIN-95420 Tornio,
Mobile: +358-40 777 9441; Fax +358 16 446 462
www.corrotech.fi
Esko Hilden esko.hilden@corrotech.fi

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Company	**Creanex Oy**
Technology:	Remote handling, teleoperation and walking platforms.
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Company:	**Delfoi Oy**
Technology:	Telerobotics, task level programming
Contact:	Delfoi Oy, Vänrikinkuja 2, FIN-02600 Espoo, Finland Tel. +358 9 4300 70; Fax. +358 9 4300 7277 www.delfoi.com Heikki Aalto heikki.aalto@delfoi.com
Company:	DIARC Technology Oy
Technology:	Diamond like DLC and DLC (Si, D) doped carbon coatings plus other coatings with potential plasma facing material in thermonuclear fusion machines.
Contact:	Diarc Technology, Olarinluoma 15, FIN-02200 Espoo, Tel. +358 9 2517 6130; fax +358 9 2517 6140 www.diarc.fi Jukka Kolehmainen jukka.kolehmainen@diarc.fi
Company:	**Ekono-Electrowatt/Jaakko Pöyry Group**
Technology:	International consulting and engineering expert within the Jaakko Pöyry Group serving the energy sector. Core areas of expertise: management consulting, hydropower, renewable energy, power & heat, oil & gas, project services for nuclear safety and industrial processes
Contact:	P.O.Box 93, Tekniikantie 4 A, FI-02151 Espoo, Finland Tel. +358-46911, Fax. +358-9-469-1981 www.poyry.com Vilho Salovaara vilho.salovaara@poyry.fi
Company:	**Ellego Powertec Oy**
Technology:	Power electronics, transformers, power sources, rectifiers based on modern chopper and thyristor technology
Contact:	Ellego Powertec Oy, P.O.Box 93, FIN-24101 Salo, Tel: +358 2 737 250, Fax +358 2 737 2530 www.trafotek.fi Pasi Lauri pasi.lauri@ellego.fi
Company:	**Elektrobit Microwave Oy**
Technology:	Product development, test solutions and manufacturing for microwave and RF- technologies, high-tech solutions ranging from space equipment to commercial telecommunication systems
Contact:	Teollisuustie 9A, FIN - 02700 Kauniainen, Finland Tel. +358 40 344 2000, Fax +358 9 5055 547 www.elektrobit.com Marko.Koski marko.koski@elektrobit.com

Company:	**Enprima Oy**
Technology:	Design, engineering, consulting and project management services in the field of power generation and district heating. EPCM services.
Contact:	P.O. Box 61, 01601 Vantaa, Finland Tel +358 40 348 5511, Fax. +358 9 3487 0810 www.enprima.com Jarmo Raussi jarmo.raussi@enprima.com

Company:	**Exel Oyj**
Technology:	Composite profiless, glass-, carbon- or aramid-fibres combined with polyester, vinylester or epoxy resins, superconducting current leads isolation profiles
Contact:	Exel Oyj, Muovilaaksontie 2, FIN-82110, Heinävaara, Tel. +358 13 73711, Fax. +358 13 7371500 www.exel.fi Matti Suominen matti.suominen@exel.fi

Company:	**Fortum Power and Heat Oy**
Technology:	Nuclear Engineering
Contact:	Fortum Nuclear Services Oy, Keilaniementie 1, Espoo, FIN-00048 Fortum, Finland Tel. + 358 10 4511; fax. +358 10 453 3403 www.fortum.com Harri Tuomisto harri.tuomisto@fortum.com

Company	**High Speed Tech Oy**
Technology:	Copper to stainless steel bonding by explosive welding
Contact:	High Speed Tech Oy, Tekniikantie 4 D, FIN-02150 Espoo, Fax. +358 9 455 5267 www.highspeedtech.fi Jaakko Säiläkivi jaakko.sailakivi@highspeed.sci.fi

Company:	**Hollming Works Oy**
Technology:	Mechanical engineering, fabrication of heavy stainless steel structures
Contact:	Hollming Works Oy, Puunaulakatu 3, P.O.Box 96, FIN-28101 Pori, Tel. +358 20 486 5040; fax +358 20 486 5041 www.hollmingworks.com Jari Mattila jari.mattila@hollmingworks.com

Company:	**Hytar Oy**
Technology:	Remote handling, water hydraulics
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Company: **Instrumentti-Mattila Oy**
Technology: Designs and manufacturing of vacuum technology devices
Contact: Valpperintie 263, 21270 Nousiainen, Finland
Tel +358-2-435 3611 , Fax +358-2-431 8744
www.instrumentti-mattila.fi
Veikko Mattila veikko.mattila@instrumentti-mattila.fi

Company: **Kankaanpää Works Oy**
Technology: Mechanical engineering, fabrication of heavy stainless steel structures including 3D cold forming of stainless steel
Contact: Kankaanpää Works Oy, P.O.Box 56, FIN-38701 Kankaanpää,
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Company: **Mansner Oy Hienomekaniikka**
Technology: Precision mechanics: milling, turning, welding, and assembling. From stainless steels to copper.
Contact: Mansner Oy Hienomekaniikka, Yrittäjäntie 73, FIN-03620
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www.mansner.com
Sami Mansner sami.mansner@mansner.fi

Company: **Marioff Corporation Oy**
Technology: Mist fire protection systems
Contact: Marioff Corporation Oy, P.O.Box 25, FIN-01511 Vantaa,
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www.hi-fog.com
Pekka Saari pekka.saari@marioff.fi

Company: **Metso Powdermet Oy**
Technology: Special stainless steels, powder metallurgy, component technology/ engineering, design, production and installation
Contact: Metso Powdermet Oy, P.O.Box 1100, FIN-33541 Tampere,
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www.metsopowdermet.com
Jari Liimatainen jari.liimatainen@metso.com

Company: **Outokumpu Poricopper Oy**
Technology: Superconducting strands and copper products.
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Technology:	Plasma diagnostics
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Company:	**PI-Rauma Oy**
Technology:	Computer aided engineering with CATIA.
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Company:	**Platom Oy**
Technology:	Remote handling, thermal cutting tools and radioactive waste handling.
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Company:	**Polartest Oy**
Technology:	3-party inspection, NDT, documentation and receiving inspection.
Contact:	Polartest Oy, Laajaniityntie 3, P.O.Box 41, FIN-01620 Vantaa, Tel. +358 9 878 020; Fax. +358 9 878 6653 www.polartest.fi Matti Andersson matti.andersson@polartest.fi

Company:	**PPF Products Oy**
Service:	Industry activation and support
Contact.	Portaantie 548, FIN-31340 Porras Tel. +358 3 434 1970, +358 50 40 79 799 Pertti Pale pertti.pale@surffi.net

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Role:	Co-ordination of the industry participation in the Fusion Programme
Contact:	Prizztech Oy, Teknologiakeskus Pripoli, Tiedepuisto 4, FIN-28600 Pori, Finland Tel. +358 2 620 5330; fax +358 2 620 5399 www.prizz.fi Leena Annila leena.annila@prizz.fi

Company:	**Rados Technology Oy**
Technology:	Dosimetry, waste & contamination monitoring and environmental monitoring.
Contact:	Rados Technology Oy, P.O.Box 506, FIN-20101 Turku, Tel. +358 2 4684 600; Fax +358 2 4684 601 www.rados.fi Erik Lehtonen erik.lehtonen@rados.fi

Company:	**Rejlers Oy**
Technology:	System and subsystem level design, FE modelling and analysis with ANSYS, studies and technical documentation, installation and maintenance instructions, 3D modelling and visualisation of machines and components.
Contact:	Rejlers Oy, Myllykatu 3, FIN-05840 Hyvinkää, Tel: +358 19 2660 600; Fax. +358 19 2660 601 www.rejlers.fi Jouni Vidqvist jouni.vidqvist@rejlers.fi

Company:	**Rocla Oyj**
Technology:	Heavy Automated guided vehicles
Contact:	Rocla Oyj, P.O.Box 88, FIN- 04401 Järvenpää, Tel +358 9 271 471, Fax +358 9 271 47 430 www.rocla.fi Pekka Joensuu pekka.joensuu@rocla.com

Company:	**Selmic Oy**
Technology:	Microelectronics design and manufacturing, packaging technologies and contract manufacturing services.
Contact:	Selmic Oy, Vanha Porvoontie 229, FIN-01380 Vantaa, Tel: +358 9 2706 3911; Fax +358 9 2705 2602 www.selmic.com Patrick Sederholm patrick.sederholm@selmic.com

Company:	**Solving Oy**
Technology:	Heavy automated guided vehicles. Equipment for heavy assembly and material handling based on air film technology for weights up to hundreds of tons.
Contact:	Solving Oy, P.O.Box 98, FIN-68601 Pietarsaari, Tel. +358 6 781 7500; Fax. +358 6 781 7510 www.solving.fi Bo-Goran Eriksson bo-goran.eriksson@solving.fi

Company:	**Sweco PIC Oy**
Technology:	Consulting and engineering company operating world-wide, providing consulting, design, engineering and project management services for industrial customers in plant investments, product development and production.
Contact:	Liesikuja 5, P.O.Box 31, 01601 Vantaa, Finland Tel. +358-9-53091 Kari Harsunen kari.harsunen@sweco.fi

Company:	Tampereen Keskustekniikka Oy
Technology:	Product development, design, production, marketing, and sales of switchgear and controlgear assemblies.
Contact:	Hyllilänkatu 15, P.O.Box 11, 33731 Tampere, Finland Tel. +358-3-233 8331 www.keskustekniikka.fi Reijo Anttila reijo.anttila@keskustekniikka.fi

Company:	**Tankki Oy**
Technology:	Production and engineering of stainless steel tanks and vessels for use in different types of industrial installations
Contact:	Oikotie 2, FIN-63700 Ähtäri, Finland Tel. +358 6 510 1111, Fax +358 6 510 1200 Jukka Lehto jukka.lehto@tankki.fi

Company:	**Technip Offshore Finland** (earlier Mäntyluoto Works Oy)
Technology:	Fabrication of Heavy Steel constructions by using an effective modulus technology, pressure vessels and piping
Contact:	Mäntyluoto Works Oy, FIN-28880 Pori, Tel. +358 2 528 2411; fax +358 2 528 2419 www.coflexipstenaoffshore.com

Company:	**TVO Nuclear Services Oy**
Technology:	Nuclear power technologies; service, maintenance, radiation protection and safety.
Contact:	TVO Nuclear Services Oy, FIN-27160 Olkiluoto, Tel. + 358 2 83 811; Fax. +358 2 8381 2109 www.tvo.fi Antti Piirto antti.piirto@tvo.fi

Company:	**Veslatec Oy**
Technology:	Micro cutting-laser welding-laser drilling-laser marking
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Company:	**Woikoski Oy**
Technology:	Production, development, applications and distribution of gases and liquid helium
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Association Euratom-Tekes. Annual Report 2005

Abstract

This report summarises the results of the Tekes FUSION technology programme and the fusion research activities by the Association Euratom-Tekes in 2005. The research areas are fusion physics and plasma engineering, fusion technology and a smaller effort to socio-economic studies. Fusion technology research is carried out in close collaboration with Finnish industry.

The emphasis in fusion physics and plasma engineering is in theoretical and computational studies on turbulent transport and modelling of radio-frequency heating experiments. The work covered the real time control of transport barriers in JET plasmas, predictive integrated modelling of tokamak plasmas. Plasma-wall studies included coating technologies for plasma facing components and material transport in the edge plasmas supported by surface analysis of the JET divertor and limiter tiles. Two projects on diagnostics started in 2005.

The work in fusion technology for the EFDA Technology Programme and ITER is strongly focused into vessel/in-vessel materials and remote handling studies. The main activity was to prepare hosting of the ITER divertor test platform (DTP2) at VTT. Other remote maintenance systems include water hydraulic manipulators for the ITER divertor maintenance as well as prototyping of intersector welding and cutting robot. Virtual modelling is an essential element in the remote handling engineering. A second domain of fusion technology covers research and characterisation of first wall materials, mechanical testing of reactor materials under neutron irradiation, studies of joining and welding methods, characterisation of irradiated CuCrZr/SS joints. Some effort was also devoted to new stainless steels, IFMIF-design, fusion neutronics and socio-economic studies. Several EFDA technology tasks and contracts were successfully completed in 2005.

Keywords
nuclear fusion, fusion reactors, reactor vessels, fusion research, fusion physics, fatique testing, construction materials, tritium breeding, radiation damages, EURATOM

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