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FFUSION Research Programme 1993 - 1998

Final Report of the Finnish Fusion Research Programme

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Fusion is the energy source of the sun and the stars.
The cover image shows the NGC 6188 nebula and NGC 6193.
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Foreword

Energy availability and its proper utilization have always played an essential role in socio-economic development. The overall world energy consumption has increased some eighteen-fold over the last hundred years and this increase is observed to correlate per capita with the level of wealth, health and education in any specific region. Fusion, the process utilized by nature as the fundamental energy source in the sun and the stars, provides us in the long term with a sustainable development path for a safe and environmentally friendly energy option. Globally, the responsibility of this long-term development of world energy options belongs to the industrialized countries. Finland is strongly committed to this international co-operation.

The competitiveness of a country today depends on its capability to create innovations, which are based on science and technology and on its industries' ability to turn them into products and services for the world market. Despite the vast amount of research already performed on fusion development, it still remains a challenge that stimulates new thinking, new technologies and new industrial capabilities. In the FFUSION programme the participation of Finnish industry has increased laudably. This verifies the chosen strategy, where it is stated that the best type of technology transfer occurs where enterprises are linked to the carrying out of the work. International fusion research provides us with a first-class platform for benchmarking new technologies and finding spin-offs and even technology jumps.

The role of education and training has been evident from the very beginning of the programme. The high degree of cohesion among the researchers and their motivation to contribute to the technology transfer have promoted many new clustering initiatives around specific themes together with industry. The programme has also gained a lot by the excellent research results, proving the level of Finnish know-how.

The future of fusion research in Finland is very closely connected to international co-operation. From the good experiences gained from this programme we are looking forward to contributing to a technology-driven international programme, which has to lead to an energy source that is both economically and socially acceptable. Many questions, such as quality of life, progress, security, and well-being are linked to the theme of energy and environment and therefore they have a direct impact on the issue of fusion energy.

The Technology Development Centre of Finland expresses its sincere thanks to all individuals, enterprises and institutes who have contributed to the programme. This gratitude is extended also to the international scientific and industrial fusion community.

In Helsinki, 18 September, 1998

Technology Development Centre of Finland

Karttunen, Seppo, Ahola, Heikki, Olgierd, Dumbrajs, Halme, Aarne, Heikkinen, Jukka, Heikkinen, Veli, Keinonen, Juhani, Korhonen, Riitta, Kurki-Suonio, Taina, Likonen, Jari, Pättikangas, Timo, Salomaa, Rainer, Siuko, Mikko, Teuho, Juhani, Tähtinen, Seppo & Wasastjerna, Frej. FFUSION Research Programme 1993 - 1998. Final Report of the Finnish Fusion Research Programme. Espoo 1998, Technical Research Centre of Finland, VTT Publications 368. 110 p. + app 48 p.

Keywords fusion reactors, fusion physics, plasma engineering, remote handling, nuclear energy

Summary

This report summarizes the results of the Fusion Energy Research Programme, FFUSION, during the period 1993-1998. After the planning phase the programme started in 1994, and later in March 1995 the FFUSION Programme was integrated into the EU Fusion Programme and the Association Euratom-Tekes was established.

Research areas in the FFUSION Programme are (1) fusion physics and plasma engineering, (2) fusion reactor materials and (3) remote handling systems. In all research areas industry is involved. Recently, a project on environmental aspects of fusion and other future energy systems started as a part of the socio-economic research (SERF) in the Euratom Fusion Programme.

A crucial component of the FFUSION programme is the close collaboration between VTT Research Institutes, universities and Finnish industry. This collaboration has guaranteed dynamic and versatile research teams, which are large enough to tackle challenging research and development projects. Regarding industrial fusion R&D activities, the major step was the membership of Imatran Voima Oy in the EFET Consortium (European Fusion Engineering and Technology), which further strengthened the position of industry in the engineering design activities of ITER.

The number of FFUSION research projects was 66. In addition, there were 32 industrial R&D projects. The total cost of the FFUSION Programme in 1993-1998 amounted to FIM 54 million in research at VTT and Universities and an additional FIM 21 million for R&D in Finnish industry. The main part of the funding was provided by Tekes, 36%. Since 1995, yearly Euratom funding has exceeded 25%.

The FFUSION research teams have played an active role in the European Programme, receiving excellent recognition from the European partners. Theoretical and computational fusion physics has been at a high scientific level and the group collaborates with the leading experimental laboratories in Europe. Fusion technology is focused on reactor materials, joining techniques, superconductor development and water-hydraulic applications, in which FFUSION teams have achieved a firm position in the EU Fusion Programme. A challenging full-scale prototype system for the ITER in-vessel viewing has been completed as a collaborative effort between VTT, Helsinki University of Technology and IVO Technology Centre.

Contents

1 FFUSION Research Programme

1.1 Background

Fusion energy research in Finland started as a small effort at the Technical Research Centre of Finland (VTT) in the mid-1970's. The main objective was to watch the scientific and technological development in international fusion research and to carry out theoretical and computational studies in fusion physics. In the early 1990's, it became evident that Finland was to have closer relations with the European Community and that this would open a possibility to join to the European Fusion Programme and with it to the world-wide ITER Project (International Thermonuclear Experimental Reactor).

In 1992, the Finnish Nuclear Energy Commission presented an initiative to the Ministry of Trade and Industry for embarking on a national fusion research programme in Finland. The aim was to organize all fusion-related research in Finland under a single programme and to increase the research volume to a relevant level in European terms before joining the Community Fusion Programme. It was also realized that the European Programme and ITER would provide challenging opportunities for Finnish hi-tech industry. Negotiations with the European Fusion Programme started in early 1993. At the same time, a systematic survey of the possibilities and interest of Finnish industry was made. Programme planning was carried out in 1993 and the Finnish Fusion Research Programme - FFUSION officially started in the beginning of 1994.

The first phase of the FFUSION programme (1993-1994) was the preparation for association into the Community Programme. The goal was to reach a critical size in research volume and to identify our own research areas. The strategy was to emphasize fusion technology parallel with the basic fusion and plasma physics and to activate Finnish industry into collaborating and participating in the FFUSION programme, and subsequently in the European Fusion Programme. The key element in the strategy was focusing our fairly small R&D efforts on a few topics, where Finland has possibilities to be competitive in Europe.

The early objectives above were met when the FFUSION programme was fully integrated into the European Fusion Programme, just after Finland had joined the European Union. The Association Euratom-Tekes was established when the Contract of Association between Euratom and Tekes was signed in Helsinki, on March 13, 1995. Tekes became the 14th Euratom Association in the EU Fusion Programme. Other contracts include the multilateral NET Agreement and the Staff Mobility Agreement. Finland, represented by Tekes, became a member of the JET Joint Undertaking on May 7, 1996.

1.2 European Fusion Programme

The EU Fusion Programme is a fully integrated programme that includes all fusion research carried out in the Member States and Switzerland. The EU is investing about ECU 840 million in fusion research in its Fourth Framework Programme (1994-1998). The European Fusion Programme consists of four elements: the JET Joint Undertaking, the NET Team, the fusion technology research at the Joint Research Centre (JRC) in Ispra and Petten, and the fusion research in the Associations. The largest fusion research installation in the world is the Joint European Torus (JET) tokamak in England. The JET Joint Undertaking will end in December 1999. The present plan is that after 1999 the JET facilities will be operated and exploited by the research teams from the Association Euratom-UKAEA and other Associations.

A significant proportion of European fusion research is carried out in national laboratories that have signed the Contract of Association with the Community programme. At the moment, there are 16 Associations in the European Programme including Switzerland. Only Greece and Luxembourg are without a Contract of Association, but Greece is participating in the EU programme on a project basis. There is a number of large and medium-size tokamaks and plasma devices in the associated laboratories. Euratom is also financing research into fusion installations alternative to the tokamak. A new JT-II stellarator in Spain started plasma operation in 1998 and a large Wendelstein 7-X stellarator is under construction in Greifswald, Germany to be completed in 2004.

The main global research project is the engineering design activities (EDA) of the tokamak test reactor ITER (International Thermonuclear Experimental Reactor). The participants are Euratom, Japan, Russia and the United States. The ITER EDA was completed in summer 1998. The partners are planning to extend the design work by three years to include site-specific studies and procurement specifications and to complete the tests of large prototype components, e.g., superconducting model coils and blanket modules. The possibilities of reducing the capital cost of ITER significantly, but still satisfying the programme objectives of the ITER EDA Agreement, are under study.

The NET Team co-ordinates fusion technology work and European Home Team work for ITER. These include several fusion technology research projects involving fusion reactor materials, tritium handling, remote maintenance equipment, large superconductor magnets, safety and environmental issues and socio-economic studies.

1.3 FFUSION Programme Objectives

The Finnish Fusion Programme FFUSION / the Association Euratom-Tekes, is fully integrated into the European Programme, which has set the long-term goal of “the joint creation of safe, environmentally sound prototype reactors, which should result in the construction of economically viable power stations”. The national objectives of the FFUSION Programme in the shorter term are:

- to carry out high-level scientific and technological research in support of the European Fusion Programme and ITER
- to promote collaboration between the research institutes, universities and Finnish industry
- to focus the R&D effort on a few competitive areas.

Active participation in the Euratom Fusion Programme and ITER Engineering Design Activities has opened challenging opportunities and projects to the Finnish science and technology community and hi-tech companies.

1.4 Research Areas

The FFUSION programme consists of fusion plasma physics and fusion technology.

The physics programme is carried out at the Technical Research Centre of Finland (VTT), Helsinki University of Technology (HUT), and the University of Helsinki (UH). The research areas in fusion physics are:

- fusion plasma engineering / radio-frequency heating and plasma diagnostics
- plasma-wall interactions.

The physics programme consists of theoretical and computational studies of radio-frequency heating and current drive, particle and energy transport and diagnostics in tokamak plasmas. Experiments on plasma-wall interactions are performed with the ion beam facility at the University of Helsinki. ITER Tasks dealing with the R&D and design of radio-frequency systems have partly been performed under the physics programme. A major part of the physics programme is conducted in collaboration with the JET Joint Undertaking and Associations IPP Garching (Germany), CEA Cadarache (France), and CRPP Lausanne (Switzerland).

The technology programme is carried out at VTT, HUT and Tampere University of Technology (TUT) in close collaboration with Finnish industry. The technology research is focused on three areas:

- fusion reactor materials – first-wall components, joining techniques and characterization
- remote handling and viewing systems
- superconductors.

The respective volumes of the FFUSION research projects and industrial R&D projects in the main research areas are given in Table 1.1. The following Association Technology Tasks, NET Contracts and JET Task Agreements were carried out during 1995-1998:

Association Technology Tasks:

Task T361	Vacuum Window Development for Ion Cyclotron Radio-Frequency Power Transmission Line
Task T212	Cu/SS Joining Technology
Task BL12.2-1	Detailed Investigation of CuAl25(IG1), its Joints with 316LN SS and Joints Testing Procedures
Task T213	Cu and Cu-Alloys Irradiation Testing
Task BL16.5-2	Titanium Alloys Irradiation Testing
Task T217	Aqueous Corrosion of 316L SS and Cu-Based Alloys
Task T301/3	High-Energy Beam Welding for Manufacture of Large Tokamak Containment Sectors
Task T226a	Evaluation of Erosion / Re-deposition
Task T227	Tritium Permeation and Inventory
Task T221	Plasma Facing Armour Materials
Task DV7a	Tritium Permeability, Retention, Wall Conditioning / Clean Up
Task M11	ITER NbTi Superconducting Wire Development

Task M2/1	ITER Nb_3Sn Superconducting Wire Development
Task T328	ITER In-Vessel Viewing System
Task T232.11	Feasibility Study of Divertor Facility
Task T308/6	Tools for the ITER Divertor Refurbishment Platform

NET Article 6 Contracts:

NET A6-404	Development of Tooling for Divertor
NET A6-402	Support of Nuclear Analysis
NET A6-467	Nuclear Analysis of the Equatorial Heating Ports
NET A6-456	Non-Destructive Examination of Primary Wall Small-Scale Mock-ups

NET Article 7 / EFET Contracts:

NET A7-851CA/DN	ICRF Vacuum Transmission Line – Dielectric Window Development (2 Contracts)
NET A7-851CG/EB	In-Vessel Viewing System – Design of Prototype Systems and Demo Imaging System (2 Contracts)
NET A7-851DJ	SEAFP-2-Improved Containment Concepts – External Hazards
NET A7-851DT	ITER FDR Costing, Task 3

JET Task Agreements:

DAMD/Tekes/01	A) The Role of Short-Wavelength Waves during Heating and Current Drive in the Ion Cyclotron Range Frequencies, B) Development and Experimental Evaluation of Theoretical Models in the Field of ICRF Heating
Tekes TA6:	Code Development for RF Modules in Transport Codes

Industry is involved in all Association technology tasks. In addition, there are seven industrial ITER design tasks through the European Fusion Engineering and Technology (EFET) Consortium.

Underlying technology in reactor materials includes the further development of fracture resistance test methods and verification of specimen size effects, measuring techniques for characterising surface film properties of metals in coolant water environments and the development of non-destructive examination techniques applicable to inspection of primary wall modules.

Association Euratom-Tekes contributes to the socio-economic research on fusion (SERF) with the project “Identification and comparative evaluation of environmental impacts of fusion and other possible future energy production technologies”.

Three NET Assignments and two JET Task Agreements have been made since 1995, and one to two persons have been working on JET. Collaboration with the NET Team, JET Joint Undertaking, ITER Joint Central Team, Association Risø (Denmark), Association FZK Karlsruhe (Germany) and Association ENEA Frascati (Italy) has played an essential role in the fusion technology activities of the FFUSION Programme.

1.5 Participating Institutes and Companies

1.5.1 The Technology Development Centre Finland (Tekes)

The Technology Development Centre Finland (Tekes) is the main funding authority and co-ordinator for technological research and development activities in Finland. Tekes is the co-ordinator of eleven national technology research programmes in the energy sector including the FFUSION programme. The fusion research co-ordinators in Tekes are Dr. Seppo Hannus (Director of Energy Technology), Mr. Martti Korkiakoski (Senior Adviser) until June 1997 and Mr. Reijo Munther (Senior Adviser) from July 1997.

1.5.2 Finnish Fusion Research Unit

Research activities in the FFUSION programme are carried out in several VTT Research Institutes and in universities. The FFUSION research programme is co-ordinated by VTT Energy. The director of the FFUSION programme is Dr. Seppo Karttunen and he is acting as Head of Research Unit of the Association Euratom-Tekes.

The following universities and VTT Institutes have been participating in the fusion research in 1993-1998:

Technical Research Centre of Finland (VTT):

VTT Energy (FFUSION co-ordination, plasma engineering, neutronics)
VTT Manufacturing Technology (materials)
VTT Chemical Technology (materials)
VTT Automation (remote handling)
VTT Electronics (remote handling)

Helsinki University of Technology (HUT):

Department of Engineering Physics and Mathematics (plasma engineering, diagnostics)
Laboratory of Automation Technology (remote handling)

University of Helsinki (UH):

Accelerator Laboratory (plasma-wall interactions, first-wall materials)

Tampere University of Technology (TUT):

Institute of Hydraulics and Automation (remote handling)
Laboratory of Control Engineering (remote handling)

The Finnish Fusion Research Unit consists of research groups from the institutes and universities above.

1.5.3 Industrial Companies

The following industrial companies have been collaborating with the FFUSION research programme:

Imatran Voima Oy (IVO, Finnish EFET partner)
Outokumpu Superconductors Oy
Outokumpu Poricopper Oy
Plustech Oy
High Speed Tech Oy
Diarc Technology Oy
Hytar Oy
Rauma Materials Technology Oy
PI-Rauma Oy
Tehdasmallit Oy

Pori Works Oy
Patria Finavitec Oy

Industrial activities related to the FFUSION programme are co-ordinated by Prizztech Oy. The Finnish Blanket Group and Remote Handling Group involving the companies above were formed in 1995 and they are with Outokumpu Superconductors accepted onto the list of qualified companies for ITER EDA. Imatran Voima Oy became a member of the European Fusion Engineering and Technology (EFET) Consortium in 1996. The other EFET members are: Siemens (Germany), Framatom (France), Belgatom (Belgium), CITEF (Italy), NNC Limited (UK) and IBERTEF (Spain).

The relative volume of the research and development work in the participating institutions can be seen in Fig. 1.1. Five VTT institutes account for about 43% of the research volume, universities 28% and industry 29%.

1.5.4 FFUSION Steering Committee

The national steering committee of the FFUSION Programme advises in the planning of fusion research and promotes collaboration with Finnish industry.

The members of the FFUSION Steering Committee are:

Chairman:	Rainer Salomaa, Professor, Helsinki University of Technology
Members:	Erkki Kare, Managing Director, Plustech Oy
	Juhani Keinonen, Professor, University of Helsinki
	Martti Korkiakoski, Senior Technical Adviser (1995-97), Tekes
	Reijo Munther, Senior Technical Adviser (1997-98), Tekes
	Lenni Laakso, General Manager, Outokumpu Poricopper Oy
	Lasse Mattila, Research Professor, VTT Energy
	Jukka Lindgren, Senior Technical Adviser (1993-1994), Ministry of Trade and Industry
	Juha Paappanen, General Manager, Imatran Voima Oy
	Pertti Pale, The NET Team / Prizztech Oy
FFUSION director:	Seppo Karttunen, Senior Research Scientist, VTT Energy
Secretary:	Timo Pättikangas, Research Scientist, VTT Energy

Since 1994, there have been 13 meetings of the FFUSION Steering Committee.

1.6 FFUSION Programme Funding

The FFUSION research programme is financed by the Ministry of Trade and Industry (in 1993), Tekes (from 1994), the Finnish Academy of Sciences, the participating institutes (VTT, HUT, TUT and UH) and industry. From the signing of the Contract of Association on March 13, 1995, European funding support has also been provided by Euratom.

Fig. 1.2 shows the yearly funding of the FFUSION programme from 1993 to 1998, including the funding from the participating industrial companies. The distribution of the total funding between the different organizations during the six-year period 1993-1998 is shown in Fig. 1.3. The total funding of the FFUSION research programme for 1993-1998 is about FIM 54.4 million. The total volume of the industrial activities related to the FFUSION programme is about FIM 21 million for the same period.

The funding details of the FFUSION research and industry R&D projects are given in Table 1.2. Table 1.2 shows that the funding of the FFUSION research projects is fairly balanced between the three research areas. In the industrial R&D, remote handling projects and superconductor development take the major share of the funding.

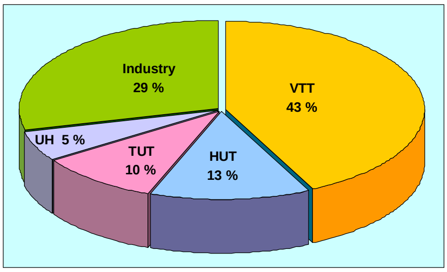


Fig. 1.1. Research volumes of participating institutions VTT, universities and industry in 1993-1998.The total amount of expenditures during the period 1993-1998 is approximately FIM 75 million.

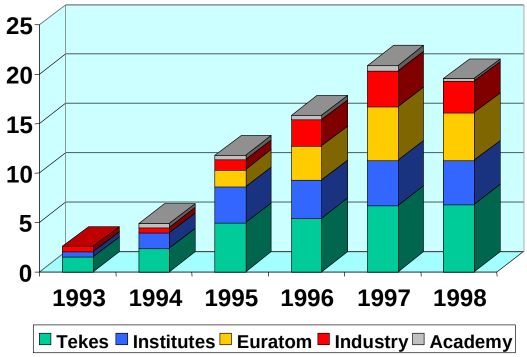


Fig. 1.2. Yearly funding (in MFIM) of the FFUSION programme in 1993-1998. A relatively large increase took place in 1995 when the FFUSION programme was integrated into the EU Fusion Programme.

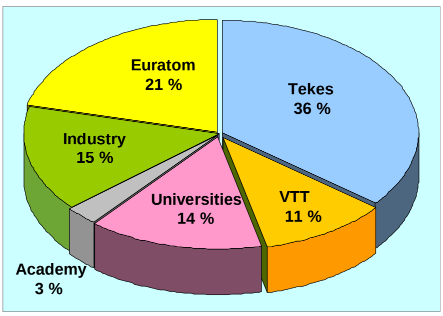


Fig. 1.3. Distribution of the funding of the FFUSION programme and the related industrial R&D projects between the different organizations during the period 1993-1998. The total value of the funding is approximately FIM 75 million.

Table 1.1. Number of FFUSION research and industrial R&D projects in 1993-1998.

Research Area	**Research Projects**						**Industrial Projects**						**All Projects**							**Share**
	93	**94**	**95**	**96**	**97**	**98**	**93**	**94**	**95**	**96**	**97**	**98**	**93**	**94**	**95**	**96**	**97**	**98**	**93-98**	**(%)**
Coordination	1	1	1	1	1	1	1	1	1	1	1	1	2	2	2	2	2	2	12	12,2
Fusion Plasma Engineering	2	2	3	5	5	4				1	2	2	2	2	3	6	7	6	26	26,5
Fusion Reactor Materials	2	4	6	5	7	5			2	3	2	2	2	4	8	8	9	7	38	38,8
Superconductors										1	2	2				1	2	2	5	5,1
Remote Handling and Viewing			3	2	2	2			1	2	2	2			4	4	4	4	16	16,3
Socio-economy (SERF)						1												1	1	1,0
Total	5	7	13	13	15	13	1	1	4	8	9	9	6	8	17	21	24	22	98	100

Table 1.2. Funding (in MFIM) of the FFUSION research and industrial R&D projects in 1993-1998.

Research Area	**Research Projects**						**Industrial Projects**						**Projects Total**							**Share**
	93	**94**	**95**	**96**	**97**	**98**	**93**	**94**	**95**	**96**	**97**	**98**	**93**	**94**	**95**	**96**	**97**	**98**	**93-98**	**(%)**
Fusion Plasma Physics	1,8	2,7	3,3	4,3	3,9	3,9				0,5	0,5	0,1	1,8	2,7	3,3	4,8	4,3	4	20,876	27,7
Fusion Reactor Materials	0,5	1,1	4	4,2	4	4,1			0,5	0,5	1	0,9	0,5	1,1	4,6	4,7	5	5	20,723	27,4
Superconductors										0,6	1,9	1,8	0	0	0	0,6	1,9	1,8	4,36	5,8
Remote Handling/Viewing Systems			1,6	2,4	6	4,6			0,7	1,3	1,3	1,1	0	0	2,4	3,7	7,2	5,6	18,951	25,1
Environmental Effects / SERF						0,4					0,1	0,2	0	0	0	0	0,1	0,6	0,768	1,0
FFUSION / Industrial coordination		0,1	0,3	0,4	0,4	0,7	1	1	1,5	1,5	1,5	1,5	1	1,1	1,7	1,9	1,9	2,2	9,818	13,0
Total	2,3	3,9	9,2	11	14	14	1	1	2,7	4,5	6,3	5,6	3,3	4,9	12	16	20	19	75,496	100

1.7 International Collaboration

1.7.1 Association Euratom-Tekes

The FFUSION programme was fully integrated into the European Fusion Programme just after Finland joined the European Union. The Association Euratom-Tekes was established when the Contract of Association between Euratom and Tekes was signed in Helsinki, on March 13, 1995. The present Contract of Association extends to the end of 1999. Finland, represented by Tekes, became a member of the JET Joint Undertaking on May 7, 1996. Other contracts of the Association Euratom-Tekes include the multilateral NET Agreement and the Staff Mobility Agreement. The FFUSION programme with participating research groups from VTT and universities forms the Fusion Research Unit of the Association Euratom-Tekes.

Association Steering Committee

The research activities of the Finnish Association Euratom-Tekes are directed by the Association Steering Committee. The Steering Committee supervises the execution of the Contract of Association, adopts the details of the programme, ensure the progress of the research activities and steers them towards the programme objectives. It also appoints the Head of Research Unit on the proposal of Tekes.

The members of the Association Steering Committee are:

Dr. Charles Maisonnier, EU Commission, DG XII, Chairman in 1995
Dr. Seppo Hannus, Tekes, Chairman in 1996, 1998
Dr. Umberto Finzi, EU Commission, DG XII, Chairman in 1997

Members:	Mr. Juhani Ahava, 1997, The Finnish Academies of Technology
	Mr. Magnus von Bonsdorff, 1997, The Finnish Academies of Technology
	Dr. Hardo Bruhns, EU Commission, DG XII
	Dr. Janos Darvas, 1995, EU Commission, DG XII
	Dr. Matti Kankaanpää, 1995-96, The Finnish Academies of Technology
	Prof. Mikko Kara, 1998, VTT Energy

	Mr. Juha Paappanen, 1998, IVO Technology Centre
	Prof. Pekka Silvennoinen, 1995-97, VTT Information Technology
	Mr. Johannes Spoor, EU Commission, DG XII
Secretary (1995-97):	Mr. Martti Korkiakoski, 1995-96, Tekes
Secretary (1997-98):	Mr. Reijo Munther, 1997-98, Tekes
Head of Research Unit:	Dr. Seppo Karttunen, VTT Energy

The Association Steering Committee has had 5 meetings during the period 1995-1998. The Steering Committee accepts annual accounts, yearly budgets and research programme and the annual reports of the Research Unit.

1.7.2 Participation in the Committees of the European Fusion Programme

The Finnish representatives on the different Committees of the EU Fusion Programme are given below.

Consultative Committee for Fusion Programme (CCFP):

Dr. Seppo Hannus, Tekes
Dr. Seppo Karttunen, VTT Energy
Mr. Martti Korkiakoski, Tekes (until June 1997)
Mr. Reijo Munther, Tekes (since July 1997)

Fusion Technology Steering Committee - Implementation (FTSC-I):

Mr. Rauno Rintamaa, VTT Manufacturing Technology

Programme Committee (PC):

Dr. Seppo Karttunen, VTT Energy
Prof. Rainer Salomaa, HUT

JET Council:

Dr. Seppo Karttunen, VTT Energy
Mr. Martti Korkiakoski, Tekes (until June 1997)
Mr. Reijo Munther, Tekes (since July 1997)

JET Executive Committee:

Mr. Martti Korkiakoski, Tekes (until June 1997)
Mr. Reijo Munther, Tekes (since July 1997)
Prof. Rainer Salomaa, HUT

The following fusion committees and expert groups have Finnish representatives:

- Dr. Jukka Heikkinen is a member of the Co-ordinating Committee for Fast Wave Heating (CCFW).
- Dr. Seppo Karttunen (1995-96) and Dr. Timo Pättikangas (1997-1998) have been members of the Co-ordinating Committee for Lower Hybrid Heating and Current Drive (CCLH).
- Prof. R. Salomaa is a member of the European Fusion Information Network (EFIN) since 1998.
- Mr. Seppo Tähtinen is a Materials Liaison Officer in the European Blanket Project
- Dr. Olgierd Dumbrajs is a member of the international experts commission on Electron Cyclotron Wave Systems.
- Dr. Seppo Karttunen was a member of the Ad-Hoc Groups, which carried out an evaluation of different heating and current drive methods and an assessment of the ITER Physics Performance presented in the ITER Detailed Design Report and ITER Final Design Report.

Both co-ordinating committees CCFW and CCLH have had meetings in Finland. Tekes hosted the 12th ITER Council and the 4th ITER Explorers´ meetings in Tampere, in July 1997. Dr. Karttunen participated in the 12th ITER Council Meeting in Tampere as a local organizer.

1.7.3 European and Other International Collaboration

All fusion research in the Euratom Associations is co-ordinated on a European level so that joint projects, collaboration between Associations and EU Home Team work for ITER EDA are the basic elements of the EU Fusion Programme.

In plasma physics, the Association Euratom-Tekes participates in the JET fusion experiments with two Task Agreements and in experiments at IPP Garching (ASDEX Upgrade tokamak and Wendelstein AS-7 stellarator) and at CEA Cadarache (Tore Supra). In gyrotron development work, collaboration with Associations FZK Karlsruhe and CRPP Lausanne has been started.

In fusion technology, there are joint research projects with the Associations Risö (reactor materials) and ENEA Frascati / Brasimone (in-vessel viewing and divertor refurbishment). The NET Team in Garching co-ordinates the European collaboration in fusion technology tasks and work for ITER EDA.

The staff mobility scheme of the EU Fusion Programme offers excellent opportunities for the exchange of scientists and engineers in Europe. There has been over 15 mobility visits of 1 to 6 months in 1996-98. Longer visits over one year under other arrangements have been made for JET, NET Team and IPP. In addition, several shorter visits both ways have taken place since 1993.

Collaboration with non-EU countries has played a minor role after the Association agreement in 1995. There is still close collaboration with the Ioffe Institute in St. Petersburg (fusion theory, Globus tokamak) and Institute for Applied Physics in Nizhny Novgorod (gyrotrons). Yearly fusion symposiums between HUT and the Ioffe Institute have been organized since 1993. Collaboration with the Centre Canadien de Fusion Magnétique (Tokamak de Varennes) in Canada will continue.

2 Fusion Physics and Plasma Engineering

During 1993–98, the work within fusion physics and plasma engineering was strongly focused on modelling and design efforts on various European fusion facilities, e.g., ASDEX Upgrade and Wendelstein 7-AS in Germany, JET in England, Tore Supra in France, and the international ITER project. The main fields of research were radio-frequency heating and transport processes, in which the fusion and plasma physics group has acquired a high level of expertise and knowledge, particularly in numerical modelling.

The locally developed code arsenal includes the orbit-following code ASCOT, various wave codes, as well as gyrotron models, all of which are internationally recognised and have formed a basis for a number of task and collaboration agreements with other Associations and JET. Currently, the most sophisticated and versatile numerical tool is ASCOT, which has been developed for several years and used for many different research topics. ASCOT follows charge particle orbits in realistic tokamak geometry. Besides external electric and magnetic fields, ASCOT also has operators modelling Coulomb scattering between the test particles and the plasma background, as well as operators for radio-frequency waves and anomalous radial transport. ASCOT has recently been parallelized to facilitate studies requiring very large numbers of test particles. The latest addition to the research tools is particle-in-cell simulations to model lower hybrid heating.

Two plasma engineering projects have also been initiated: in collaboration with industry, the group is engaged in the development of the central solenoid for a spherical tokamak, and in the design and construction of a radio-frequency vacuum window. Particularly the latter project has significantly increased contacts and common research activity of the group with industry. The group members have also actively participated in various international committees and ad-hoc groups for co-ordination and evaluation of physics and engineering in their field, and have also been entrusted with several international review and referee duties.

2.1 Radio-Frequency Heating of Tokamak Plasmas

Before the energy-producing fusion reactions can occur in a deuterium-tritium plasma, the plasma has to be heated up to a very high temperature. In modern tokamaks, the external heating is provided by either neutral beam injection (NBI), or by radio-frequency (rf) waves. Fusion reactions produce alpha particles, which are nuclei of He-4 atoms. These alphas have a kinetic energy of 3.5 MeV, which is collisionally transferred to plasma ions and electrons. Ignition is achieved if the alpha heating alone is able to sustain the temperature of the plasma fuel, and the auxiliary heating can be turned off.

In ion cyclotron heating, waves with relatively low frequencies of 10–60 MHz are used. The ion cyclotron waves have traditionally been launched to tokamak plasmas using inductive loop antennas. The waves are absorbed in the region where the wave frequency equals the ion gyro-frequency (the ions gyrate around the magnetic field line), or a multiple of the gyro-frequency.

In lower hybrid heating, waves with frequencies of 1–10 GHz are used. The wave is launched using a phased waveguide array, called a grill. The main application of the lower hybrid waves is non-inductive current drive.

In electron cyclotron heating, waves with high frequencies of 10–200 GHz are typically used. Gyrotrons are used as power sources. The electron cyclotron waves can be launched simply from the waveguide aperture, or the radiation pattern can be taylored with the aid of mirrors inside the vacuum chamber. The waves are absorbed in the region where the wave frequency equals the electron gyro-frequency.

In addition to heating a plasma, the radio-frequency waves are also used for generating a toroidal electric current in a tokamak. Traditionally, the plasma current is produced inductively, thus making a tokamak an inherently pulsed machine. A non-inductive current drive using rf-waves could allow a continuous operation of a tokamak.

In the following, the most important results on radio-frequency heating and current drive, obtained in the FFUSION Programme, are described.

2.1.1 JET Task: Ion Cyclotron Heating and Current Drive

The main motivation of the present Task Agreement, initiated in 1995, has been to develop and use efficient codes to model high performance ion cyclotron radio-frequency heating (ICRH) experiments in JET tokamak. The work continues the close collaboration of JET, Helsinki University of Technology, and VTT on radio-frequency physics which started in 1988.

A combined power deposition and Fokker-Planck code, PION, developed for dynamical evaluation of ICRF-heated ion distributions has been developed at JET. PION, in parallel with the transport code TRANSP, has been used to model neutron production rate and heating in the recent high-power deuterium-tritium experiments. The record fusion amplification by pure ion cyclotron heating was obtained in these experiments. The world record fusion power was achieved with the combined neutral beam and ion cyclotron heating. PION has accurately reproduced the observed neutron emission in most ICRH scenarios in JET.

Fundamental minority heating of deuterium was tested at JET for the first time ever. The scheme was very successful, maintaining the record Q value of about 0.22 for steady state discharges over a period of about three energy confinement times. In general, the simulated and experimental deuterium-tritium fusion reactions rates are in very good agreement as can be seen in Fig. 2.1.

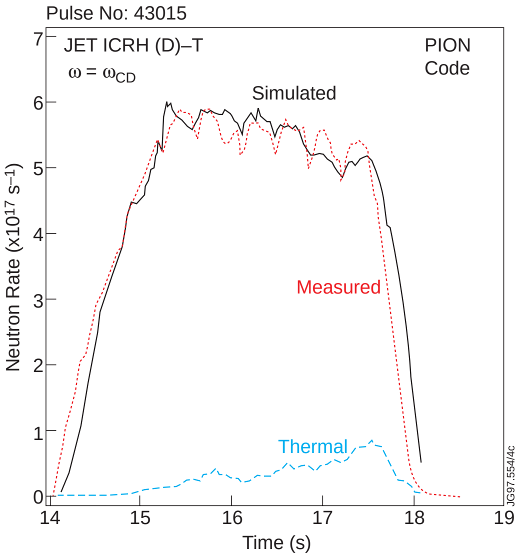


Fig. 2.1. Comparison of the simulated and measured deuterium-tritium (DT) neutron rate for $\omega \approx \omega_{cD}$. Also shown is the simulated thermal DT neutron rate.

Among the notable rf physics effects that have been identified is and ICRF-induced trapped particle pinch. Evidence of wave-induced fast-ion radial drift was observed in ICRH experiments on JET, which can be important from the point of view of adjusting fast ion distribution and fusion reactivity in reactors.

ASCOT has been interfaced with JET magnetic background making it possible to calculate heated ion distributions for JET, and to study their sensitivity to ion wave phasing. Work in progress includes using ASCOT to evaluate ICRF-heated ion distributions and comparing them against JET experimental data. ASCOT is also used to study transitions in plasma confinement at JET.

2.1.2 JET Task: Development of Radio-Frequency Modules for Transport Codes

Development of lower hybrid radio-frequency modules for the transport codes used at JET was started during 1998. First, the code will be validated against JET experimental data, and then the rf- modules will be implemented into the transport codes. Finally, experimental data from JET will be analysed using the integrated rf- and transport codes.

This work is a continuation to the ray-tracing studies for lower hybrid current drive which were performed with the coupled ASTRA transport code and the Fast Ray Tracing Code FRTC, developed at the Ioffe Institute, St. Petersburg. The ray-tracing studies addressed the role of the so-called fast wave in lower hybrid current drive. Initially, the lower hybrid waves launched from the grill are so-called slow waves, but part of the wave power may transform to fast waves, which have different propagation and absorption properties. The results show that, at high plasma densities, up to half of the wave power can be absorbed as a fast wave in JET and ITER.

2.1.3 ITER Task: Support of Physics and Engineering Design of the Ion Cyclotron System

Mode Conversion and Minority Current Drive for Plasma Current Control. In a mode conversion, ion cyclotron waves are converted into electrostatic, short-wavelength waves, that are rapidly absorbed to the plasma. A high conversion factor would allow using ion cyclotron waves in mode conversion current drive. Mode conversion of ion cyclotron waves has been modelled numerically and, in certain conditions, 100% conversion has been demonstrated. Recently, in a number of experiments world-wide, the high conversion have been verified. A parameter analysis of the conversion coefficients and optimisation of antenna spectrum have been performed for ion cyclotron heating scenarios of ITER. It shows that mode conversion current drive is difficult to realize in the large-size ITER because of competing absorption mechanisms.

Current drive with ion cyclotron wave minority absorption has been studied by solving the drift-kinetic Fokker-Planck equation, and by performing full Monte Carlo simulations with ASCOT for the ICRH scenarios of ITER. Diamagnetic effects appear to dominate the current generation over the

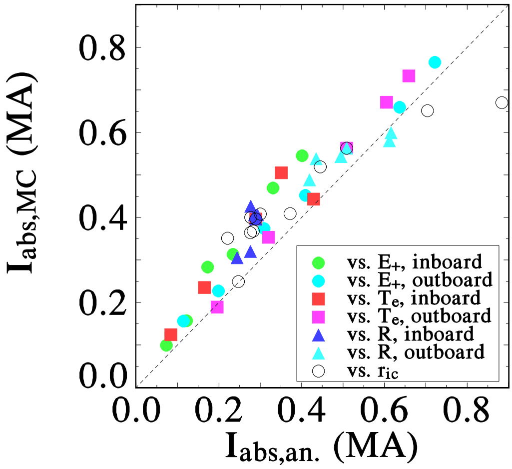


Fig. 2.2. Comparison of driven currents from the simulation and the analytical estimate.

momentum transfer and asymmetric heating mechanisms. The driven current is only found to be significant in ITER for hydrogen minority heating, which at the moment is not one of the candidates for ITER.

We have developed an analytical estimate to describe driven current in a wide parameter range. Fig. 2.2 shows the results obtained from a Monte Carlo simulation, together with the analytical estimate for the integrated absolute current. The fairly good agreement between the analytical results and the simulations supports the conclusion that most of the driven current originates from the fast ion diamagnetic current. Also, the dependencies of the current on the wave and plasma parameters, obtained from toroidal Monte Carlo simulations, are satisfactorily reproduced by the analytical estimate.

Folded Waveguide Antennas as Advanced Launchers for Ion Cyclotron Radio-Frequency Heating. The requirement of reducing the size of the rf-launchers in a compact ITER device may call for advanced antenna solutions. A folded waveguide (FWG) and dielectric-filled waveguide (DWG), see Fig. 2.3, are two candidates. The power handling and coupling of the antenna arrays have been modelled numerically for ion cyclotron heating in ITER reactor conditions. For the first time, the field structure has been reconstructed self-consistently both inside the waveguides, and in the vacuum gap between the coupler front and the plasma. The electric field values stay within the experimental breakdown limits for an antenna array radiating three times more power than conventional loop antenna arrays, but still fitting in an ITER port.

A three-dimensional plot of the absolute value of the electric field at the plasma surface is shown in Fig. 2.4. Assuming 60 MHz frequency and specific ITER-relevant waveguide, the present model with idealized non-perturbed feeder excitation predicts a voltage less than 40 kV along the current probe source, a maximum electric field below 30 kV/cm, and a maximum voltage of the order of 250 kV inside the FWG with ten folds for 10 MW radiated power per unit. For a waveguide complex composed of eight units, the mutual coupling between the units is found to be non-negligible, indicating a need for tuning in the system circuit. Work is in progress to model the feeder excitation inside the waveguides.

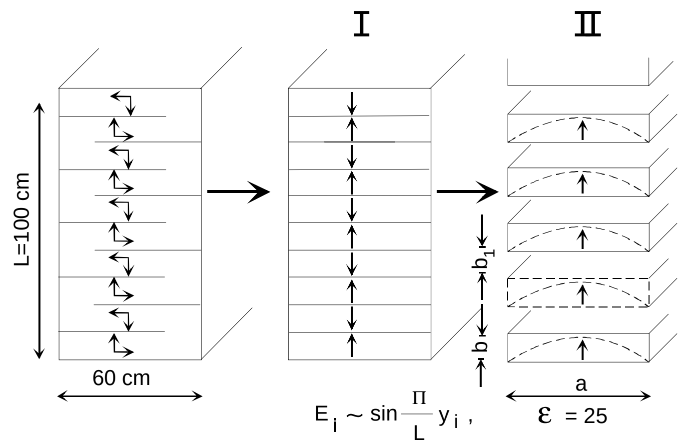


Fig. 2.3. Folded waveguide (FWG) and dielectric-filled waveguide (DWG) units with the same aperture geometry. In the FWG geometry, every second aperture in the subplot I is covered with a front plate so that the same aperture geometry results as for the DWG geometry sketched in the subplot II.

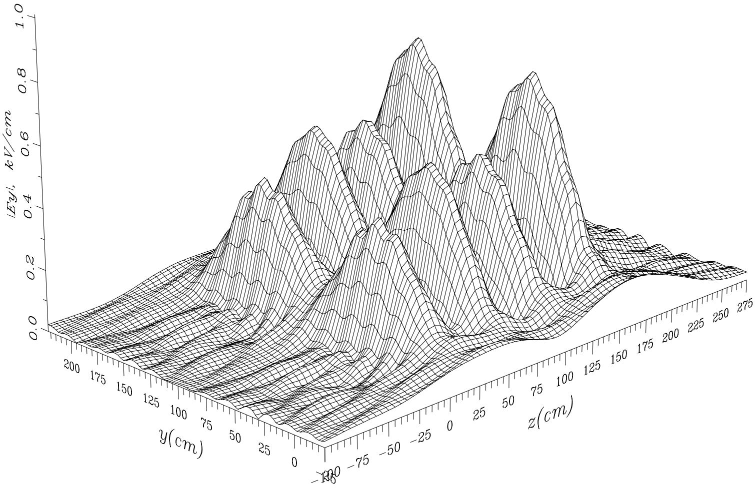


Fig. 2.4. Electric field in the vacuum layer at the plasma surface for a 4×2 FWG array, P_{rad} =32 MW and vacuum gap is 10 cm.

Alpha Power Channelling with Waves. It has been suggested that locally constrained waves could be used for converting fusion alpha particle power to fuel ion energy and for enhancing alpha particle removal from the fuel. 5D Monte Carlo simulations with ASCOT, taking into account a realistic alpha particle birth distribution, full collision operator, and a realistic tokamak geometry, indicate that there are severe problems in implementing this scheme for standard heating methods. However, ASCOT calculations for frequency-chirped Alfven eigenmodes appear promising, and may suggest a scheme for harnessing alpha power for current drive, and for power channelling to fuel ions.

2.1.4 Tore Supra Collaboration: Analysis of Parasitic Absorption of Lower Hybrid Power

Generation of impurities has been observed in Tore Supra and Tokamak de Varennes (TdeV) when lower hybrid waves at frequencies of 3.7 GHz have been launched. At Tore Supra, heat fluxes of 5–10 MW/m^2 on plasma facing components have been measured by infrared video imaging. Toroidally asymmetric heat loads have been observed on the divertor plates and limiters of TdeV. Melting of the grill mouth due to strong local heating has occurred in JET, as well as on LH grills in other tokamaks.

A possible explanation for the impurity production is sputtering caused by fast electrons generated by the near field of the rf-launcher. Such electrons can be generated when part of the rf-power is absorbed within a short distance from the launcher. When the launched power is several megawatts or even tens of megawatts, fast electrons containing a few percent of the rf-power may damage the launcher or limiter structures.

The parasitic absorption of lower hybrid waves and the generation of the fast electrons near the launcher has been investigated via particle-in-cell (PIC) simulations in collaboration with Tore Supra and JET. A particle-in-cell model of a lower hybrid grill has been developed and coupled with the SWAN code, which calculates the launched wave spectrum.

Fig. 2.5 shows the wave potential in a simulation performed for Tore Supra. The fine structure close to the grill mouth is due to the short-wavelength modes emitted by the antenna. Further away from the grill, the field structure is very smooth because the short-wavelength part of the spectrum is absorbed within a distance of a few millimetres from the launcher.

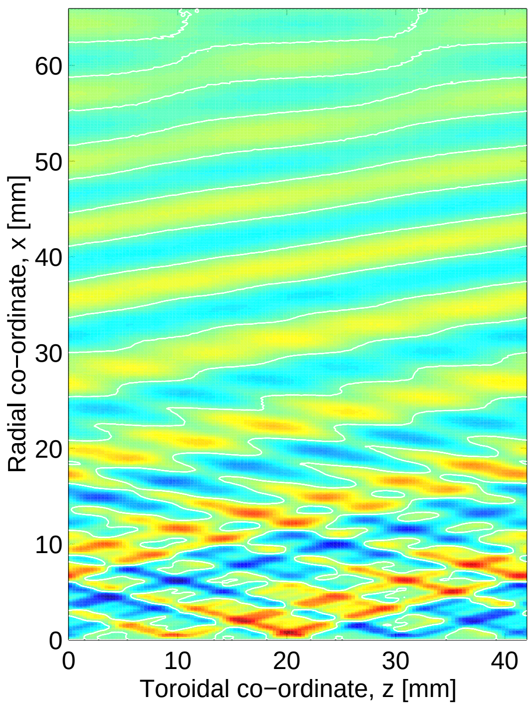


Fig. 2.5. Contour plot of the potential of the lower hybrid wave obtained from a particle-in-cell simulation of the Tore Supra lower hybrid grill. The grill mouth is located at the bottom and the plasma density increases upwards.

Fig. 2.6 shows typical absorption profiles calculated for JET and Tore Supra when the coupled rf-intensities are 25 MW/m^2 and 48 MW/m^2, respectively. The parasitic absorption is peaked very close to the plasma edge and 210 kW/m^2 is absorbed in JET within a distance of one millimetre. In the simulation for Tore Supra at higher rf-intensity, the parasitically absorbed power is 730 MW/m^2.

The amount of absorbtion in the near field of the grill depends strongly on the launched spectrum and is typically 0.5–10% of the launched power. The results of the numerical calculations are in rough agreement with the experimental results from Tore Supra, where the parasitic absorption stays below 2%, and TdeV, where it can exceed 10%.

In simulations, fast electrons are generated with keV-range energies, which is compatible with measured values from a few hundred eV to several keV. The particle-in-cell model indicates that the parasitic absorption and the fast electron generation occur within a very short distance, of the order of one

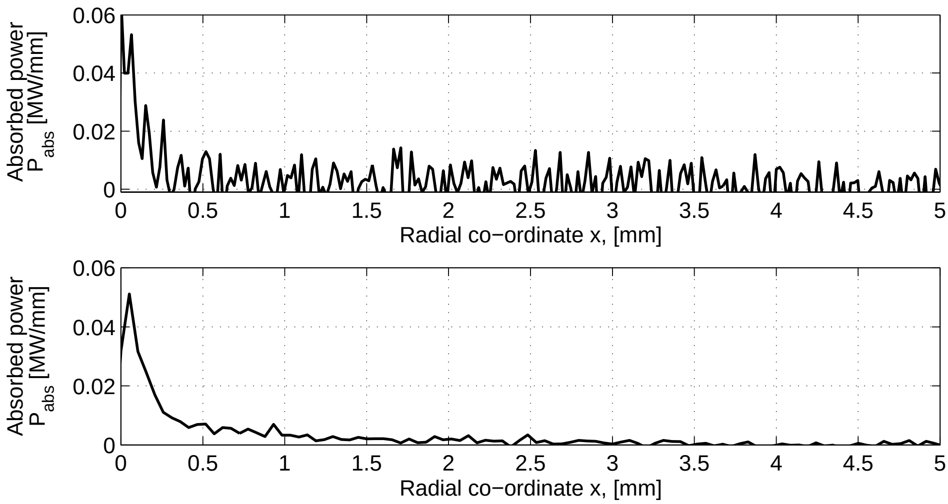


Fig. 2.6. Absorbed power per unit length versus radial co-ordinate in the near field of the lower hybrid launchers of JET (top) and Tore Supra (bottom).

millimetre, which is shorter than the experimental result. A satisfactory explanation for this difference is missing at the moment.

Modelling of lower hybrid launchers at JET was performed as a part of the above described JET Task Agreement on “Development for RF Modules for Transport Codes”. Corresponding calculations for ITER are in progress.

2.1.5 Gyrotrons for Electron Cyclotron Heating and Microwave Diagnostics

Electron-cyclotron heating is playing an increasingly important role in tokamak and stellarator plasma research. The power for this heating method is provided by gyrotrons.

Conventional hollow waveguide-cavity gyrotrons are already commercially available. However, they are limited in both output power and frequency (about 1 MW, 140 GHz). Current issues in gyrotron research and development include techniques for increasing the efficiency of gyrotrons above 50%, and techniques for increasing the unit power of gyrotrons to over 3 MW at frequencies of about 170 GHz. Only gyrotrons with coaxial cavities have the potential to meet these requirements, and are in the front line of the research.

The gyrotron research in Finland started in 1993. It is done at the Helsinki University of Technology. There are no gyrotron experiments. The research work can be grouped as follows:

- Participation in theoretical work on the world-wide development of specific advanced gyrotrons.
- Development of the general gyrotron theory.
- Data analysis for various experiments (e.g. W7-AS stellarator) where gyrotrons are used for plasma heating, current drive, and diagnostics. The results of this analysis are used as feedback for gyrotron research.

The most important achievements are summarized below.

Resonator design. Resonators have been designed for specific gyrotrons, e.g., the 140 GHz, 1.5 MW, $TE_{28,16}$ mode coaxial gyrotron developed in a collaboration between the Forschungszentrum Karlsruhe in Germany and the Institute of Applied Physics at Nizhny Novgorod in Russia; and for the 280 GHz, 1 MW, $TE_{23,16}$ mode and 140 GHz, 3 MW, $TE_{21,13}$ mode coaxial gyrotrons developed at the Massa-chusetts Institute of Technology in USA. Nonlinear calculations of mode competition in these resonators, including the effect of electron beam velocity spread, have been carried out using numerical tools developed at the Helsinki University of Technology.

The third harmonic 280 GHz quasi-optical gyrotron designed at the Centre de Recherches en Physique des Plasmas of the Ecole Polytechnique Federale de Lausanne in Switzerland for plasma diagnostics has been analyzed. A new special code, taking into account the space-charge effects as a source of electron energy spread, has been developed and applied. Based on results obtained from this code it has been proposed that the magnet system of this gyrotron should be modified to increase its efficiency.

General theory of gyrotrons. A general theory describing symmetry breaking in coaxial cavities has been developed. This theory includes both the resonator wall - electron beam and the resonator wall - coaxial insert eccentricities.

New schemes of tuning the frequency of a gyrotron have been proposed and investigated. These include the fast frequency step tuning by means of changing operating voltages, and the continuous frequency tuning by moving the inner conductor in a coaxial cavity. A design of a multifrequency gyrotron, i.e., a gyrotron which generates microwaves simultaneously at several frequencies, has been presented. For this purpose, the multimode, time-dependent and self-consistent codes developed at Karlsruhe and Nizhny Novgorod have been modified and improved.

A general theory describing the effect of fluctuations in technical parameters (voltages, beam current, external magnetic fields) on the linewidth of gyrotron radiation has been developed.

Various schemes of transition of gyrotron radiation from regular to chaotic regimes have been investigated. It was shown that chaos which can develop in a resonator for some values of control parameters can be only transient.

Use of specific gyrotrons in tokamaks and stellarator. A gyrotron has been used to study effects of the off-axis plasma heating at the W7-AS stellarator at the Max-Planck-Institut für Plasmaphysik at Garching in Germany. In particular, it has been shown that such a heating leads to electron density increase at the plasma center.

Suitability of frequency tunable gyrotrons for various plasma heating and diagnostics applications has been investigated. These include the possibility of using a single gyrotron for plasma heating and collective Thomson scattering experiments, as well as for stabilization of tearing modes.

INTAS coordination. An INTAS project was co-ordinated in 1995–1997 to study modern frequency-tunable gyrotrons, in order to improve the diagnostics and heating schemes in fusion plasmas. Fast and slow, discrete and continuous tunability of the source frequency, as well as new methods to widen the frequency window at high power, were investigated.

A fully kinetic code for calculating the collective Thomson scattering cross section was developed in the sub-millimeter and millimeter wavelength regime. The code has been validated by benchmarking against the warm plasma codes at JET, and has been used for the design of microwave scattering diagnostics of fast particles.

2.2 Plasma Confinement and Transport

Confining the hot fusion plasma is a fundamental problem en route to a commercial fusion reactor. The initial, optimistic predictions on the availability of fusion power were based on the assumption that the transport of heat and particles in a tokamak geometry were solely due to Coulomb collisions. Experimentally, however, it has been found that the plasma transport is strongly driven by turbulence, thus leading to poorer confinement characteristics than anticipated. A comprehensive transport theory for a tokamak plasma is still being fomulated, while experimentally various regimes of improved confinement have been discovered. In certain circumstances a tokamak plasma is found to make a rapid transition from Low confinement- or L-mode to High confinement- or H-mode. The mechanism behind these transitions is yet to be understood, but it seems that a radial electric field plays a crucial role. Because attaining good confinement properties is crucial for a commercial fusion reactor, efforts to understand and control L-H transition are of utmost importance.

2.2.1 ASDEX Upgrade Agreement: Transition from Low to High Confinement

Collaboration between VTT Energy, HUT, and Association IPP Garching to carry out detailed simulation studies to explain the L-H transition characteristics at ASDEX Upgrade in IPP started in 1996. ASCOT has been interfaced with the ASDEX Upgrade background data, including the ripple in the magnetic field, neutral beam injection, and charge exchange diagnostics. Presently, this code can consistently follow electromagnetic fields generated in the plasma, and is therefore appropriate for studies of transport and neoclassical mechanisms.

<u>Detecting radial electric field by NPA of ripple-trapped slowing-down ions.</u> Several theories that include the radial electric field have been proposed to explain the mechanism with which a tokamak plasma switches from L- to H-mode. It is widely believed that the sheared E×B-rotation associated with a non-uniform radial electric field suppresses turbulence and thus reduces anomalous transport. However, the time resolution of the spectroscopic measurements probing the dynamics of the L-H transition has not yet been good enough (about 0.5 ms on DIII-D) to decide whether the changes in the radial electric field are

sufficiently fast, and if they indeed precede the L-H transition.

In neutral beam heated discharges on ASDEX Upgrade, neutral particle fluxes, originating from slowing-down ions trapped in local magnetic ripples, are observed to evolve at the L-H transition: the flux starts increasing at the transition, and the increase in a given energy channel starts the earlier the lower the energy of the channel. As a result, the energy spectra of these neutrals are qualitatively different in L- and H-modes. The particle flux also responds to ELMs (Edge Localized Modes), with the particle flux collapsing simultaneously with the observed increase in the Dα-signal.

Monte Carlo calculations using ASCOT code have reproduced the experimentally observed neutral particle analyzer (NPA) signal by turning on a radial electric field near the plasma periphery, and assuming a background neutral particle density profile consistent with experimental conditions. The flux for relevant values of poloidal and toroidal angles is shown in Fig. 2.7 both in the presence and absence of the radial electric field. The signal is found to respond fast (in about 50-100 μs) to the appearance of the electric field, when the negative electric field has a relatively large radial extent (> 1-2 cm). The strong depletion in the absence of a radial electric field is due to the very fast, uncompensated downward drift of the ripple-trapped ions. The Monte Carlo analysis could not, however, but hint what would be the mechanism that repopulates the depleted region, once the radial electric field is introduced.

Using 3-D Fokker-Planck analysis, convective filling from the inner, well-filled ripple domain has been identified to be the main mechanism behind the fast growth in the NPA signal after the onset of the electric field. Near the plasma periphery, at small poloidal angles, a strong suppression in the deeply ripple-blocked ion distribution is observed in the absence of a radial electric field. The deficit of the ripple-blocked ions is found to be rapidly filled by the onset of an inward radial electric field of a sufficient magnitude. The fast time scale of the filling is explained by the different drift orbit topologies generated by the radial electric field, and it is determined by the convective drift time of the ions along these orbits as seen in Fig. 2.7. Consequently, the time scale can be much faster than the collisional time scale, and the blocked ion distribution function should faithfully follow the changes in the radial electric field.

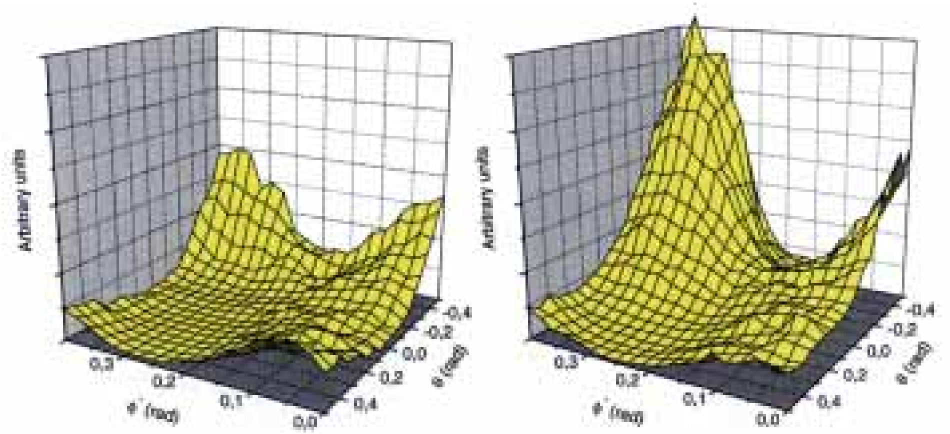


Fig. 2.7. A 3-D plot of the neutral flux from the simulation as a function of the toroidal (φ) and poloidal (θ) angle. The detector's energy window is from 5 to 15 keV, and the pitch window is $|\xi|<0.07$; (a) without a radial electric field; (b) with the radial electric field.*

The fast response of the NPA signal to the radial electric field assists in diagnosing the dynamics of the radial electric field in the transition from L- to H-mode. The obtained experimental and simulation results appear to exclude a fast (<< 1 ms) jump in the radial electric field at ASDEX Upgrade, at least for a finite halfwidth of the field radial profile.

<u>Radial electric field generation at L-H transition.</u> One of the most popular theories for L-H transition assumes that the generation of the radial electric field is due to direct ion orbit losses. With ASCOT, we have evaluated the ion orbit loss trajectories in ASDEX-Upgrade plasma using a toroidal magnetic coordinate presentation appropriate both inside and outside the separatrix. The ion orbit loss current has been calculated for the real ASDEX Upgrade geometry. The theory, based on a multivalued balance between the orbit loss current and neoclassical current, was then tested by comparing the calculated orbit loss current to an analytical estimate for the neoclassical return current. The approximate analytical theory, based on cylindrical geometry, predicts that a bifurcation happens for normalized collisionality $\nu_{*i} \approx 1$. This holds true also for more realistic calculations, carried out for the ASDEX Upgrade shot 8044, with collisionality $\nu_{*i} \approx 3.8$. The ion orbit loss current ensuing from neoclassical diffusion is far too small to explain the bifurcation for the experimental conditions of ASDEX Upgrade.

In order to evaluate the dynamics of the radial electric field, using ASCOT, the ion trajectories have been calculated self-consistently with the radial electric field obtained from the Maxwell equations including the radial currents by parallel and perpendicular viscosity, and polarization. The simulations carried out for a wide collisionality regime show a strong but narrow negative radial electric field just inside the separatrix even in the plateau regime. An example of the time dynamics of the electric field radial profile is shown in Fig. 2.8. This field, which arises from the particle loss mechanism taking place over the separatrix, does not experience bifurcation, but a gradual broadening and strengthening as the collisionality decreases.

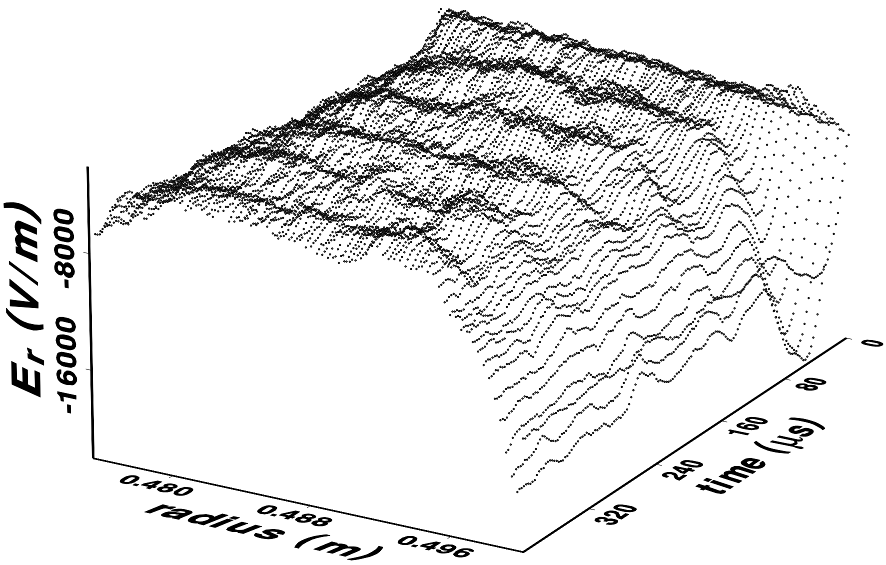


Fig. 2.8. Radial electric field profile as a function of time as calculated with a gyrokinetic version of ASCOT for ASDEX Upgrade L-H transition conditions.

2.2.2 Electron Density Profile Measurements and Particle Transport Studies with Multichannel Interferometer at the Wendelstein 7-AS

A 10-channel microwave interferometer was build 1995 for the Wendelstein 7-AS stellarator in Garching. The interferometer started its operation in June 1996. Within the agreement between the Helsinki University of Technology and Max-Planck-Institut für Plasmaphysik (IPP), software for interpreting data from measurements has been developed. The electron density profile has to be reconstructed from line-integrated interferometer data. The reconstruction problem is similar to tomographic inversion problems. A new algorithm based on Fisher-information as the regularizational functional was developed, and corresponding software built. The new method was checked and benchmarked by comparing the resulting density profiles to density profiles measured with Thomson scattering, Lithium beam-diagnostics, and reflectometry. The agreement was very good. The density profile reconstruction software has been a part of everyday diagnostics at W7-AS since 1996. Since the interferometer has a good time resolution, it is possible to study fast changes in plasma density.

The main emphasis in the interferometer measurements was in transient particle transport studies. Gas feed to the plasma was modulated harmonically producing a density perturbation, which propagated from the plasma edge to the centre. Diffusion coefficient and the convective velocity of the plasma electrons can be determined from the magnitude and phase of the propagating density perturbation, both of which can be extracted from the interferometer data. An example of a large magnitude and slow density perturbation can be seen in Fig. 2.9, where the density oscillates during modulated gas puffing.

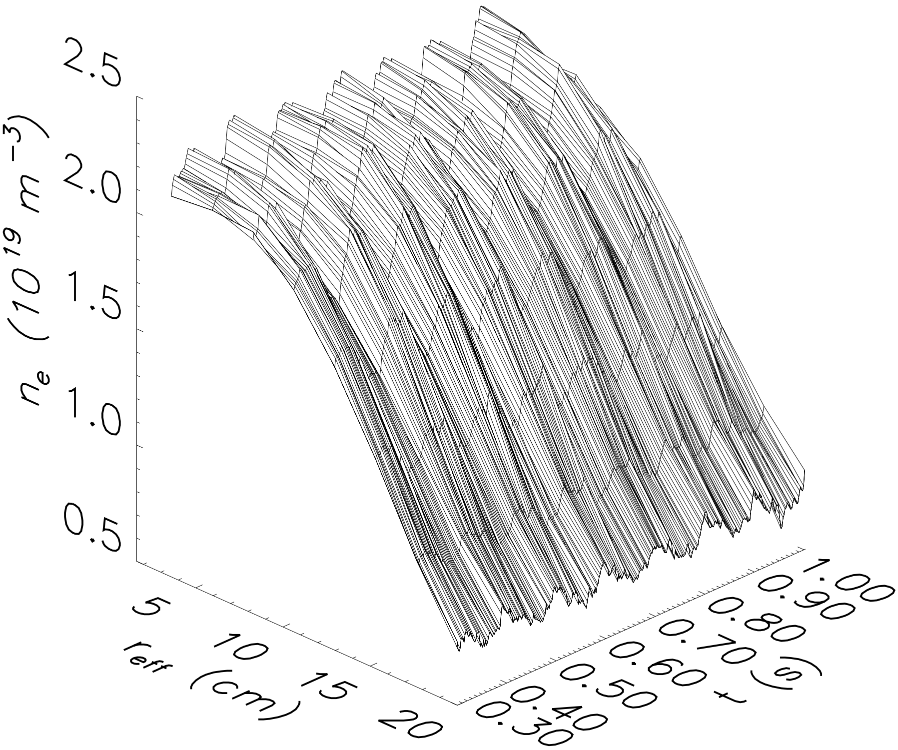


Fig. 2.9. Temporal evolution of the plasma density during gas feed modulation experiments in W7-AS. Large density perturbation propagates from the plasma edge to the centre. The density profiles are reconstructed from the multicannel interferometer data.

The magnitude and phase data was modelled by radial Fourier-transformed particle transport equation, and fitted by adjusting suitable transport coefficients. Constant diffusion coefficient was sufficient to describe the propagation in the inner plasma, but the behaviour in the edge region required introduction of an inward convective term.

The diffusion coefficients from extensive scaling studies at W7-AS were compared with corresponding particle balance diffusion coefficients. The agreement was good, which excludes a strong dependence of diffusion coefficient on density gradient. The diffusion coefficients were also comparable to the neo-classical diffusion coefficients.

In January 1998, a new observation was made with the multichannel interferometer. Radially peaked density profiles were detected in discharges with small central particle sources. Transport analyses revealed the existence of inward pinch in the core plasma. There is evidence that in discharges with peaked temperature profiles, the inward pinch is cancelled out by outward directed thermodiffusion.

2.3 Dielectric Window Prototype for the Reactor Vacuum Transmission Line of Ion Cyclotron Power

The ITER Tasks on "Dielectric Window Development for the ITER Vacuum Transmission Line of Ion Cyclotron Power” were conducted 1995-1998 at VTT, IVO Technology Centre, and Helsinki University of Technology. The goal was to obtain specifications for constructing two prototype vacuum windows for the ITER-like vacuum transmission line. The task objective was to present a design of the window which is compatible with the ITER radiation fluence, withstands the strong dielectric heating and related thermal stresses, is resistant against breakdown with appropriate arc monitoring, can be remote handled, and can be manufactured by welding the ceramics to the conductor.

2.3.1 Requirements for a Reactor Window

The double dielectric window is an essential and the most delicate component of the ICRF Vacuum Transmission Line for ITER, as it provides the ultimate vacuum, as well as tritium containment. In the current baseline design, there are eight windows for each of the four ICRF arrays. The ITER window requires a specific design and careful selection of the dielectric material because of the long ICRF pulse length, high electric field strengths, possible degradation of the dielectric properties due to neutron or gamma irradiation, and possible changes in mechanical and thermal properties and in gas permeation. Furthermore, the metal-ceramic joints required for the windows and the support structures need to retain reliable vacuum tightness under cyclic operation conditions.

In the ITER vacuum transmission line, the window assembly will be located at the feedthrough of the vacuum vessel. The double window structure includes a dynamically evacuated intermediate region between the ceramics, cooling of the inner and outer conductor, potential rings to reduce the tangential electric fields close to the joints, and the joining structure. The design is based on the maximum of 50 kV peak RF voltage, with arbitrary amplitude modulation, and matching 30 Ohm line. A remote handling tool which can approach the window from the generator side through the interspace between the inner and outer conductors is considered. With such a tool the window could be cut from the outer conductor and could be withdrawn inside the outer conductor. This type of arrangement is obviously possible if the window is placed at the vacuum vessel feedthrough and is inserted in a casing attached to the outer transmission line conductors on both side of the window as schematically shown in Fig. 2.10.

2.3.2 Design

A Monte Carlo program MCNP4A was used to evaluate the neutron fluence at the window for 1500 MWth reactor power. The fast neutron flux at the vacuum windows is about 7×10^{10} n/cm^{2}s, if they are not shielded by an additional plug closing the port. By closing the end of the equatorial port with a 55 -30 cm thick plug of steel and water, and locating the vacuum windows outside this plug, the fast flux will be 5×10^8 - 5×10^9n/cm^2 s.

By inspection of the material data of irradiated ceramics, unirradiated data for the loss tangent, thermal conductivity, and mechanical and electrical strength can be used for alumina and beryllia with fluences below 10^{16} n/cm^2. For fluences up to 10^{18} n/cm^2 the changes can also be regarded small (<30%). The dielectric heating calculations with a disk shaped ceramic showed that for the dielectric material in real construction, BeO or Al_2O_3 (97.5%) (latter only in unirradiated situations) can be accepted, and that only titanium and niobium are reasonable alternatives for conductor. This is due to the small difference in the value of the thermal expansion coefficient between these metals and the ceramics.

Extensive finite element calculations of the temperature, stress, and electric field on the vacuum window were performed to optimise the conical-shaped dielectric geometry in a coaxial with proper material choices. BeO ceramic and titanium conductor were chosen to minimise the dielectric heating, alleviate the stresses, and to help the brazing. Based on the obtained fluences, it has been decided to place the window at the vacuum vessel feedthrough.

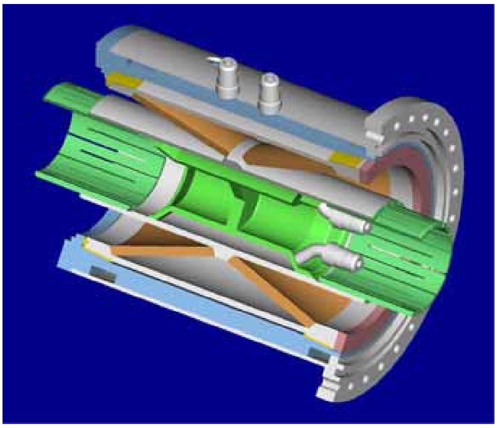


Fig. 2.10. Sketch of the prototype vacuum window immersed in a stainless steel casing and with bullet-type inner conductor connections to the rest of the transmission line. Water cooling at both ceramic-conductor joints (inner and outer) is arranged. The window was designed at VTT Energy, VTT Manufacturing Technology, IVO Technology Centre, and Helsinki University of Technology

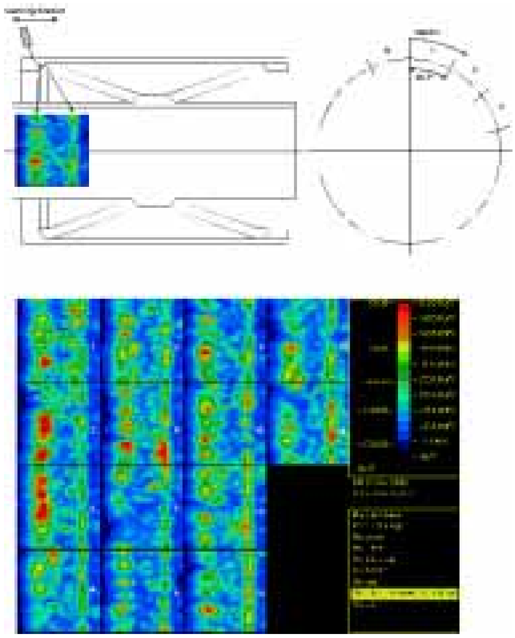


Fig. 2.11. The C-SAM image from the joint at the larger end of the pre-prototype No 1 showing discontinuities in the brazed joint.

The optimized shape for the ceramics is conical with an angle of inclination of approximately 18^{o}. In the present model, the casing is thermally connected to the outer conductor at the ends above the ceramic/outer conductor joint. A sketch of the design is shown in Fig. 2.10. An X-shaped geometry of the two ceramics has been adopted as suggested by B. Walton and A. Kaye (JET, 1996), because it gives a comparable strength as the corresponding one with two parallel conical septa while decreasing the tangential electric field. The maximum tangential field obtained is 0.60 MV/m at the inner side, and the maximum field normal to the surface of ceramics is 0.19 MV/m. The electric field levels which remain significantly below 2 MV/m are not expected to be prone to a breakdown discharge along the ceramics.

Temperature time histories are evaluated using discharge duration of 1000 s and a volume 1000 cm^3 of the ceramic. The source of the volumetric heating is a high frequency electric field, together with the Joule heat generated by the current in the titanium conductors. Ohmic losses with Ni coating will be two times lower. The maximum temperature was found to be 270 ^{o}C at the centre of the Al_2O_3 ceramic and 185 ^{o}C at the centre of the BeO ceramic, when the ohmic heating of the conductor was 750 kW. A stationary temperature was reached after 400 s with maximum principle stress less than 125 MPa . The stress values at different positions of the window are given in Table 2.1.

Table 2.1. Maximum stresses (MPa) in the models.

Component	**Location**	**Stress/ Al_2O_3-model**	**Stress/ BeO-model**
Max. Principal stress (tensile)	Casing (near interface between casing and outer conductor)	122	68
Max. principal stress (tensile)	Ceramics (interface between casing and outer conductors)	80	78
Maximum shear stress	Casing (interface between casing and outer conductor)	61	56
Maximum shear stress	Ceramics (interface between outer conductor and ceramics)	103	30

2.3.3 Joining Experiments

The choice of titanium as the conductor material was necessary to provide a design with water duct cooling in conductors without active gas cooling on ceramics. However, joining ceramics to titanium at this scale, with ITER requirements for vacuum tightness, has not yet been demonstrated. Successful shear stress and leak tests for brazed reduced-size Ti/alumina specimens were performed. The objective of the joining experiments was to optimize both the manufacturing of the joint surfaces and the bonding process for vacuum brazed Al_2O_3 -Ti (Grade 2) joints. The main service requirements are leak tightness of the joint, good thermal conductivity over the joint, and relatively massive material thickness to avoid diffusion of tritium through the component. To avoid mechanical attachment during the manufacturing of the component, vacuum brazing technique was considered as a method for producing high quality joints.

The brazing alloy was chosen from the commercially available alloys. The first screening of the joining process was carried out in preliminary tests. Five potential brazing alloy and bonding process combinations were chosen for the second round of tests. The second screening was carried out by performing vacuum tightness and shear strength tests. The third stage was to study the application of the filler materials, and the effects of the circular joint gap geometry in a real window application.

The He-leak test samples were made with a tube-to-plate configuration. Before the He-leak test, six of the bonded samples were heat-treated. A thermal cycling from room temperature to 200°C was repeated ten times to simulate the thermal conditions at the joints in service. In the He-leak test, the specimens were first, one by one, tightened mechanically with the aid of a clamping chuck on the chromatographic apparatus. Based on the test, all the samples except those brazed with the alloys CB6 and Gold ABA, are acceptable. The rejection of these brazing alloys was due to a porous structure of the joint revealed by metallography.

Non-Destructive Evaluation (NDE) using ultrasonic examination was performed for butt-joined samples (used for the shear strength tests) with the sample immersed in water, and the ultrasonic transducer was scanned in an X-Y-pattern over the sample. The results were in accordance with the vacuum tightness test results.

The verification of the potential manufacturing procedures obtained from the experimental phase was done with a full-scale alumina to titanium component sketched in Fig. 2.10. The first trials were done with one ceramic window and half of the titanium tubes. The weight of such a half-component is around 7.3 kg, being relatively massive for a ceramic-to-metal component. The main conclusions from these full-scale pre-prototype tests are:

- the amount and spreading of the brazing filler material must be optimised carefully
- the type of the brazing alloy is critical, not only the brazing temperature but also the wetting properties must be taken into account
- the preheat treatment and the dimensional control of the metallic titanium parts are essential in order to control the dimensional changes in titanium during the bonding cycle
- the accurate finishing and tight tolerances of the joined parts are important for the proper bonding.

Fig. 2.11 shows the C-SAM image from the joint at the larger end of the prototype showing discontinuities in the brazed joint. Work shall continue to improve the quality of the joint in the post-EDA phase after 1998, together with the production of the prototypes as well as with the design and production of new ITER-like vacuum windows for high power density rf-launcher prototypes. In particular, a vacuum-tight brazing of full-sized BeO ceramics to the titanium housing is to be experimentally verified.

2.4 Central Solenoid Development for Spherical Tokamaks

Central solenoid is the most critical magnet component in tight aspect ratio tokamaks. Its function is to provide inductively the plasma current required for confinement. In present devices, water-cooled copper conductor is used for its winding. A project to design and construct an appropriate solenoid conductor for spherical tokamaks started in 1994 as a Tekes project at VTT, HUT and Outokumpu Poricopper. In due course of the project, a manufacturing method was developed for producing a world record of 66m-long high- strength hollow conductor and a full specimen was delivered to the Efremov Institute in St. Petersburg, where it is wound and tested during 1998. Outokumpu Poricopper provided the required solenoid conductor also to the MAST tokamak in Culham.

2.4.1 Design Characteristics

In the Globus-M spherical tokamak under construction at Ioffe Institute, St. Petersburg, the goal is to generate and sustain a plasma current of up to 0.5 MA for 0.2s pulses. The central solenoid conductor material must provide reliable operation over a lifetime of at least 4×10^4 full power shots with a flux swing every 10 minutes, which requires heat removal by water cooling. It is important also to wind the solenoid from a continuous conductor, precluding the need for additional electrical contact joints. High magnetic and thermal cyclic loads acting on the solenoid require that it be manufactured from high strength hollow conductor. CuAg0.1(OF) was selected as the conductor material.

2.4.2 Solenoid Design, Fabrication, and Testing

A solenoid prototype with full scale radius and decreased length (one-sixth of full length) was fabricated from a ten-meter piece of silver-bearing hollow copper conductor. The conductor after extrusion was cold-drawn with a 30% reduction in area, and subjected to mechanical static and cyclic tests. In the experiments, the tensile load amplitude was in the range 140-220 MPa, which is greater than the stress intensity in the solenoid’s inner layer, where it reaches 127 MPa near the cooling hole. The conductor fractured after 2×10^5 cycles with an applied load of 220 MPa. The mechanical properties of silver bearing copper were measured at room temperature with virgin material and with material initially strained cyclically using a load amplitude 150 MPa for 6×10^5 load cycles. This load did not reveal any significant variation in mechanical strength.

The optimum conductor profile was designed by using a finite element code DEFORM 3D to simulate the bending process. To verify the simulations, several test pieces of the conductor were manufactured and bent to the required radius. The experimental results were in good agreement with the calculations.

A 2D axisymmetric finite element model of the solenoid was developed to analyze the stresses. Within a range of currents up to 92 kA, the measured strains had an elastic character. The values of Tresca stress, calculated from strains measured at the horizontal midplane, are plotted in Fig. 2.12 as a function of current. The stress averaged over two layers was well below the static allowable limit for the conductor material. The shear stress did not exceed 10 MPa in the outer turn of the insulation. The conductor was wrapped in three layers with polyimide tape, followed by one layer of protective glass fibre tape.

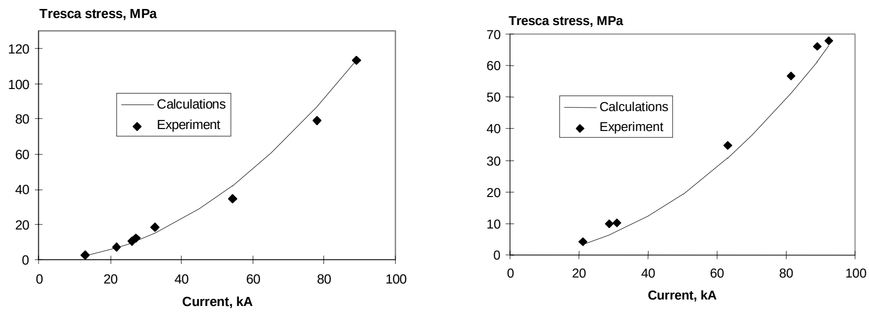


Fig. 2.12. Measured and calculated Tresca stresses at the solenoid prototype inner and outer surface as a function of current.

3 Fusion Reactor Materials

The selection of materials and joining technologies to be used in ITER is a trade off among multiple and often conflicting requirements derived from the unique features of the fusion environment. Material selection must encompass a total engineering approach, by considering not only physical and mechanical properties and processing, but also the maintainability and reliability of each material. As far as technically feasible, the material choice has been oriented toward industrially available materials and well established manufacturing techniques. This is very much the case for the structural materials of the basic machine e.g. cryostat, magnet case and vacuum vessel for which a critical factor is the availability of industrial suppliers with experience in forming and joining technology. The structural integrity of these components after manufacturing and throughout the entire design lifetime is important for the machine availability and safety. The materials for the in-vessel components will operate under the simultaneous influence of different life-limiting factors such as neutron irradiation, hydrogen atmosphere, dynamic stresses, thermal loads, cyclic mode of operation and water cooling environment. Even though no safety functions are attributed to the in-vessel components to achieve good performance and adequate availability of the whole machine they have to remain highly reliable throughout the design lifetime. Ease of fabrication, good weldability, resistance to corrosion, good strength and fatigue resistance, adequate ductility and fracture toughness after neutron irradiation are essential requirements.

Austenitic stainless steel 316LN-IG (ITER Grade) is the most suitable for structural material of the ITER basic machine and in-vessel components as it is qualified in many national design codes, has satisfactory resistance to stress corrosion cracking and high level of strength and fracture toughness. There is also extensive database in the unirradiated and irradiated conditions and a large industrial experience in nuclear applications. Two copper alloys have been selected for the heat sink of Plasma Facing Components one age-hardened CuCrZr-IG alloy and one dispersion strengthened CuAl25-IG alloy. Mechanical properties of both alloys are sufficient for the components to sustain thermal and mechanical loads. For plasma facing materials carbon fibre composite, beryllium and tungsten have been selected as reference materials. Reference methods for component manufacturing is high temperature HIPing and brazing.

Main focus of fusion reactor material research work has been on characterisation of in-vessel materials and development of component manufacturing techniques applicable for ITER blanket modules. Critical properties of candidate copper alloys and their joints with stainless steel have been determined in the ITER relevant temperature and neutron fluence ranges. Fracture mechanical characterisation of dissimilar metal joints with special emphasis on miniaturised specimen technology have also been carried out. Explosion welding and hot isostatic pressing methods have been evaluated as possible manufacturing techniques for ITER primary wall module. Non-destructive examination methods suitable for examining dissimilar metal planar and tubular interfaces in ITER small scale primary wall mock ups have also been developed.

3.1 Characterisation of Irradiated Copper Alloys

The current design for ITER utilises copper alloys in the first wall and divertor structures. The function of the copper alloy in the first wall is to dissipate heat produced by plasma disruptions and therefore the copper alloy is not designed to provide structural support for the first wall. However, the copper alloy for the divertor is designed also for structural support of the divertor cassette in addition to heat dissipation. On the basis of the currently available data, the dispersion strengthened CuAl25 IG0 alloy is being considered as the primary candidate alloy and the precipitation hardened CuCrZr alloy has been chosen as the backup alloy. Within the frame work of the ITER technology programme, screening experiments are being carried out to determine the effect of irradiation on physical and mechanical properties of these alloys, however, there is very few results on the effect of irradiation on the fracture toughness behaviour of these alloys in the open literature.

3.1.1 Microstructure

The microstructure of candidate copper alloys is markedly different from each other due to different routes to produce the alloys. The CuCrZr alloy studied is produced by Outokumpu Poricopper Oy using conventional melting and hot rolling practices where the optimum strength is obtained by intermediate cold deformation before final precipitation heat treatment. The resulting microstructure consists of homogeneously distributed precipitates and equaxed grain structure. Mechanical properties of CuCrZr alloy are inherently sensitive to the temperature cycle and presently foreseen component manufacturing cycles do not alloy any intermediate water quench or cold deformation stages resulting to about 20-30% lower than optimum strength values for CuCrZr. The CuAl25 IG0 alloy is produced by powder production, internal oxidation together with extrusion and cross rolling which results in markedly heterogeneous microstructure and very fine grain size in CuAl25 IG0 alloy. However, due to stable alumina dispersions mechanical properties of CuAl25 IG0 alloy are practically insensitive to any temperature cycles foreseen in component manufacturing. Due to obvious differences in microstructure and differences in strengthening mechanisms both materials have some generic limitation in certain areas of fusion applications.

3.1.2 Fracture Toughness

The initiation fracture toughness (J_Q) of the CuAl25 IG0 alloy decreased significantly with increasing testing temperature and showed marked anisotropy as can be seen in Fig. 3.1. In longitudinal direction of the original plate the fracture toughness was somewhat higher than in long transverse direction but in short transverse direction the fracture toughness was 40-50% lower in the temperature range from 22^{o}C to 350^{o}C. The anisotropy in fracture toughness decreased with increasing temperature. The fracture toughness of CuAl25 IG0 alloy in longitudinal and transversal orientations was relatively high at ambient temperature, on average 92 kJ/m^2, and decreased continuously with increasing temperature to about 6 kJ/m^2 at 350°C. In short transversal orientation the corresponding fracture toughness values were 40 kJ/m^2 and 2,5 kJ/m^2, respectively. A minimum in J_Q was not observed in the temperature range studied. On the other hand, CuCrZr alloy showed only a moderate orientation dependency in initiation fracture toughness. At ambient temperature the fracture toughness of CuCrZr alloy was about 220 kJ/m^2 and decreased first to about 130 kJ/m^2 at 200°C and remained at about 170 kJ/m^2 at the test temperature of 350°C

A marked about 75% decrease in fracture toughness of CuAl25 IG0 alloy due to neutron irradiation to the dose level of 0.3 dpa was observed at temperatures in the range from 22°C to 350°C. Due to irradiation the fracture toughness of CuAl25 IG0 alloy decreased to about 26 kJ/m^2 at ambient temperature and to about 1.3 kJ/m^2 at 350^{o}C. No significant effect of neutron irradiation was observed in the fracture toughness of CuCrZr alloy at or below 200°C. However, a clear decrease in fracture toughness was observed at the irradiation and test temperature of 350°C. Fracture toughness of CuCrZr alloy in the unirradiated condition was about 170 kJ/m^2 which decreased due to neutron irradiation to about 95 kJ/m^2 at 350°C. No changes in fracture mode was observed due to irradiation.

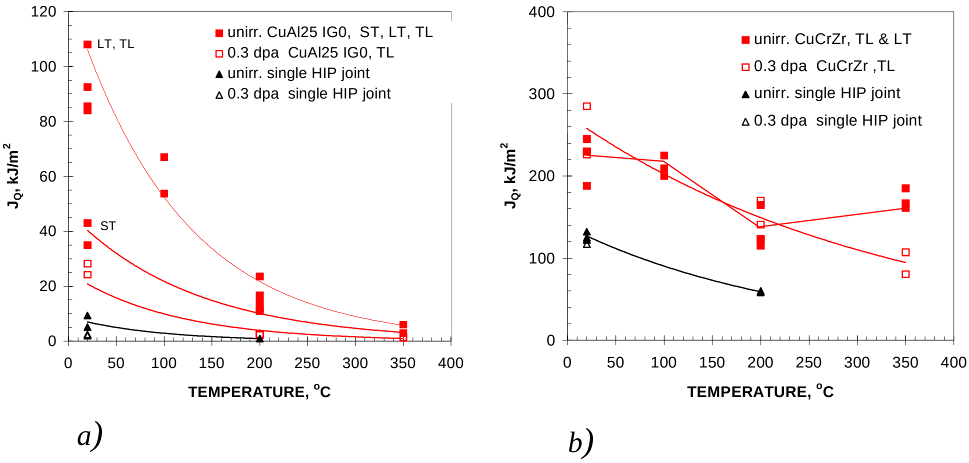


Fig. 3.1. Initiation fracture toughness of a) CuAl25 IG0 alloy and b) CuCrZr alloy together with corresponding HIP joint with austenitic stainless steel 316LN IG0 in unirradiated and neutron irradiated (0.3 dpa) conditions.

These results indicate that fracture toughness of CuCrZr alloy is markedly higher than that of CuAl25 IG0 at the temperature and neutron fluence ranges representative to ITER primary wall and divertor applications. Low fracture toughness have serious limitations on the use of CuAl25 IG0 alloy especially in divertor cassette where copper alloy is used as a structural material in directly water cooled components.

3.1.3 Creep

The limited work hardening ability and the loss of fracture toughness of CuAl25 IG0 alloy at elevated temperatures is commonly related to heterogeneity in alumina particle distribution. However, mechanisms and remedies for temperature dependent fracture behaviour of CuAl25 IG0 alloy are not well understood.

One possible mechanism explaining the observed low fracture toughness values is sensitivity to creep crack growth at elevated temperatures. Creep crack growth was studied by applying standard fracture resistance curve determination procedures with decreasing loading rates. Precracked single edge notched bend specimens SEN(B) were tested under displacement controlled three point bend method with varying loading rates. At 200°C the initiation fracture toughness,

J_Q, decreased from about 10 kJ/m^2 to a value of about 2-3 kJ/m^2 when the load-line displacement rate decreased from $1.5 \cdot 10^{-2}$ mm/min to $8 \cdot 10^{-5}$ mm / min, respectively, see Fig. 3.2. Additionally, creep crack growth tests were performed using similar test arrangement under constant load and constant displacement conditions. Crack propagation was also observed during these constant load and constant displacement tests at 200^{o}C when the corresponding J-integral at the crack tip was less than 1.5 kJm^{-2}.

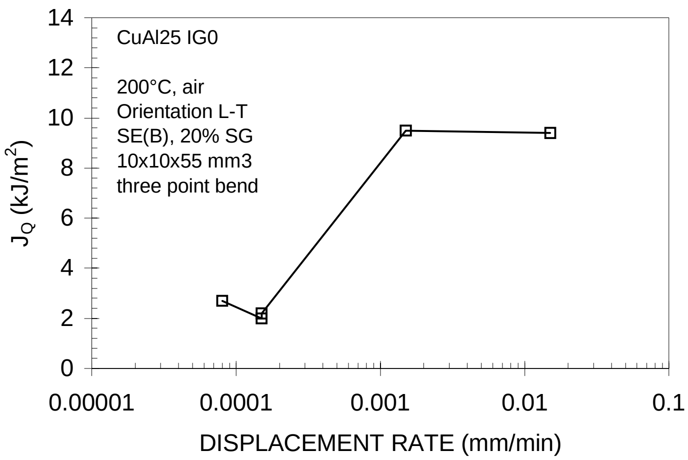


Fig. 3.2. Effect of load line displacement rate on initiation fracture toughness of CuAl25 IG0 alloy at 200^{o}C.

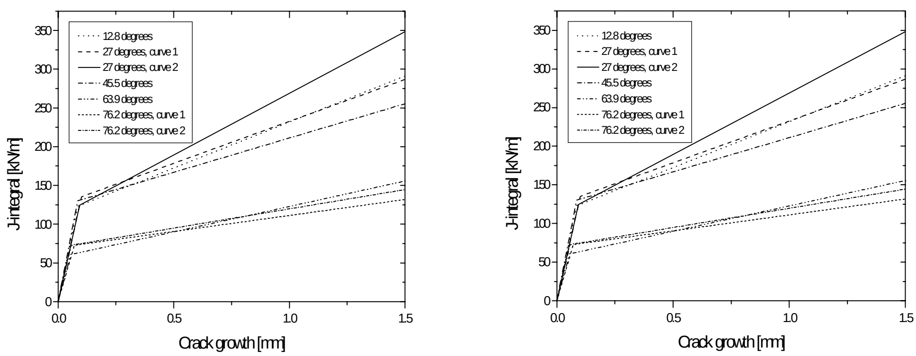
These experimental results indicate that the fracture toughness of CuAl25 IG0 alloy at elevated temperatures, already at 200^{o}C, is dominated by crack growth induced by creep mechanism.

3.1.4 Mixed Mode Loading

The actual loading situation of primary wall modules under ITER operating conditions is complex and mixed mode type in nature. The general understanding behind the mixed-mode fracture behaviour of materials is that mode I fracture toughness is the conservative lower estimate in all loading conditions. The concepts are based on experimental results related to testing of brittle materials. However, recent studies with ductile elastic-plastic metallic materials have indicated that the fracture toughness locus, especially between modes I-II, behaves differently. In these cases mode II fracture toughness has proven to be significantly lower than the mode I counterpart, of the order of several tens of percents at room temperature and inert conditions.

The fracture toughness between modes I and II for martensitic stainless steel F82H and CuAl25 IG0 in the form of fracture resistance curves was studied and a drastic drop in fracture resistance values as the portion of mode II loading increased was observed, Fig. 3.3. In F82H the decrease in fracture toughness was more gradual but in CuAl25 IG0 a more severe drop occurred at certain location of the envelope, approximately at $\psi = 50°$. The differing behaviour of CuAl25 IG0 is to be attributed to an orientation effect in combination to a sharp change in crack nucleation mechanism due to extensive shearing.

In complex loading situations the mixed-mode fracture behaviour of metallic materials can lead to lower values of fracture toughness which is contrary to the general understanding of mixed-mode fracture. Quantitatively, differences are in the range of 40-50% and, therefore, can not be neglected. Differences in fracture toughness are a consequence of alterations in micromechanisms of fracture between modes I and II. The transitions can be explained through changes in stress and strain states, which cause a transition in the fracture mechanisms as a result of a competition between typical mode I and II type of fractures.



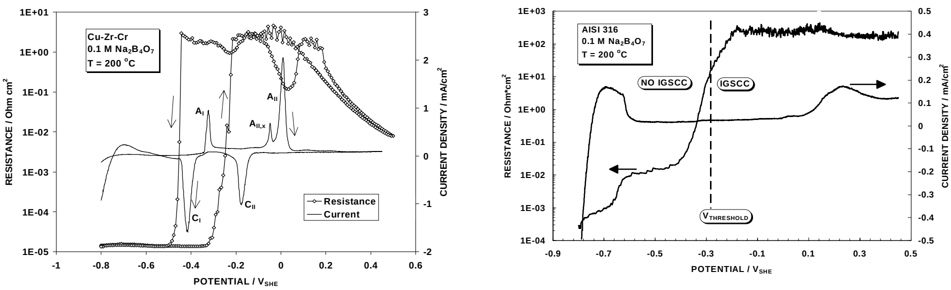
a) *b)*

Fig. 3.3. Mixed mode fracture resistance curves for a) martensitic stainless steel F82H and b) dispersion strengthened CuAl25 IG0 alloy.

3.1.5 Corrosion

The ITER blanket and divertor components are water cooled during the plasma operation. The long term corrosion properties as well as the stress corrosion cracking susceptibility of the materials in a given environment are controlled by the type of surface films present on the material. The surface films on precipitation hardened copper alloy CuZrCr and austenitic stainless steel 316LN in a borate buffer solution (0.1M $Na_2B_4O_7$, pH_{200C} = 8.0) were investigated *in situ* at 200°C using electrochemical impedance spectroscopy (EIS), capacitance measurements, potentiodynamic polarisation and contact electric resistance (CER) techniques as well as *ex situ* using X-ray photoelectron spectroscopy (ESCA) and secondary ion mass spectrometry (SIMS) techniques.

Fig. 3.4 shows the polarisation and the potential-resistance curves of CuCrZr alloy and 316 LN. In CuCrZr alloy the anodic current peaks A_I and A_{II} are interpreted to be due to the oxidation of Cu to Cu_2O and Cu^+ and further oxidation of Cu_2O to $Cu(OH)_2$ and CuO_2^{2-}, respectively. The peak A_I coincides well with observed large increase in the resistance in the positive going scan. The following decrease in the resistance may be correlated with the peak $A_{II,x}$ and the following increase with the peak A_{II}. The additional peak $A_{II,x}$ is possibly due to the oxidation of Cu_2O to CuO. The cathodic current peaks C_I and C_{II} are proposed to be caused by the reduction of Cu_2O to Cu and of $CuO/Cu(OH)_2$ to Cu_2O, respectively. In the negative going scans the small decrease in the resistance observed at about –0.2V correlates with the reduction peak C_{II}. When the voltage is further lowered, the resistance decreases to the level indicating oxide free surface right after the reduction peak C_I.



a) *b)*

Fig. 3.4. Comparison of the potentiodynamic polarisation curve and the potential - resistance curve of a) CuZrCr and b) 316LN.

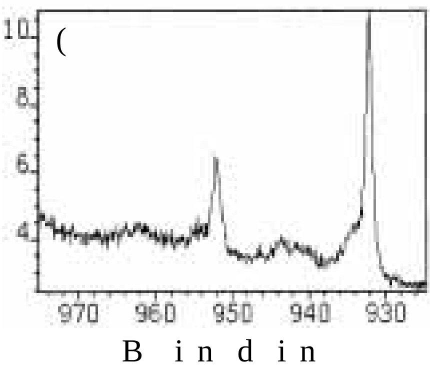
In 316LN at low potentials the surface film resistance measurement indicates presence of a film of a rather low ohmic resistance. In this potential range both iron oxide (Fe_3O_4) and chromium oxide (Cr_2O_3) are thermodynamically stable, although it is clear that the surface film on AISI 316LN is most probably a spinel oxide of type MN_2O_4, where M and N can be

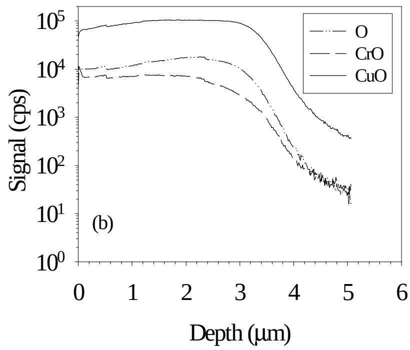
Fe, Cr or Ni. When the potential exceeded about 0.1 V_{RHE}, the polarisation current density, after going through a maximum, decreased to a very low (but positive) level indicating passivation of the surface. At the same potential the surface film resistance increased to a level of about 1 Ω, which also indicates passivation of the surface. The potential at which both techniques indicate passivation of the surface to occur is close to the potential where nickel oxide NiO is expected to become thermodynamically stable.

In the ESCA spectrum in Fig. 3.5a there are two 2p peaks, 2p ½ (952eV) and 2p 3/2 (932 eV), respectively. Copper is present either as metallic or as Cu^+ in the case of final polarisation voltage –0.15V. Zr was not detected on the surface and Cr was enriched when compared to the bulk amount. From Fig. 3.5b it can be observed that the thickness of the oxide layer is about 3.5 μm.

According to the ESCA and SIMS analyses copper forms a stable oxide Cu_2O at polarisation voltage –0.15V. Therefore it can be assumed that the processes occurring within the film are the outwards movement of Cu^+ and the inwards movement of vacancies V'_{Cu}. The molar volume of Cu_2O is considerably larger than that of metallic Cu indicating that during the growth of the film a significant amount of Cu^+ has to migrate through the film and dissolve into the electrolyte.

In the case of final potential +0.45V copper forms CuO oxide. Both Cr and Zr are depleted in the oxide layer at final potential +0.45V according to the SIMS analyses.





a) *b)*

Fig. 3.5. ESCA Cu 2p high resolution spectrum (a) and SIMS depth profiles of O, CrO and CuO (b) for film formed at –0.15V.

The behaviour of the CuZrCr alloy resembles closely that of pure copper. The bulk of the film growing on the alloy is a very good electronic and ionic conductor. Both the oxides Cu_2O and CuO present at different polarisation voltages had a thin p-type semiconducting layer located at the interface of the film and electrolyte solution. The semiconducting layer was shown to have a thickness of 10^{-8} m while the thickness of the film was of the order of 10^{-6} m.

Finally it can be concluded that the in-situ methods give information merely on the electrical properties of the surface films whereas the ex-situ methods provide additional information on the chemical structure and thickness of the films. The knowledge on the properties of the surface films can be used in deeper understanding of the corrosion and environmental induced cracking phenomena.

3.2 Cu/SS Joining Technology and Characterisation

The present design of ITER primary wall modules is based on multimaterial concept with stainless steel as a structural material, copper alloy as a heat sink and beryllium as a plasma facing material. The candidate joining method for manufacturing these multimaterial blanket modules is hot isostatic pressing (HIP). Other joining methods for stainless steel copper components considered are friction welding, explosion welding (EXW), fusion welding and rheocast brazing. Assessment and validation of joining methods and evaluation of joint integrity of ITER primary wall module and other in-vessel components and characterisation of the joint properties in non- and post-irradiated conditions is essential for reliable reactor design. In this work HIP and EXW methods were studied and corresponding joint properties were characterised and compared.

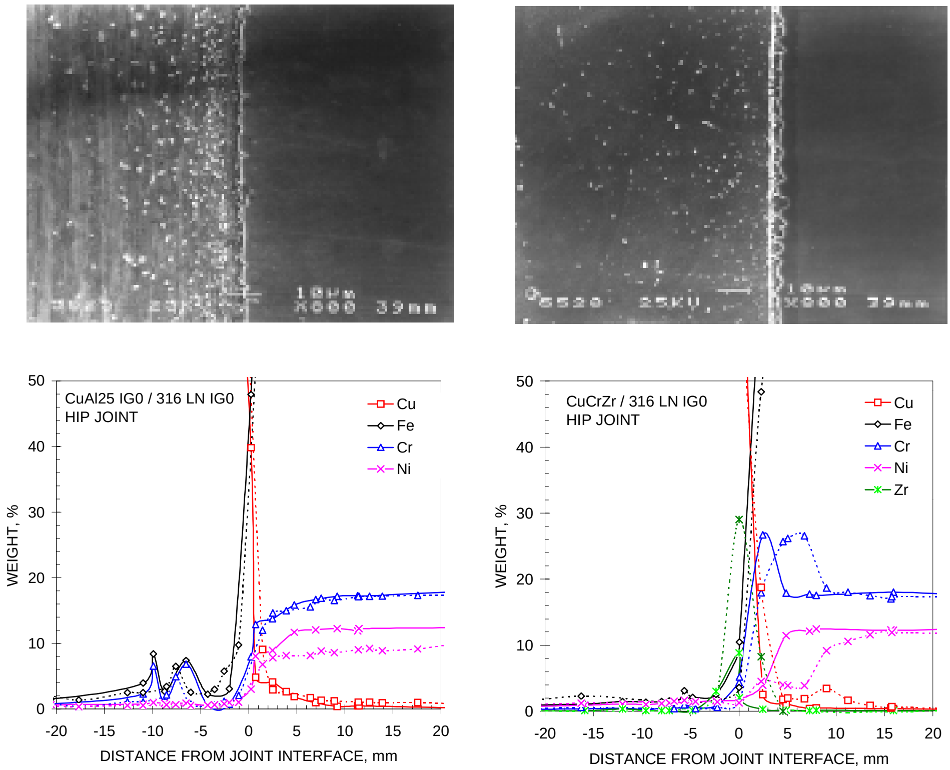
3.2.1 Metallurgy of Joints

HIP and EXW are both solid state bonding methods although HIP can also be applied for powder compacting. However, due to differences in bonding mechanisms also the subsequent metallurgy and mechanical properties of constituent metals and their joints will be different after bonding treatment. HIP bonding is based on diffusion of alloying elements across the joint interface at elevated temperatures which is further enhanced by applying isostatic pressure. EXW bonding is based on high impact pressure induced by explosives at ambient temperature and therefore there is practically no diffusion of alloying elements across the joint interface.

The microstructure of copper alloys and stainless steel after typical HIP cycle are close to those of solution annealed structures because the applied temperature ranges during HIP bonding and solution anneal heat treatments of both copper alloys and stainless steel are almost similar. However, diffusion of elements across the joint interface may induced extra phase transformations and precipitation reactions in the area close to joint interface. During the HIP thermal cycle alloying elements of CuCrZr alloy like zirconium readily reacts with nitrogen and carbon in stainless steel and precipitates as zirconium nitrides at the joint interface. Diffusion of elements like nickel and carbon decreases the stability of austenite and induces ferrite phase transformation in stainless steel close to joint interface. The chromium dissolved from precipitates of copper alloy and from austenite diffuses and enriches in ferrite layer. In the case of CuAl25 alloy the diffusion of elements results in precipitation of chromium and iron rich phase in copper alloy but no ferrite phase formed in the stainless steel side of the joint.

A typical wavy like, solid interface with front and rear vortices is formed due to explosion welding of stainless steel to copper alloy. Vortices are typical for explosion welded interfaces and they are composed of a mixture of the component metals. Cavities and pores are commonly observed at the edges of the explosion welded plate associated with vortices. According to simulations the velocity of the flyer plate and also the impact pressure are lower close to free ends of the plate compared with the middle section of the plate. These factors can deteriorate the weldability and should be taken in to account in component manufacturing. On the other hand, the shock waves induced by explosives results in macroscopic plastic deformation close to the joint interface and in generation of high dislocation density while macroscopic strains are negligible further away from the joint interface.

The manufacturing of prototype primary wall modules and divertor components will need several joining operations e.g. multiple HIP and/or brazing thermal cycles and the various joints must withstand all these temperature cycles. Multiple HIP thermal cycles was shown to further enhance diffusion of alloying elements and to increase thickness of diffusion layers and amounts of precipitates, Fig. 3.6. After high temperature post weld heat treatment simulating HIP thermal cycle the microstructure of explosion welded copper to stainless steel joint is quite similar to corresponding HIP joint. In the case of explosion welded CuCrZr alloy to stainless steel joint zirconium nitrides and ferrite layer were observed at the joint interface. However, due to gradient in plastic deformation and internal dislocation structure induced by shock waves, the recrystallisation and extensive grain growth in copper alloy near the joint interface is observed after post weld heat treatment.



a) *b)*

Fig. 3.6. SEM micrographs of triple HIP joints and corresponding EDX analysis across the HIP joint interface between austenitic stainless steel 316 LN IG0 and copper alloys a) CuAl25 IG0 and b) CuCrZr. Solid and broken lines indicate EDX analysis after single HIP and triple HIP thermal cycles, respectively.

3.2.2 Mechanical Properties of Copper to Stainless Steel Joints

Bimetallic joints may exhibit substantial heterogeneity with respect to strength and deformation properties of constituent metals. During mechanical testing this heterogeneity affects the stress and strain distribution close to joint interface and consequently have a direct influence on interpretation of e.g. tensile, fatigue, fracture toughness test results which are normally based on procedures developed for homogeneous materials. This obvious difficulty is clearly demonstrated in behaviour of explosion welded stainless steel to CuCrZr alloy joints when mechanically tested using different specimen types and strain rates at different temperatures, Fig. 3.7. Also the strength mismatch, which varies due to testing temperature or post weld heat treatments, between the constituent metals was shown to affect the observed fracture behaviour. Tensile and cyclic fatigue tests with cross weld specimens at room temperature and 300°C resulted in ductile failure of copper alloy. However, at elevated temperature with tensile hold at peak strain during cyclic fatigue test or in a constant load creep test, the resulting failure mode changed from ductile failure of copper alloy to ductile interface failure. Also in fracture toughness tests the crack propagation followed the copper stainless steel interface at elevated temperatures. The work on bimetallic materials will continue by validating suitable fracture toughness test method for characterising joint properties of industrially manufactured primary wall modules.

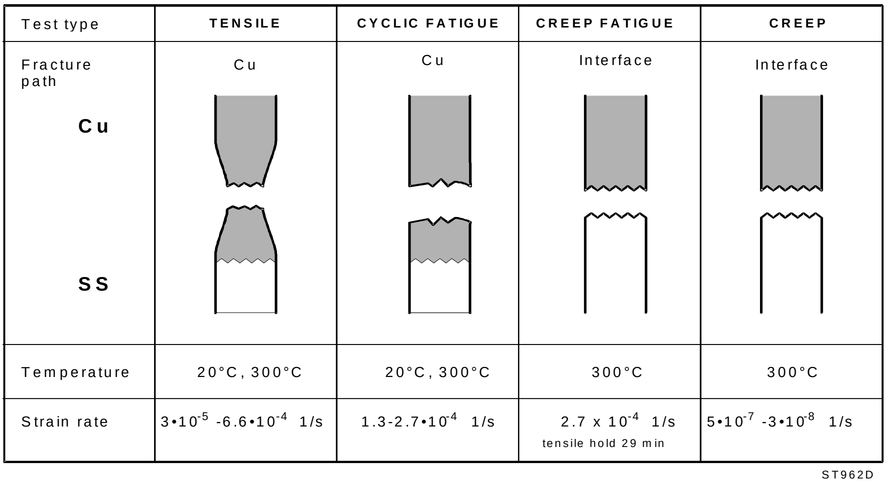


Fig. 3.7. Typical fracture behaviour of explosion welded cross weld tensile test specimens.

One of the most simple screening test method to compare bimetallic joint properties is shear strength testing. The shear strength clearly shows the effect of different joining methods on the strength of copper alloys and their joints, Fig. 3.8. When the shear strength of CuCrZr alloy in prime aged condition, e.g. water quenched after solution annealing, is compared with the shear strength after HIP thermal cycle a clear about 20-30% reduction is observed. However, the shear strength of CuAl25 IG0 alloy experienced only a moderate reduction due to HIP thermal cycle. This difference in behaviour can be understood by different strengthening mechanisms of the two copper alloys. The strength of CuCrZr alloy is therefore more sensitive to temperature cycles induced by joining methods than the strength of IG0 alloy.

The shear strength of the copper to stainless steel joint was also higher for CuAl25 IG0 alloy than for CuCrZr alloy although the difference in shear strengths was not as marked as for copper base alloys. EXW copper stainless steel joints had higher shear strength values when compared with HIP joints. After EXW the shear strength of CuCrZr alloy was comparable to shear strength of prime aged alloy.

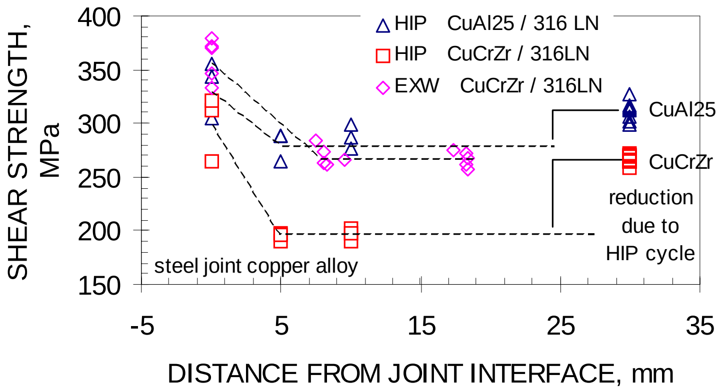


Fig. 3.8. Shear strength of CuAl25 and CuCrZr alloys and their EXW and HIP joints between 316LN stainless steel. HIP cycle was performed at 960°C for 3 hours at 120 MPa followed by slow cooling and separate precipitation anneal for CuCrZr alloy joint at 460°C for 2 hours

3.2.3 Fracture Toughness of Joints

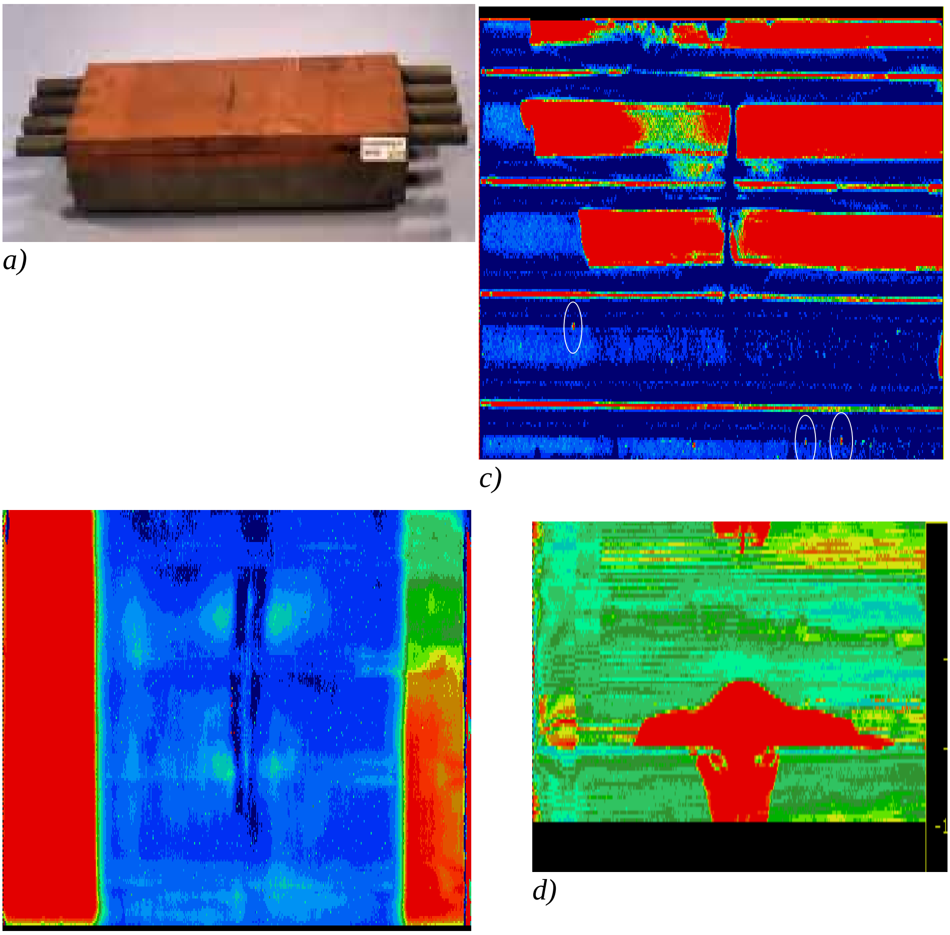
A significant reduction in fracture toughness of both copper alloy HIP joints were observed at temperatures in the range 22°C to 350°C compared to the corresponding base copper alloys, Fig. 3.1. The fracture toughness of CuCrZr alloy HIP joints was higher than that of CuAl25 IG0 alloy HIP joints. The HIP joints of CuAl25 IG0 showed very low fracture toughness of about 7 kJ/m^2 already at ambient temperature which further decreased to about 3 kJ/m^2 at 200°C. In the case of CuCrZr the fracture toughness at ambient temperature and at 200°C was 150 kJ/m^2 and 60 kJ/m^2, respectively. Examination of fracture surfaces after fracture toughness tests showed that crack propagation did not occur at the joint interface but close to the interface within copper alloy side of the joint interface. In CuAl25 IG0 alloy crack propagated parallel to joint interface and in CuCrZr alloy crack propagation seemed to deviate from the joint interface. It should be pointed out that the HIP joint geometry was plate to plate type of joint which leads to fracture orientation of S-T or S-L in reference to copper alloy part of the joint. Therefore, the properties of the HIP joints should be compared with short transverse properties of corresponding copper alloys which was shown to be particularly weak in CuAl25 IG0 alloy.

Neutron irradiation to dose level of 0.3 dpa reduced the fracture toughness of both copper alloys to stainless steel HIP joints when compared to fracture toughness of unirradiated HIP joints at 22^{o}C. After neutron irradiation the CuAl25 IG0 HIP joint specimens showed a `brittle-like´ behaviour with sudden drop in load at low displacement values compared to unirradiated HIP joint specimens. However, the fracture mode was ductile and the fracture surface morphology was similar to those of unirradiated HIP joints or base alloy in S-T orientation. On the other hand the CuCrZr alloy HIP joint specimens showed extensive plasticity similar to unirradiated HIP joints or base alloy specimens. The preliminary examination of the irradiated fracture surfaces of HIP joint specimens indicated that ductile fracture occurred within copper alloy side of the joint.

3.2.4 Non Destructive Examination of Joints

An essential part of quality control of manufacturing stages and reliable operation of the fusion reactor requires high precious non destructive examination methods capable to find discontinuities in dissimilar metal interfaces. Non-destructive examination of multimetallic plate and tube interfaces produced by EXW, HIP and rheocast methods were studied by applying various ultrasonic techniques e.g. reflection type C-mode scanning acoustic microscope, internal rotating inspection system and eddy current techniques. Ultrasonic examination of EXW interfaces showed to be sensitive to scanning angle and direction relative to direction of explosion front propagation. This was attributed to microstructural features e.g. formation of vortices associated with wave like interface which is typical to explosion welding.

Ultrasonic and eddy current examination for plane and tube interfaces of small scale primary wall mock ups was also developed. Small scale primary wall mock-ups were successfully examined before and after high heat flux tests. Fig. 3.9 shows one of the primary wall mock ups made of dispersion strengthened CuAl25 IG0 alloy by HIP method and corresponding ultrasonic C-scan images taken after high heat flux test performed in electron beam test facility FE200 at Le Creusot. The mock up was cycled at 5 MWm^{-2} with 15 s on / 15 s off cycle frequency and the test was interrupted when the surface



a)

c)

d)

b)

Fig. 3.9. a) ITER primary wall mock up made of CuAl25 IG0 alloy by HIP method after high heat flux testing at 5 MWm^{-2}, b) ultrasonic C-scan image on heated copper surface showing strong change in attenuation properties of leaky Rayleigh waves, c) ultrasonic C-scan image on copper to copper interface showing large areas of interface separation and d) ultrasonic C-scan image on stainless steel tube No. 3 (tube length in horizontal and tube circumference in vertical direction) to copper interface showing large area of interface separation.

temperature of copper reached 900°C after 962 cycles. During the test the mock up was continuously cooled by cooling water at 140°C and 2.6 MPa pressure. The ultrasonic examination showed large change in attenuation properties of leaky Rayleigh waves on the heated copper surface which was related to extensive crack formation on the copper surface, crack depth was less than 100μm. Also the copper to copper interface and stainless steel tube to copper interfaces were separated from large areas but no defects were found on copper to stainless steel interface. It is noteworthy that simultaneously tested precipitation hardened CuCrZr mock up was successfully tested without any failures at 5 MWm^{-2} for 1000 cycles. Also both types of mock ups were tested at 0.75 MWm^{-2} for 13000 cycles without no failures. It should be noted that the design heat load for the ITER primary wall modules is 0.5 MWm^{-2}.

Ultrasonic and eddy current techniques were successfully applied for non destructive examination of both plane and tube interfaces of small scale primary wall mock ups. Resolution for detection of discontinuities depends in principle on materials, particular geometry and ultrasonic frequency used for examination. In the case of CuAl25 IG0 alloy ultrasonic frequencies of 25 MHz were used whereas due to strong attenuation frequencies of only 10 MHz were used in the case CuCrZr alloy. However, based on detection of different calibration defects the resolution for detecting interface discontinuities was less than 1 mm on all plane and tube interfaces in small scale primary wall mock ups made of both candidate copper alloys.

3.3 Behaviour of Hydrogen Isotopes in First Wall Materials

The aim of this project was to understand and control hydrogen retention in first wall materials. The project comprised studies of physical interactions between plasma and first wall materials, namely the retention of hydrogen isotopes (H, D) in metals and diamond-like carbon (DLC) films. The research included investigations of the properties of physical vapour deposited (PVD) DLC films on Si and stainless steel substrates and the migration of hydrogen isotopes in implanted and co-deposited DLC films.

Study of hydrogen migration in Ta, Ni, W and stainless steel AISI 316L was carried out. The effect of ion induced damage and annealings on the H/He retention was investigated. Stopping of 5-100 keV He-ions in Ta, Ni, W, AISI 316L, Cu, Ni, Mo and Cr was carried out to understand the mechanisms of damage production under ion bombardment.

An arc-discharge coating process developed by the DIARC-Technology Inc. has been used to produce amorphous hydrogen free DLC films. The deposition of the DLC films takes place inside a vacuum chamber at room temperature. In the process an arc-discharge is generated between a carbon cathode and an anode. A dense amorphous diamond-like structure is formed when a 40 - 60 eV C^+ plasma beam, guided using magnetic fields, hits the surface of a substrate. Before the deposition stage a sputter cleaning is performed using a broad-beam Ar ion source in order to remove an oxide layer and organic impurities from the surface of the substrate. In some cases a tungsten interlayer is deposited in order to improve the adhesion of the DLC film.

DIARC process has also been used to produce hydrogen, deuterium and methane co-deposited carbon films. The gas flow through the vacuum chamber is controlled during the growth of the film and the deposition pressure is measured using a cold cathode gauge. The manufacturing of a co-deposited carbon layer takes place at room temperature in a partial pressure of a selected gas.

The presence of heavy impurities in the samples was investigated by the particle induced X-ray emission (PIXE) technique. The depth distribution of the impurities was obtained by secondary ion mass spectrometry (SIMS). Measurements showed the presence of V, Fe and Ni impurities. Further analyses showed that these impurities originate from the graphite used as a cathode in the film preparation. Tungsten observed at the interface between the DLC films and Si substrates, was deposited during the etching process of the substrate surface. The W concentration in these layers was obtained to be about 0.7 at.%. The total amount of the V, Fe and Ni impurities was obtained to be about 0.12 at.%. The mass density of the films was determined by Rutherford backscattering spectrometry (RBS) and SIMS, and was obtained to be 2.6 ± 0.1 g/cm^3. The amount of sp^3 (diamond) –bonds was measured with X-ray photoelectron spectroscopy (XPS) and it is typically between 40 and 60%.

Two sets of samples were prepared for H migration studies. In the first set samples were implanted with 30-keV $^1H^+$ ions to a dose of 1 x 10^{16} ions cm^{-2}. In the second set both 35-keV $^4He^+$ and 30-keV $^1H^+$ ions were implanted to doses of 1 x 10^{16} ions cm^{-2}. The isochronal annealings (40 min) were made in a quartz-tube furnace (pressure below 0.05 mPa) at temperatures between 100 ^oC and 1100 ^oC. For the depth profiling of H atoms the nuclear resonance broadening (NRB) technique with the 6.39-MeV resonance of the $^1H(^{15}N,\alpha\gamma)^{12}C$ reaction was used (see Fig. 3.10).

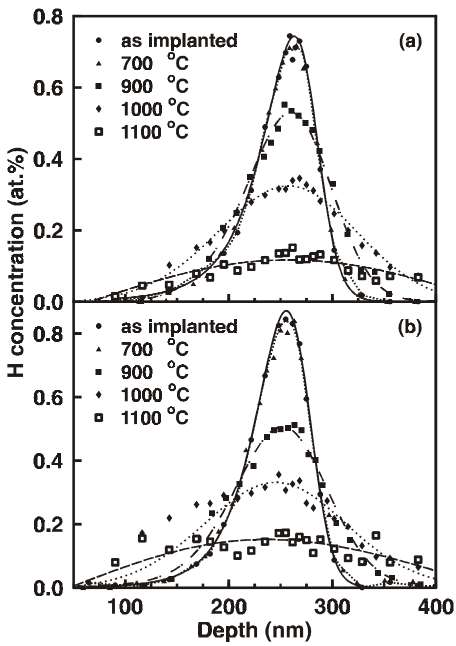


Fig. 3.10. Hydrogen concentration distribution observed in NRB measurements in H^+ (a) and He^+ (b) implanted samples. Distributions were observed after the implantation and after annealings at different temperatures. Solid line is the Pearson IV fit of the implanted depth profile. Dashed lines show the depth profiles calculated using a diffusion model developed.

The concentration profiles of implanted He and H atoms were measured by the elastic-recoil-detection-analysis technique (ERDA). It was obtained that no significant loss of He took place in the annealings. Results of NRB measurement showed that background hydrogen concentration was 0.07 at.%.

Migration of implanted hydrogen is described well with a concentration independent diffusion equation. The diffusion coefficients for hydrogen in the temperature range 700 – 1100 ^{o}C were extracted from NRB and SIMS measurements (see Fig. 3.10). The diffusion coefficients exhibit a good Arrhenius behaviour with an activation energy of 2.0 ±0.1 eV.

To study diffusion of deuterium in DLC films, a set of samples was implanted by 54-keV D_2^+ ions to a dose of 1 x 10^{16} ions cm^{-2}. The depth profiles of D ions were measured with SIMS. The measured D concentration profiles were fitted with a concentration-dependent diffusion model, assuming that D exists as immobile pairs and diffusing atoms (see Fig. 3.11). The results show that the concentration of D clusters relative to the total D concentration increases when the total D concentration decreases, leading to a concentration dependent diffusion. The diffusion coefficient for atomic deuterium exhibits a good Arrhenius behaviour with an activation energy of 2.9 ± 0.1 eV. A decreasing solid solubility of D in DLC films with increasing temperature was observed.

In the study on H diffusion no concentration-dependent process was observed due to the initial hydrogen background of about 0.07 at.% in these films, whereas in the case of D the fits were made to the low concentration regime. This explains the difference in activation energies (E_a = 2.0 ± 0.1 eV for H). Therefore, the diffusion coefficients for hydrogen and deuterium can not be compared directly with each other. However, by employing the matrix method to fit D profiles and choosing the lowest concentration limit of 0.07 at.% one gets an activation energy of 2.0 ± 0.1 eV matching the value for H. The ratio of the pre-exponential factors for H and D diffusion is 1.3, which is explained by the isotope effect.

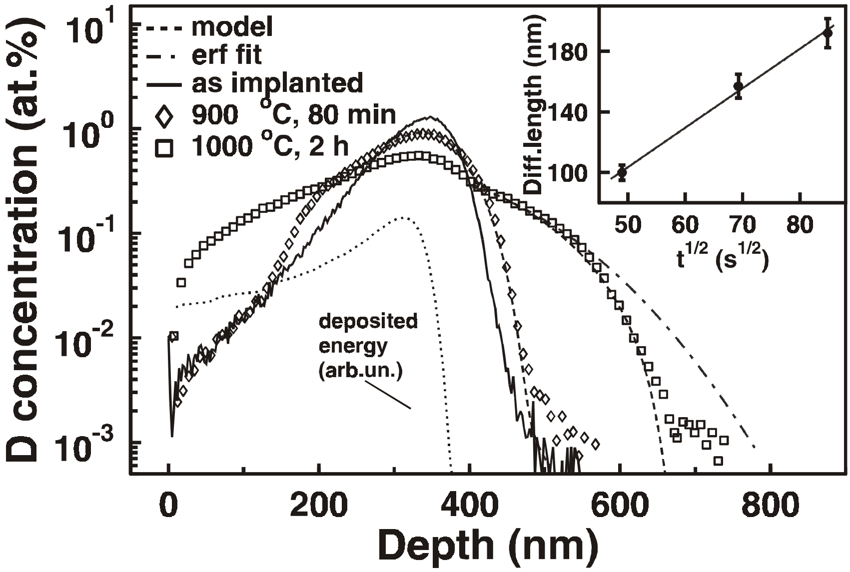


Fig. 3.11. SIMS depth profiles for deuterium obtained after implantation and after annealing at different temperatures with numerical fits by the diffusion model (dashed line) and error function (dot-dashed line). Dot line is the deposited energy calculated by SRIM-96. The inset shows D diffusion length vs. square root of the annealing time for 1000 °C 40 , 80 and 120 min annealings. Solid line is the linear fit to the experimental data.

In addition to ion-implanted samples hydrogen, deuterium and methane co-deposited samples for migration studies were produced. Hydrogen concentration in different depositions was varied by changing the pressure of hydrogen atmosphere between 0.06 and 0.6 mPa. H concentrations in the samples deposited at different pressures were relatively constant throughout the film. Hydrogen content is proportional to the square root of the deposition pressure up to 0.6 mPa. Annealing experiments showed a decrease of the hydrogen concentration with increasing temperature, H release and migration to the interface. It was observed that the release temperature varied between 950 °C and 1070 °C depending on the H concentration.

Outdiffusion of deuterium was studied in deuterium co-deposited films. The films having thickness of 700 nm were deposited on silicon at a deposition pressure of 0.6 mPa. The concentration profile was measured by ERDA and SIMS techniques. The D concentration was found to be uniform throughout the film and the amount of D was obtained to be of 7.0 ± 0.4 at.%.

The annealing of these films between temperatures 700 ^{o}C and 1000 ^{o}C, showed two clear effects to the initial uniform D concentration profiles. The first effect is that the solid solubility limit of D in DLC decreases with increasing temperature from about 7% at 800 ^{o}C to about 2.5% at 1000 ^{o}C (see Fig. 3.12). The second observed effect was the diffusion and ejection of deuterium at the surface. The model developed to calculate D diffusion in implanted samples was used to numerically simulate the experimental profiles. The agreement between the experimental profiles and the theoretical fits is good (see Fig. 3.12). The analyses are under progress and the activation energy for diffusion will be obtained.

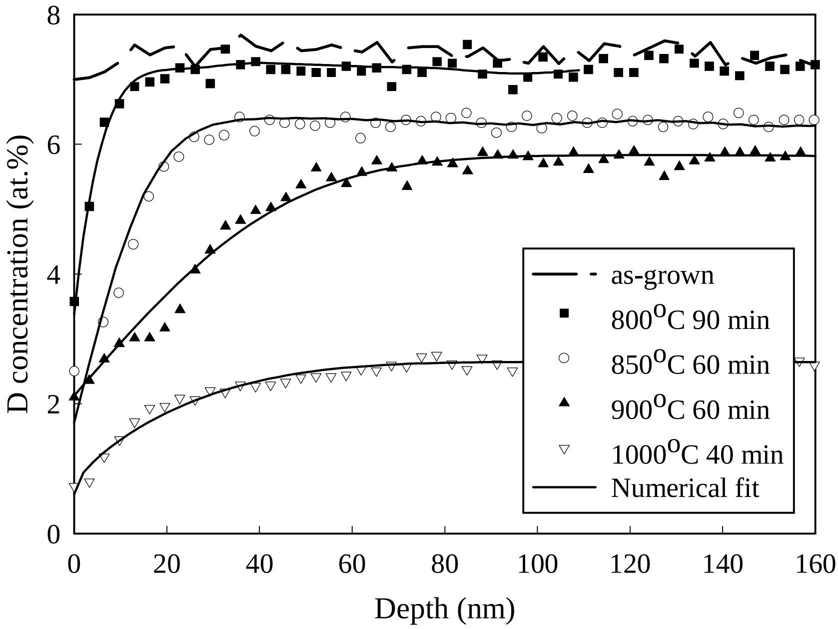


Fig. 3.12. SIMS depth profiles for deuterium obtained after annealing at different temperatures with numerical fits by the diffusion model (solid line).

3.4 Fusion Neutronics

The main objective of the work described here was to check the adequacy of the shielding provided by radio-frequency supplementary heating antenna assemblies in the equatorial ports of ITER. The heating systems considered were Ion Cyclotron Resonance Frequency (ICRF), Electron Cyclotron Resonance Frequency (ECRF) and Lower Hybrid (LH) heaters.

The calculations were performed using MCNP4B, the most recent version of the MCNP program, and a point wise cross-section library based on FENDL-1. Only the contribution to the flux coming through the equatorial port itself was calculated, neutron leakage through the bulk shielding and through other ports was suppressed.

The goal was to find shielding arrangements that would satisfy certain criteria. These criteria were:

- To limit the dose rate in biological tissue behind each port near the cryostat 10^6 s after shutdown to acceptable levels, the fast neutron flux (above 0.1 MeV) in the zone behind the port closure plate must not exceed 10^7 n/cm^2 s.
- The nuclear heating (neutron+gamma heating) in the cryogenic systems (mainly the toroidal and poloidal field coils and the intercoil structure) must be less than 500 W for one port.
- The nuclear heating density must be below 2 mW/cm^3 in the TF coil case (which was taken to include the intercoil structure as well) and 1 mW/cm^3 in the winding pack.

The second and third of these criteria were rather easily met, but the first criterion turned out to be much tougher. For none of the three supplementary heating systems considered was the original design of the antenna array adequate to meet the first criterion. However, the fast flux beyond the port closure plate can be reduced to acceptable levels with additional shielding. Certainly there is more than enough space inside the port for the required shielding, though the weight of the shielding may be a problem.

For the ICRF antenna array, the measures needed to ensure adequate shielding include curving the coaxial cables, to avoid straight streaming paths through the vacuum between the inner and outer conductors. The all-metal supports for the inner conductor of the coaxial cables should be as massive as possible to further decrease the streaming. The front part of the metal/water blocks serving as a substitute for the shield/blanket in the equatorial port should preferably have a minimum thickness of at least 1 m, with an additional 40 cm of shielding between the tails of these blocks. The box surrounding the antenna array must be designed so that straight streaming paths from the plasma to the closure plate are avoided. Moreover, the gap between this box and the port walls should be kept as narrow as feasible. A gap of 2 cm is acceptable, but much more cannot be accepted.

Even for the ECRF heaters, the original design does not provide sufficient shielding. Additional shielding material is needed, and moreover a second dogleg in the waveguides is required.

The LH heaters are especially problematic from a shielding viewpoint, with their wide waveguides providing opportunities for streaming. However, it is possible to obtain adequate shielding with a design involving split and curved waveguides, surrounded by several shielding layers of steel and water, with a total thickness of 45 cm. The thickness of the port walls should also be increased, since otherwise the rather open LH array may have problems with neutron leakage through these walls, leading to high flux levels beyond adjacent ports.

Since the weight of thick steel/water shielding causes problems in the handling of the heater arrays, there is an incentive to consider other materials. For this reason the possibility of using polypropylene instead was studied for the LH case. (Polypropylene was chosen in preference to the more usual polyethylene due to its somewhat better ability to stand high temperatures.) It turned out that, for a given material thickness, polypropylene is almost as effective a shielding material in ITER as a mixture of 75 % steel and 25 % water. Its lower density makes it much more effective in cases where weight rather than space is the limiting factor. Unfortunately, it seems that even polypropylene cannot really stand the temperatures prevailing in the equatorial ports of ITER. Thus, although the idea of using light materials with a high hydrogen content is attractive from a weight saving viewpoint, it would be necessary to find a material that can stand substantially higher temperatures than polypropylene.

3.5 Development of ITER Superconductors

3.5.1 Development of Superconducting Niobium-Titanium Wires for ITER

During this project a superconductor billet was designed and manufactured to the final diameter following the wire specification for ITER poloidal fiels coils. The main objectives of this work were to test whether this specification would be practicable, to get a good overall conception of the reliability of manufacturing this type of demanding superconducting NbTi wire, and also to review the principal constituents forming the price of the final product.

Starting with a production-size billet, a NbTi strand of what is needed for the poloidal field coils of ITER, has been successfully manufactured, see Fig.

3.13. This strand is very near the strand needed for the dipoles of the CERN Large Hadron Collider (LHC). The addition of a CuNi layer of 10 µm within the outer copper shell, to control AC losses, has not resulted in any degradation of the strand properties.

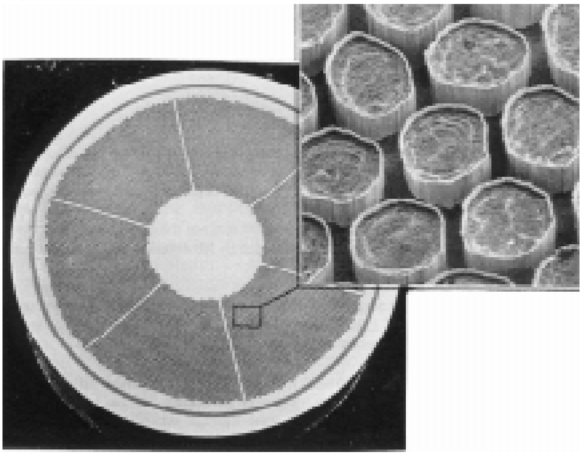


Fig. 3.13. Cross-section of the ITER-type NbTi superconductor developed at Outokumpu Superconductors Ltd.

The critical current density, the RRR and the effective filament diameter are well within the specification. Mechanically the programme was success. After small adjustments the processing was finalized without any breaks and the unit length was 16 700 m, which proves that the process is applicable in industrial scale, too. It was anticipated that the differences between ITER poloidal field coil and CERN LHC-wires could slightly increase the price of the material.

3.5.2 Development of Superconducting Niobium-Tin Wires for ITER

The main objective of this project was to develop, manufacture and test Nb_3Sn superconducting wire as a final goal to meet the ITER HP II specification. The main product of Outokumpu Superconductors Oy is NbTi superconducting wire and there has been no earlier experience of Nb_3Sn wire production. The development of Nb_3Sn started from the manufacture of the matrix material and continued with laboratory-scale wire production trials and design optimization for ITER HP II wire. About 25 kg of ITER HP II wire was set for the deliverable of the project.

The first set of Nb_3Sn superconducting wire was successfully produced. The production process was demonstrated to work and valuable information about the workability and achievable properties of Nb_3Sn wire were obtained. A small quantity of Nb_3Sn wire designed to meet the ITER HP II specification is at present in production.

4 Remote Handling and Viewing

4.1 In-Vessel Viewing System - IVVS

4.1.1 Introduction

Frequent inspections of the interior of the ITER Tokamak vacuum vessel will be required to check for damage caused by plasma operations and to plan maintenance interventions. The environmental conditions in the vessel between plasma pulses are extremely severe - ultra high vacuum (10^{-7} Pa), radiation (about $3x10^{4}$ Gy/h), temperature (about 200°C) and magnetic field (about 5.7 T). The In-Vessel Viewing System (IVVS) originally proposed by the JET (Joint European Torus) Team applies line scanning technology where linear arrays of optical fibres send images from a distance to sensors (CCD cameras). The image selection and focusing is done by off-line computer analysis. This system has few movable parts and allows the inspection of the entire vessel in a short time (viewing time about 6 min.).

The development work has been distributed over the years 1996-98. In 1996 the system level specifications were created and a system level design completed. The feasibility of the viewing concept was demonstrated with a proof-of-principle model. In 1997 separate prototypes of the illumination and imaging optics, and of the view probe and scan mechanism were built and tested. In 1998 these prototypes together with a computer system for probe control and image formation were integrated into a full scale system demonstrating the end to end imaging capability of the IVVS.

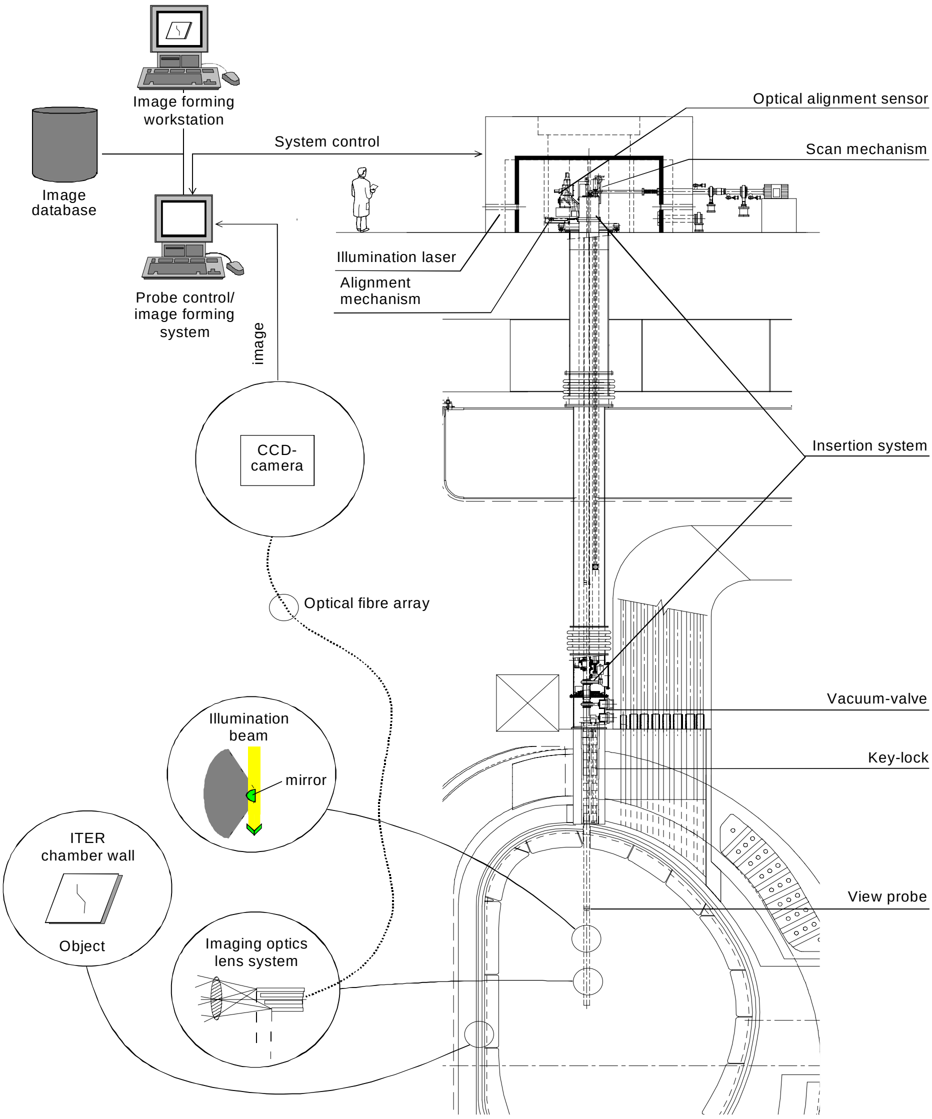


Fig. 4.1. The ITER IVVS system.

4.1.2 IVVS Design Work

The IVVS consists of 10 identical units, installed on top of the ITER machine at the increments of 36°. The probes are inserted into the vessel through the main vertical ports using a long bellows as the vacuum barrier and through key-lock mechanisms which provide neutron shielding during plasma operation. A single probe and its operation is schematically shown in Fig. 4.1.

The illumination is provided by 100 Hz pulsed power laser beams, which pass through the probes and are diffused by mirrors on the viewing plane. The optical fibres are coupled in the lower end to the imaging optics (lens packets) and in the upper end to CCD camera chips. Up to 10 linear arrays of 1000 coherent optical fibre are positioned vertically, one next to the other, in focal planes of each lens. Each image line is focused at a different object distance, consequently there is always one line in focus when the probe is rotating. The output picture frames of the CCDs are grabbed by an image processor and sent it in an organized manner to the image memory. The picture analysis is carried out off line after the viewing is completed.

The IVVS mechanics partly shown in Fig. 4.2 consists of various subsystems:

- insertion system for guidance, alignment and insertion of the view probe
- view probe for illumination and viewing, all optics mounted in the optical bench
- scanning mechanism for probe rotation
- key-lock for primary shielding against neutron radiation
- vacuum hardware for separating the primary- and secondary vacuum from atmosphere
- radiation shielding

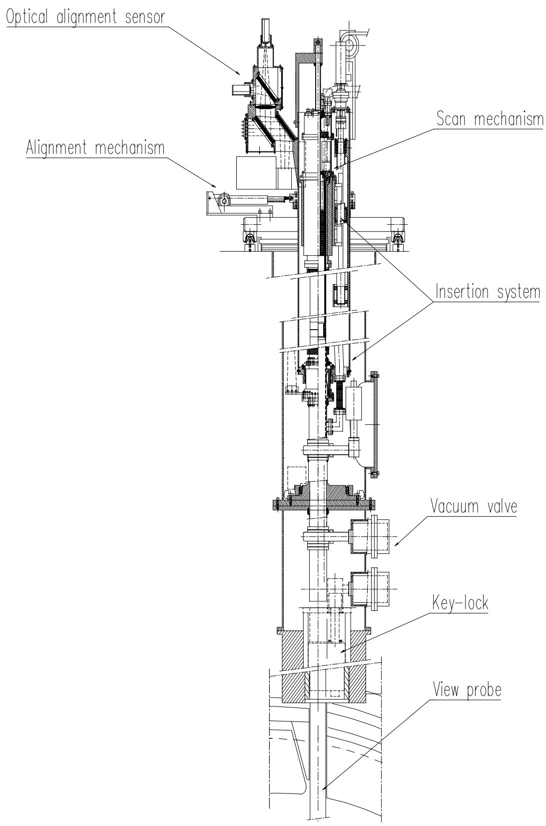
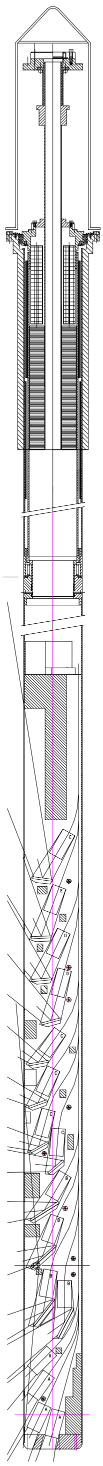


Fig 4.2. ITER/IVVS mechanical system.

The insertion system consists of a ball screw driven by a servo motor and linear guides mounted on the inner wall of the IVVS guide tube. With the aid of the long bellows surrounding the view probe, the insertion can be performed without affecting the primary ITER vacuum.



The view probe (shown in the left) is exposed to the plasma chamber and is designed to experience the harsh environment. It consists of a non-rotating support tube, a rotating viewing tube, imaging and illumination optics (which include the mirrors, the lenses, the optical fibres, and the CCDs). The rotating tube carries the optics and is rotated by the scan mechanism. Also, the illumination laser beam is guided within the rotating tube into the dispersing mirror. The probe outline is a smooth Ø150 mm diameter tube for its entire length to facilitate the insertion to the vessel.

The illumination system (see Fig 4.3) is designed to use four lasers emitting collimated beams that are expanded and combined together. The upper part of the illumination lobe is generated on the upper surface of the primary (dispersing) mirror, whereas the lower fan uses the beam reflected from the secondary mirror. The illumination beam has divergence angles of 2.5° ∗ 170° (horiz. ∗ vertical).

The imaging optics consists of 16 lens-fibre packages that each cover a FOV of $12° \cdot 12°$ (combined $12° \cdot 162°$). All are mounted into one rigid frame called the optical bench. The packages are of five different designs to meet the field-of-view and pointing requirements and to fit in the rotating probe tube. The optical fibres are bundled to cables that are attached to the rotating tube wall by stainless steel clamps. The picture resolution is determined by the number of fibres per degree and by the number of laser pulses during 360° rotation of the probe. With the proposed parameters (80 fibres/°, 36000 pulses/ 6 min) the quasi-spherical field of view contains appr. $5 * 10^8$ pixels. This corresponds to a spatial resolution slightly better than the requirement (1 mm at 3 m).

Fig 4.3. The view probe.

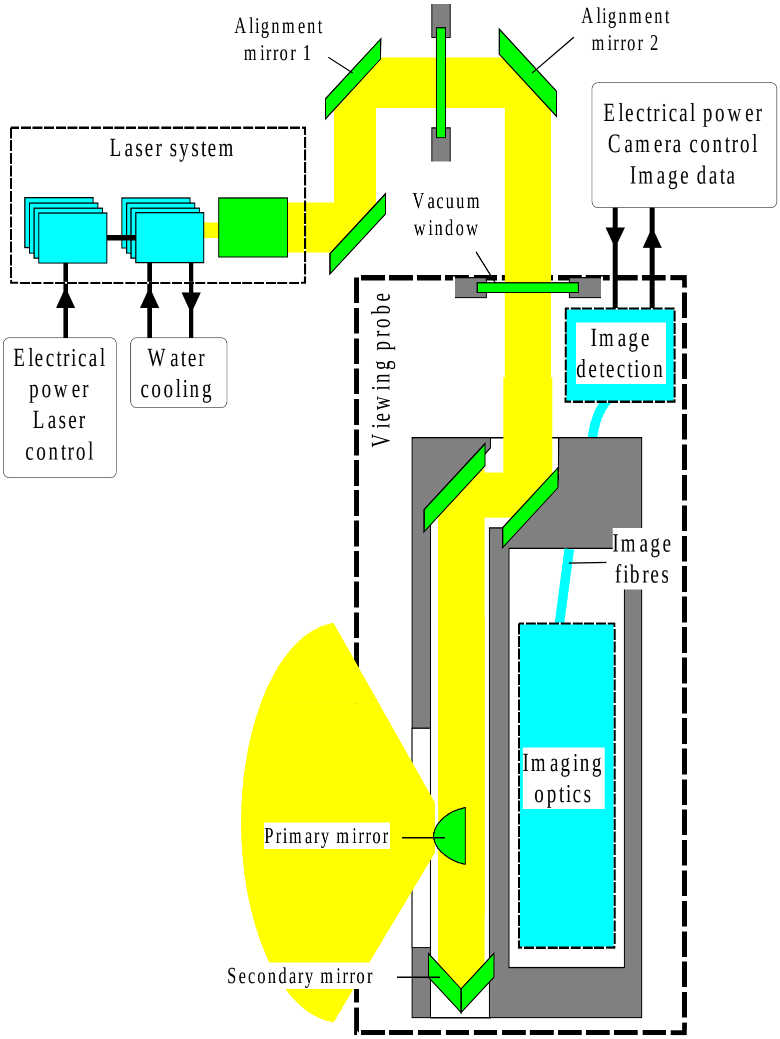


Fig. 4.4. Schematic layout of the IVVS illumination and imaging optics.

Pulsed Nd:YAG laser with second harmonic generation (λ = 532 nm, pulse repetition frequency = 100 Hz, E_p = 200 mJ) was selected as the illumination light source, but, in the future, it should be replaced with a more powerful one. The state-of-the-art CCD detectors with 1024 · 1024 elements have a maximum speed of 40 frames/s, but this is increasing all the time. Therefore it is probable, that within few years suitable detectors with a speed of 100 fr/s will be commercially available.

The optical coupling between the fibre array and CCD detector can be made either with lens optics or with a fibre optic faceplate. The lens coupling is advantageous, because it allows the free selection of the detector type whereas only a few detector manufacturers offer faceplate options.

The computerized control and image processing system for IVVS (see Fig. 4.5) can be divided into two parts: 1) viewing, which includes probe control and image forming operations and 2) the user interface, which includes only tasks for visualizing the complete images. The only hardware component shared by the two is the image database where the complete images are stored.

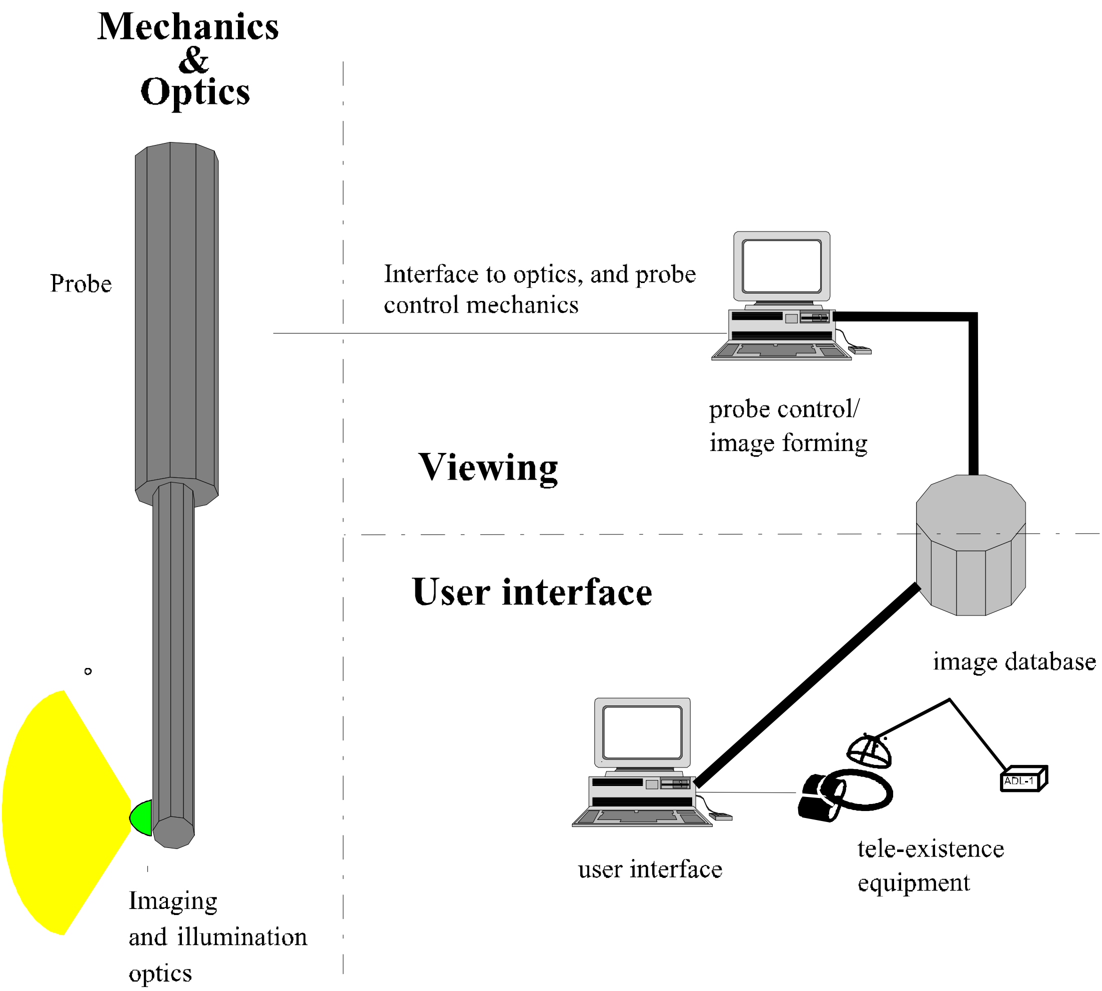


Fig. 4.5. Probe control and user interface hardware.

The computer control system is required to:

- control probe movements into and out from the vessel
- open and close keylock
- control the probe rotation during viewing
- adjust and align the probe
- control the auxiliary equipment like cooling, etc.
- control (synchronize) the illumination laser
- grab the line images (pixel stripes)
- form the complete images from grabbed pixels

Computer hardware consists of three parts:

1. For probe and keylock control: I/O-cards, motor controllers, a laser controller and a control computer
2. For image forming: A/D converters, frame grabbers, image forming computer (same as control computer)
3. For user interface: user interface computer, head tracker, virtual helmet, database for images

The image formation is an integrated process of image capture, transfer, storing, processing and visualization. A 10-probe IVVS requires a 100 Mbits/sec data transmission rate, 12.5 Mbyte/sec total disk transfer rate, and 5 Gbyte storage capacity per 1 image of the whole vessel.

4.1.3 Optomechanical Prototype

When the IVVS feasibility study and system level design was completed it was decided to build a complete end-to-end imaging prototype in order to investigate all factors affecting the image quality. This was accomplished in two stages. First, a full scale mechanical prototype of the scan mechanism and view probe was erected in the IVO Technology Center in Helsinki and it was outfitted with a CCD imaging system and an illumination laser. Separately an optical prototype of the illumination and imaging optics was built at VTT Electronics in Oulu. Preliminary imaging tests were made with both systems. Finally, the optical prototype was integrated in the mechanical prototype and representative imaging tests were performed.

The 14-m tall full scale prototype has all the essential components of the IVVS needed for viewing. No insertion system or vacuum hardware is included. The components selected are commercial units, which can be developed to ITER qualified versions. The scan mechanism includes a stepper motor with internal planetary gear, an optical position encoder and structures required to support and rotate the view probe. The view probe consists of two concentric AISI 316 stainless steel tubes. The inner tube is rotating and carries the optical bench in the low end. The whole prototype is supported by a rigid platform where the scan mechanism is bolted. The prototype is surrounded by a tower, which is sealed and air conditioned to maintain cleanliness for the optical components. A sector of the lower part of the tower can be opened for viewing of targets at appropriate distances.

The illumination is provided by a frequency-doubled Nd:YAG laser with beam expander optics, both mounted on the top of the scan mechanism. The optical bench (see Figs. 4.6 and 4.7) contains the laser beam steering mirrors and, for imaging, one lens module and one coherent fibre array (3 m in length).The CCD camera has SVGA detector (1280 * 1024 pixels), which can be cooled down to -12°C. In the prototype the illumination beam has divergence angles of 5° * 17° only, which, however, is sufficient in the case of one lens module. This was found necessary, because the laser has a very limited output power for this application.

Fig. 4.6. The optical bench with CCD camera and optical fibre bundle.

The control computer for the prototype is a Pentium with a 200 MHz processor. It is used for probe rotation and imaging only. All control functions for imaging are done with the frame grabber card. The maximum grab rate of the CCD camera with a full image is about 8 frames/s. The laser is synchronized with an external trigger signal and the maximum pulse repetition rate is 20Hz.

In the beginning of an imaging sequence the probe is driven to the start position identified by a limit switch. From this position the probe is rotated with constant angular velocity set by the user. The grab control program uses the angular position to trigger a single stripe image grab. Stripe images are collected to sector images and, finally, the whole panorama image consists of several sector images.

4.1.4 Imaging Tests

The performance of the IVVS optics only was measured by using a modulation transfer function (MTF) test chart. The target distance was 2 m, and the chart was illuminated by a white light source. The test results indicated that the minimum resolvable object size at 2 m distance is about 0.5 mm. This value is limited by the MTF of the fibre array, and it corresponds to the maximum theoretical resolution that can be obtained with this arrangement.

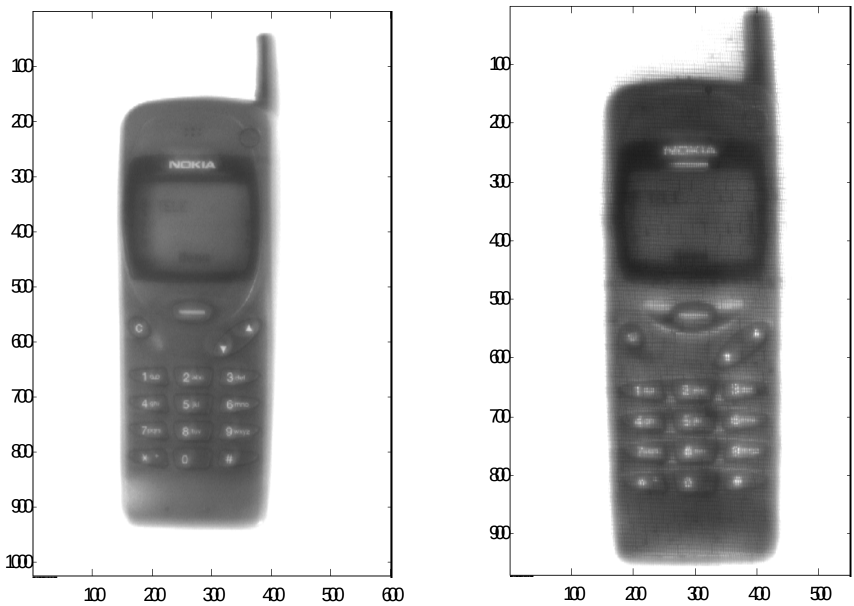
Some natural targets were also used in order to determine, how human eyes see the received picture. One of the targets was a mobile telephone, whose image with and without the fibre array is shown in Fig. 4.7. The target distance was 3 m, and the chart was illuminated by a bandpass filtered white light source. The line width of the button numbers is about 0.5 mm, and the line width in letters adjacent to numbers is about 0.25 mm.

The performance of the complete prototype was evaluated with various test patterns and natural objects. One example is shown in Fig. 4.8: this image consists of 0.1° stripes, which were obtained by the rotating view probe in steps, and, between each step, illuminating the object with 20 laser pulses.

4.1.5 Conclusions

The preliminary imaging tests indicate that the optomechanical prototype operates satisfactorily and the picture quality is as expected. Mechanical tests (vibration, bending) have revealed that the prototype remains functional under realistic external loads. Further work is still required to measure the imaging resolution at different object distances and observation angles. In addition, objects more representative to the ITER vacuum vessel surface will be tested. Also other candidates for optical fibre should be characterized. The major problem with the prototype is insufficient illumination (laser power), which prevents viewing with the probe rotating uniformly. Alternative solutions to enhance the illumination (more powerful laser, narrow illumination beam) should be considered.

The current IVVS design is based on the requirements in ITER Interim Design Report and ITER Design Description Document, and it is intended for viewing the ITER blanket only. In July 1997 revised specifications were issued by the JCT. In order to meet the requirements for divertor viewing and location under the ITER bioshield, the effective stroke of the probe must be increased. An additional telescoping motion must be added to the insertion system designed, although this may be extremely difficult. Also in this case, alternative solutions for insertion should be investigated.



(a) *(b)*

Fig. 4.7. Image of a mobile telephone at 3 m distance
a without any fibre array,
b with the prototype fibre array

The IVVS project has been useful for the participating research organizations in providing access to the ITER organization. The expertise developed can be applied to other areas of technology, where ultimate environmental conditions limits the use of more conventional methods. In fusion research, co-operation with ENEA in Frascati, Italy is under consideration. They are developing a laser viewing system for JET and the modifications required for the ITER application will be studied.

Fig.4.8. A test image (distance 2.5 m) obtained by rotating the view probe in steps.

4.2 Water-Hydraulic Remote Maintenance Tools for ITER

4.2.1 Introduction

The Institute of Hydraulics and Automation (TUT/IHA) has been working for ITER since 1994 as a member of the team developing ITER divertor maintenance and component-handling systems. The work has been carried out under the supervisory control of the NET Team, the ITER Joint Central Team and the ITER EU Home Team from the very beginning. Other project partners have included ENEA CR Brasimone, Hytar Oy, NNC ltd and IBERTEF-SENER.

IHA's work started in 1994 with a feasibility study of water-hydraulic technology for fusion reactor use. In 1995 IHA studied divertor cassette replacing and transporting equipment and developed systems for heavy reactor component handling. In 1996 IHA worked on a divertor cassette refurbishment system developing equipment for disassembling and assembling used reactor components. During 1997 IHA continued development work and started manufacturing prototypes of the most interesting maintenance tools. IHA has also developed a new type of power unit and control system for the tools. During 1998 IHA delivered prototype tools to ENEA, Brasimone and will continue test and development work in co-operation with ENEA. New joining ideas are also being tested in the IHA laboratory, for which a tool prototype will also be manufactured.

Apart from the actual contract-based work, IHA has carried out research work aimed at future applications of ITER remote handling systems. One important project is the research and development of motion control and force control for water-hydraulic systems. Another one is the research work on a hybrid model-based and force-feedback teleoperation interface for hydraulic systems.

4.2.2 Water-Hydraulic Feasibility

IHA projects started in 1994 with a feasibility study on water-hydraulics in the fusion environment. During the project, the emphasis was on the applications inside the reactor vessel, where operation conditions and requirements are extremely harsh. During the project the interactions between the hydraulic systems and fusion environment were studied.

The conclusion of the study was that hydraulic technology can be used in fusion environment with some limitations:

∗ Oil or glucol hydraulics is not allowed due to the risk of contaminating the reactor and its components. High radiation also activates oil, which turns into dangerous waste after usage. Instead, demineralized water can be used even inside the reactor.

* High radiation inside the reactor makes commonly used elastomeric materials brittle after a short period of use, which limits the use of many standard components, like common sealing materials and hoses. Radiation-tolerant components set their own requirements on design, which has to be taken into consideration.

* Temperature inside the reactor vessel during the maintenance period is relatively high (> 50 °C), and therefore cooling of water-hydraulic systems has to be arranged.

The dangers of radiation and contamination are present in any operation in the ITER environment, but are more serious in the reactor vessel than outside the vessel.

In many ITER applications, the advantages of water-hydraulic systems are clear compared to pneumatics and electromechanics:

* Inside the reactor vessel space is very limited, which highlights the importance of actuator power density.

* Simplicity and reliability are essential for the systems. With hydraulic technology, ~~-~~linear and rotational motion can be achieved without transmission.

* Static loading of hydraulic systems does not overload the system. Instead, it is a very typical situation in many hydraulic applications.

A disadvantage of water hydraulics is the limited component selection, especially as regards advanced control components. However, some water-hydraulic servos or comparable components are already at the laboratory stage and are soon expected to be on the market.

In a hostile environment like ITER, the critical factors are space and reliability. The simplicity provided by hydraulics has a close relation to reliability. However, the design work of maintenance equipment has to be done with special care to select the right operation principles, mechanics, actuators and control in the right place.

4.2.3 Divertor Cassette Replacing and Refurbishment

After the water-hydraulics feasibility study, IHA participated in the design and development of a divertor cassette replacing and refurbishment system. Divertor replacement and refurbishment operations are all done by remote handling and they are classified to require test platforms to verify operations and to analyze their requirements further. The DTP (Divertor Test Platform) and DRP (Divertor Refurbishment Platform) are two test platforms for verifying

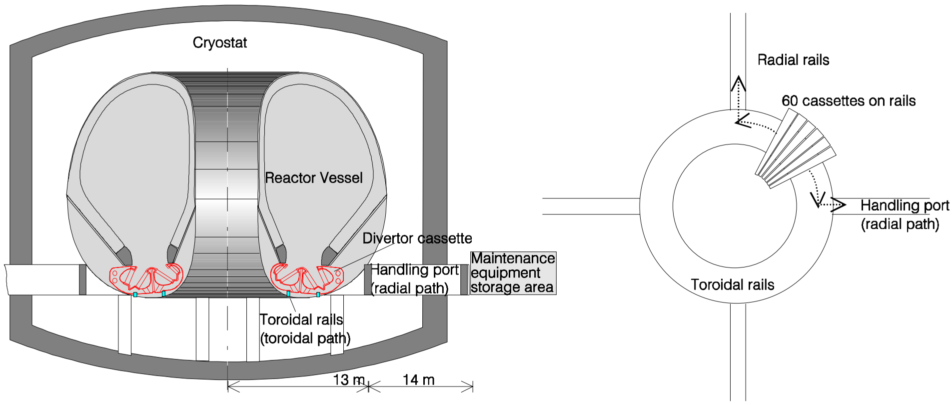


Fig. 4.9. At the bottom of the ITER toroid-shaped reactor vessel are the divertor cassettes which are replaced through four handling ports.

operations and systems for divertor cassette replacing and refurbishment operations. The platforms are located in Italy, at the ENEA C R Brasimone site.

4.2.4 The Divertor Cassette Replacing

At the bottom part of the ITER fusion reactor toroid-shaped vessel is the divertor region (Fig. 4.10), which acts as a target for reaction waste particles. The divertor consists of 60 cassettes, each weighing about 25 tons. Due to the harsh operation conditions, the cassette plasmafacing components (PFCs) may be damaged and need to be replaced eventually. Due to the remaining radioactivity of the reactor vessel and the used cassettes, all the refurbishment operations are carried out by remote-controlled equipment.

During 1995 IHA was participating in the design work of the system used to remove, transport and replace the divertor cassettes. The aim of the project was to study the maintenance of the divertor components and to generate a reference for the final design.

In-vessel operations of divertor cassette replacing and transporting include heavy component lifting and supporting. Inside the reactor a carriage operates with two lifting-forks that move along the rails. The forks operate with water-hydraulic cylinders for lifting the cassette away from the rails and for supporting it during the transportation. Space for the lifting mechanisms under the cassette is very limited, and therefore the water-hydraulic lifting system is advantageous due to its compactness and simple design. High radiation prevents the use of polymeric sealing materials. Therefore, the cassette-lifting cylinders are sealed with metallic sealing rings and the cylinders are hermetically sealed with metallic bellows that collect internal leakage. During the design, the remote

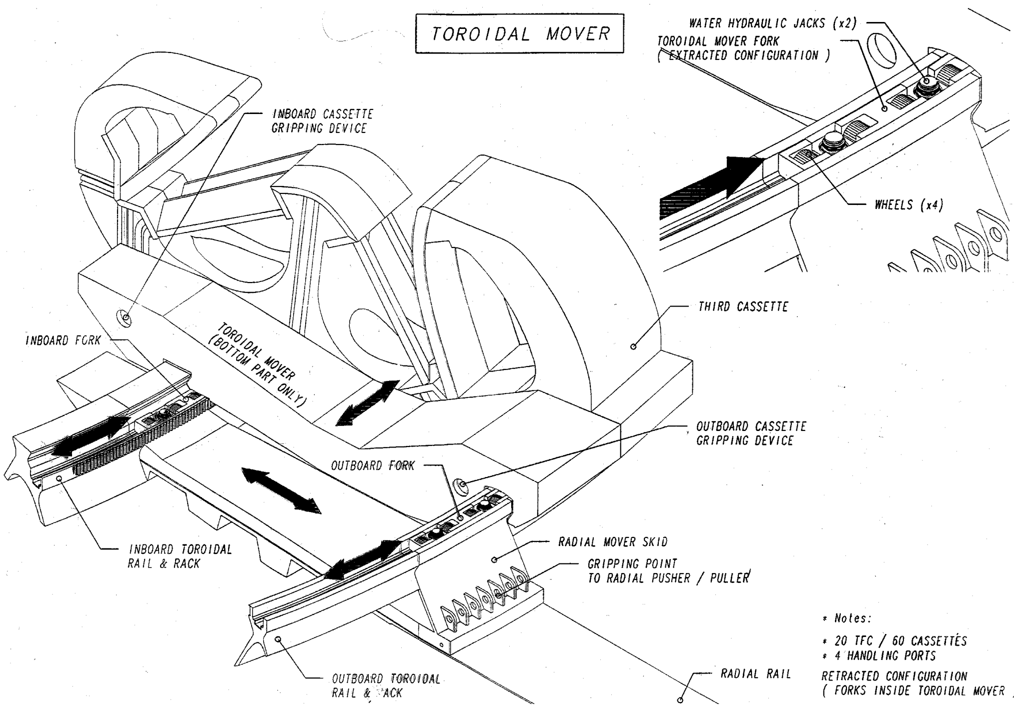


Fig. 4.10. The cassette mover with hydraulic lifting-forks.

maintenance of the lifting system was also taken into consideration.

Preliminary design work of some other special systems for the divertor component mover system was also carried out, such as cassette-gripping and docking systems and RH system umbilical management. The preliminary design of a water-hydraulic power unit was made. There were two options for the location of the power unit: 1) operate inside the reactor vessel, or 2) operate outside the reactor vessel.

Inside the reactor vessel, the power unit was designed to be operated only for very short periods in order to avoid overheating because the temperature is about 50 °C. Therefore, large accumulators were used. Inside the vessel, long hydraulic hoses are not needed. The idea of placing the power unit inside the reactor was rejected due to uncertainties of the temperature. In addition, the radiation presented problems with sealings, accumulators and electronics. For the power unit option located outside the reactor vessel, the practical problems lay mainly in the fail-safe umbilical system requiring a long free length and 90-degree angle.

The system designed during the project was used as a reference solution in the Call-for-tender for DTP equipment, which was submitted in late 1995. In May 1998, most of the test systems were delivered and operating.

4.2.5 The Divertor Cassette Refurbishment

Due to the harsh operation conditions, the divertor cassettes have to be replaced eventually. To minimize the amount of high-active waste generated during the divertor maintenance, the heavy divertor cassette body is designed to be re-usable, and its plasma-facing components are designed to be changeable. Divertor cassette refurbishment (i.e., component changing) is carried out by a remote-controlled system in the Hot Cell.

The plasma-facing components (target in Fig 4.11) are fixed to the cassette body by four locking elements (shear keys, Fig. 4.12 and Fig. 4.13), which are designed to withstand the harsh operating conditions and to be replaced by remote-controlled tools. The shear key has two ends and a connecting tie bar between them. Once inserted into the keyhole, the shear key is tightened by pushing in three wedges integrated into the ends of the shear key.

Locking elements (shear keys) and tools for their replacement and handling were developed at the same time. The first prototypes of shear keys and tools were designed for outer vertical target element replacement. Handling equipment for the DRP was also considered when designing replacement tools.

The shear key is designed to provide 0.5 mm clearance when inserting the key into the key way. The assembly clearance is closed and the key is tightened by inserting the three wedges approximately 22 mm, which expands segments at the key end. The connection is loosened by extracting the three wedges of the shear key, after which the key can be withdrawn from the key way. For generating adequate pre-loading for connection, the insertion force of the two wedges of the cylindrical end of the key is 2 300 N each, and 4 600 N for the dovetail end wedge. The estimated extraction force of the wedges is double the insertion force. For proper locking with the key, both wedges of the *cylindrical end of the key* should be *inserted synchronously*, maximum synch. error being ±0.5 mm. No special requirements were set for the dovetail end wedge-insertion motion, or extraction motion of any of the wedges.

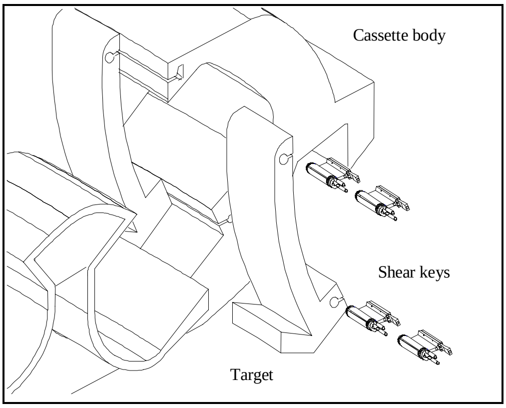


Fig. 4.11 Target is connected to divertor cassette body with four shear keys.

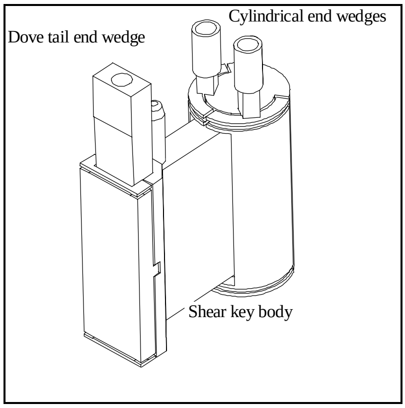


Fig. 4.12. Shear key ends are expanded by inserting three wedges. On key body, there are also two studs for connecting key to SKWE-tool.

IHA's task was to design and manufacture tool prototypes for shear key tightening, releasing and handling, and for target supporting and handling. While opening target-to-cassette connecting shear keys, the target is supported by a bridge crane with a special type of lifting interface, C-hook, providing 'floating' support for the target during shear key installation. The C-hook was designed and delivered to ENEA CR Brasimone by IHA.

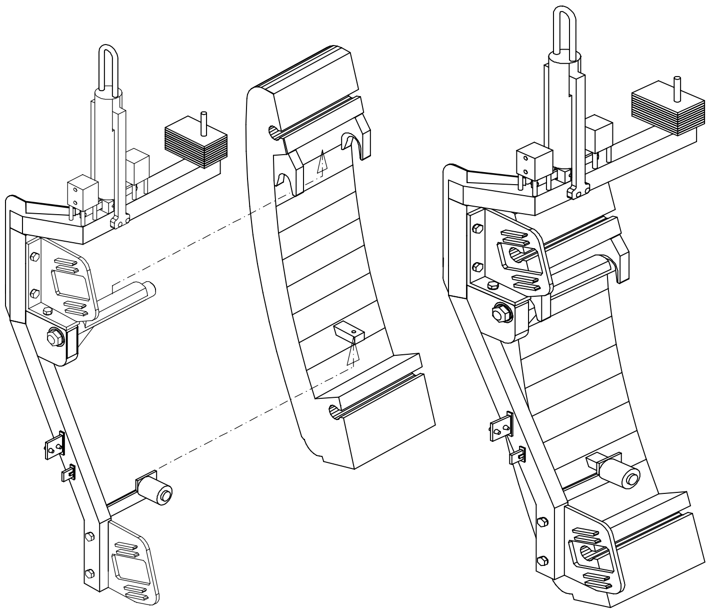


Fig. 4.13 Target element handling tool, C-hook.

In the shear key and wedges' insertion and extraction tool (SKWE-tool, Fig. 4.14) are integrated three hydraulic cylinders and their position transducers (potentiometers). The insertion/extraction force of the hydraulic cylinders are transmitted to the wedges via push/pull rods going inside the tool extension probe. The push/pull rods are connected to the wedges with threads by turning the rods at the opposite end with an electrically driven bolting tool. The tool body is connected to two threaded studs of the key body by two connection rods. In the middle of the tool there is also a pneumatic impact cylinder, which can be used to supply impact to the key to increase the effect of tensile extraction force in case the wedges get jammed.

The size and location of the cylinders are determined by the key and key-hole sizes. The required fluid amount for one cylinder stroke is 7.4 ml / cylinder. Due to very small flow and the required synchronization of the two wedges on the cylindrical end of the wedge, any standard solution could not be used for flow control, and therefore a new concept for control was developed.

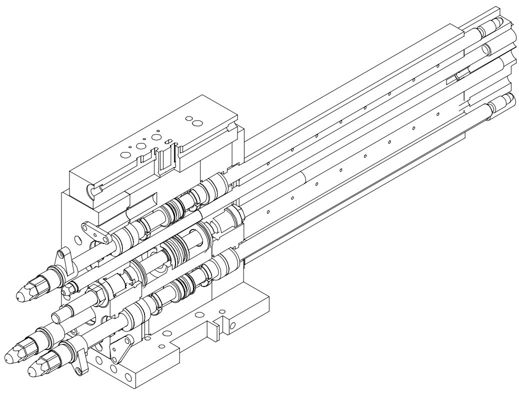


Fig. 4.14 SKWE-tool block includes three hydraulic cylinders and one pneumatic impact cylinder. The tool extension probe is inserted into the key way and two connection rods and three push/pull rods (cylinders) are connected to the key by turning them at the opposite end.

The three insertion/extraction cylinders of the SKWE-tool are controlled by a hydraulic power unit. The cylinders to be moved and motion direction are controlled by direction valves. Cylinder speed, and synchronization, are controlled by controlling two separate pump units feeding the SKWE-tool. The two pump units are stepper motor-driven water-hydraulic pump cylinders, Fig. 4.15. With the stepper motor and ball screw transmission, and pump displacement can be controlled with high accuracy. To feed two extraction cylinders simultaneously, two pump units are necessary. Stepper motors are controlled by PLC according to information received from position transducers integrated into the SKWE-tool.

System control is achieved with a PLC SIEMENS S7-200, which controls two stepper motors, five hydraulic valves and three pneumatic valves of the SKWE-tool impact cylinder, and also other auxiliary systems. The operator can change several operation parameters of the pumps. These parameters are: Two stepper motor speeds (normal speed and closing speed), upper and lower limit values for two pump synchronization, pressure limits and travel limits.

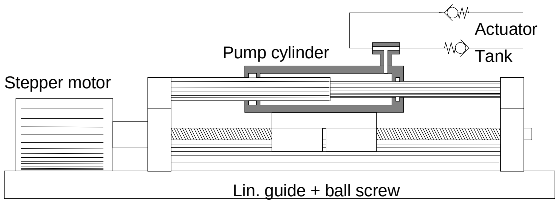


Fig. 4.15 One of the two pump units of the hydraulic power unit.

4.2.6 Other ITER-generated Research Projects

Water hydraulic test bench

One optional locking element for the component-cassette connection required high extraction forces. The most suitable way to disassemble and assemble these elements is to use a water-hydraulic cylinder tool to generate the extraction force. The control of the extraction motion at the very moment of breaking the jam and controlling extraction/insertion speed under varying load/friction are very important properties. To meet the control requirements of the ITER tasks, the IHA laboratory has established a test bench for developing and testing different control methods. The test bench is a two-axis water-hydraulic x-y table equipped with a gripping system for gripping and breaking test bars for materials testing. Both the axes are controlled by a PC. The PC is connected to a graphical model in the Telegrip-software in the Silicon Graphics workstation via Ethernet.

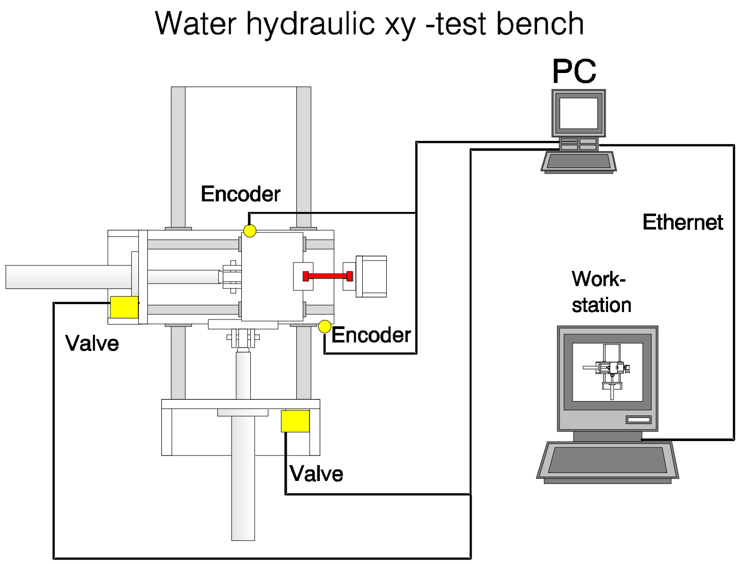


Fig. 4.16 Water-hydraulic test bench.

The test bench will be used to develop the motion control for the ITER shear key extraction. By breaking apart the test bars, sudden load changes can be simulated and methods for their control developed. The results can be utilized to better control also stick-slip phenomena, which is especially difficult with water-hydraulics.

Besides the control problem in breaking the test bars, the test bench provides a test platform for different valve-types and control methods for single-axis and x-y motion, position and force control. The test bench will also be used to verify the general properties of water-hydraulic proportional and servo valves and to verify water-hydraulic simulation models and thus develop the simulation of water-hydraulic systems.

For position control, the known hydraulic control methods, like 3-state and 5-state control and position servo control will be tested for water-hydraulics. Also, on-off and servo operating together will be tested. Water-hydraulic on-off type valves with sophisticated control algorithms developed, for example with pulse width modulation, will also be tested.

Telegrip with the test bench's graphical model in the workstation is used to teleoperate the test bench. With Telegrip, the user generates target points in the x-y plane. The target co-ordinates are sent to the PC that controls the test bench, against which the measured positions are sent to the workstation when motion proceeds. The user gets visual feedback on the workstation screen. The aim of the study is to test different options in transmitting target locations to the device, how to get the best feedback and how to be able to control the interruptions of the manipulator.

4.2.7 Conclusion

During the projects it has become obvious that water-hydraulic technology provides a simple and reliable solution for many ITER divertor cassette refurbishment operations, part-handling operations and cassette-manipulating operations. The considered technology also offers several advantages for devices operating in critical environments.

Water-hydraulics has traditionally been used in very harsh applications. The recent strong development of components provides the possibility to build more sophisticated applications and devices with similar capacity and control properties as those of oil-hydraulics without the disadvantages of oil-hydraulic systems. Tool prototypes delivered to DRP have also proved that the accurate control of a water-hydraulic system is possible with current technology.

5 Socio-Economic Studies

The Socio-Economic Research on Fusion (SERF) was started in autumn 1997. The SERF programme includes five subtasks: Long-Term Scenarios, Production Costs, External Costs and Benefits, Fusion as a Large Technical System, and Fusion and the Public Opinion.

VTT has participated in the subtask External Costs and Benefits of Fusion. Other participating institutes are Studsvik, Risø, Max-Plank-Institut für Plasmaphysik, CIEMAT and ENEA. With the aid of the concept of external costs, preferences of parties other than producer and consumer are taken into account by evaluating these preferences in an extra price term. In evaluating the external costs, the project uses the EU ExternE methodology, where the bottom-up, site-specific, marginal approach is used. Monetarization of caused damages is an important point in the evaluation process. The monetarized environmental costs calculated as mECU/kWh might be used as a part of price to determine the total costs of energy production.

Material from the earlier SEAFP study (Safety and Environmental Assessment of Fusion Power) has been used to find the necessary information for model plants. Some parts are taken from the ITER design and conventional fission reactors. Two conceptual power plant designs producing 1 000 MW of electricity were considered. Model 1 applies helium cooling and vanadium alloy structures for the components near the plasma, thus emphasizing the use of low-activation materials. Model 2 is based on reduced activation martensitic steel for the structures and uses water-cooling. The plants are assumed to be situated in Lauffen near the River Neckar in the south-western part of Germany.

On the basis of the preliminary studies performed, external costs of fusion have been evaluated to be very low for the Model plant 1 (less than 1 mECU/kWh) and also relatively low for the Model plant 2 (about 2 mECU/kWh). So far the disposal component has not been monetarized. If that component has to be low in the evaluations using the ExternE methodology, it is necessary that decommissioning is performed using deep repositories where solubility and water flow are very small.

In the environmental costs of fusion, some cost components seem dominant. Carbon dioxide emissions caused by the production of plant materials and from constructing the plant give rise to a relatively high cost component due to the global-warming impact, which is the same for both model plants. Of course, the component is very small compared to energy technologies using fossil fuels. These costs might also be avoided if future energy systems do not emit large amounts of greenhouse gases. Then, however, the alternative energy production measures also have lowered environmental costs. (Another possibility is to consider that the necessary electricity for material production is produced by using fusion power and not the average energy system.) Other important emissions are C-14 emissions due to normal operation (Model 2) and due to decommissioning.

A relatively high amount of C-14 has been evaluated to be produced in shield materials in the SEAFP project. If the effects of this C-14 inventory are estimated taking long-term global impacts into account, the C-14 component becomes very important. Disposal can cause, using ExternE methodology, relatively high extra contributions to external costs. If, in commercial fusion plants, C-14 releases due to normal operation can be avoided, the disposal component is even more important. A conclusion is that shield materials should be further studied in order to avoid to a large extent activation to C-14.

Both important cost components are due to global impacts. Local impacts are, in the preliminary study, considered to be lower by some orders of magnitude than the global impacts. Therefore, the site of the fusion power plant is not very important in environmental considerations. On the other hand, the site for the waste disposal has still to be chosen rather carefully.

Methodological questions are very important in the evaluation of environmental impacts of fusion. In the SERF project, EU ExternE methodology is used. It offers good possibilities for comparisons with other energy production technologies that have been studied using the same methodology. On the other hand, comparison inside ExternE is a bit too straightforward if valuation questions are taken as given. For instance, the comparison of the most important cost components — global warming and the long-term doses — is not as easy as must be assumed in monetarization approach. The context within which the costs are evaluated also influences the costs considerably. Emissions as well as impacts caused are dependent on the energy production scenario under consideration. The SERF task “Long-Term Scenarios” includes material that could also be used in the calculation of external costs. It is also still possible that some rather important sociological aspects have not at all been taken into account so far.

6 Summary of Objectives and Main Results

6.1 Meeting of the FFUSION Programme Objectives

The early objective of the FFUSION programme was to collect and organize all the fusion-related activities under one single research programme and to identify potential research areas where Finland could make a relevant contribution to the European and international fusion effort. This was the first step in preparing the formal association with the European Fusion Programme. The important role of industry in the future R&D of fusion technology was realized and a systematic survey to find potential Finnish companies was undertaken.

Those early objectives were well achieved when the FFUSION programme was fully integrated into the European Fusion Programme in 1995 and the Association Euratom-Tekes was established. By this time a number of Finnish hi-tech companies had expressed their interest in fusion technology and started to collaborate with the Finnish Fusion Research Unit. In 1996, eleven Finnish companies were included on the list of qualified industry for ITER EDA and Imatran Voima Oy became a member of the EFET consortium.

The national objective is to provide a high-level contribution to the EU Fusion Programme in focused areas of fusion science and technology. The vital element in reaching this goal has been close collaboration between research institutes, universities and industry.

During the six years of FFUSION, over 100 publications in international scientific journals, over 120 conference articles, 64 research reports, 11 general articles have been published. In addition, two patents have been granted. Three doctoral, three licentiate and eight graduate theses have been produced during the programme period.

6.2 Fusion Physics and Plasma Engineering – Objectives and Main Results

Characteristic of the work within fusion physics and plasma engineering in 1993-98 was a strong concentration on modelling and design effort on the European fusion facilities, e.g., ASDEX Upgrade and Wendelstein 7-AS in Germany, JET in England, Tore Supra in France and on the international ITER project. The main lines of the research were radio-frequency heating and transport processes, where distinguished expertise and knowledge have been acquired by the fusion and plasma physics group during the past two decades. This experience has also been valuable in supporting other activities outside the scope of physics within the fusion programme.

The orbit-following code ASCOT, various wave codes, and gyrotron models, developed at VTT and HUT, are internationally recognised, and have formed a basis for a number of task and collaboration agreements with other Associations and JET. New lines of research have been opened by starting particle-in-cell simulations to model lower hybrid heating. Two important plasma engineering projects were also launched on the central solenoid of a tokamak and radio-frequency vacuum window design and construction. The latter have significantly increased the contacts and common research activity of the group with industry.

6.2.1 Physics of Radio-Frequency Heating and Current Drive

<u>JET Task Agreement: Ion Cyclotron Heating and Current Drive.</u> The main motivation of the present Task, initiated in 1995, was to develop and use efficient codes to model high-performance ion cyclotron resonance heating (ICRH) experiments in JET tokamak. The work continues the close collaboration of JET, HUT and VTT on radio-frequency (rf) physics. The PION-code developed in JET, in parallel with with the transport code TRANSP, has been used to model the neutron production rate and heating in the recent high-power deuterium-tritium experiments. It has accurately reproduced the observed neutron emission in important high-power shots.

The 5D Monte Carlo ASCOT, which follows trajectories of charged particles in a tokamak, was developed at HUT and VTT during 1991-98. It has been interfaced with the JET magnetic background. Work is in progress to apply ASCOT for a detailed comparison of ion cyclotron heated ion distributions and transport with the JET experimental data.

<u>JET Task Agreement: Development of Radio-Frequency Modules for Transport Codes.</u> The development of lower hybrid radio-frequency modules for the transport codes used in JET was started during 1998. The work will consist of participation in the code validation on JET experimental data and the implementation of rf-modules into the transport codes and analysing experimental JET-data.

<u>ITER Task: Support of Physics and Engineering Design of the Ion</u>

Cyclotron System. *1) Mode Conversion and Minority Current Drive for Plasma Current Control.* The mode conversion of rf-waves has been modelled at VTT and HUT, and in certain conditions 100 % conversion has been numerically predicted. A number of experiments have verified the high conversion factors for the use of ion cyclotron waves in mode conversion current drive. Within the present Task, a parameter analysis of the conversion and optimization of antenna spectrum have been performed for ion cyclotron heating scenarios of ITER. Current drive with ion cyclotron waves has been studied by solving the drift-kinetic Fokker-Planck equation and by performing full Monte Carlo simulations with ASCOT. The driven current is mostly of diamagnetic origin and is significant in ITER only for hydrogen minority heating.

2) Advanced Launchers for RF-Heating. The requirement of size reduction of the rf-launchers in a compact ITER device may call for advanced antennas. The power handling and coupling of a folded waveguide antenna in ion cyclotron heating have been numerically modelled in ITER reactor conditions. The electric field values stay within the experimental breakdown limits for an antenna array radiating three times more power than a conventional loop antenna, whilst still fitting to an ITER port.

3) Alpha Power Channelling with Waves. It has been suggested that locally constrained waves could be used for converting fusion alpha particle power to D/T-ion energy and for enhancing alpha particle removal from the plasma. ASCOT-simulations have been performed to take into account a realistic alpha particle distribution, collisions, and the tokamak geometry. Harnessing this scheme in standard heating methods of tokamaks is found to be very difficult.

Tore Supra Collaboration: Analysis of Parasitic Absorption of Lower Hybrid Power. The generation of impurities has been observed in Tore Supra and Tokamak de Varennes (TdeV) when lower hybrid waves at 3.7 GHz have been launched in tokamaks. A possible explanation for the impurity production is the sputtering caused by fast electrons generated by the near field of the rf-launcher. Such electrons can be generated when part of the rf-power is absorbed within a short distance from the launcher. When the launched power is several megawatts, fast electrons containing a few per cent of the rf-power may damage the launcher structures.

The parasitic absorption of lower hybrid waves and the generation of the fast electrons near the launcher have been investigated by particle-in-cell simulations in collaboration with the Tore Supra Team. A particle-in-cell model of a lower hybrid grill has been developed and coupled with the SWAN code, which calculates the launched wave spectrum. Numerical simulations show that, depending on the launched spectrum, between one to ten percent of the rf-power is absorbed within a distance of a few millimetres from the launcher. This is in agreement with the experimental results. The modelling of lower hybrid launchers of JET has been performed as a part of the above described JET Task Agreement on “Development of RF Modules for Transport Codes” and also for ITER.

Gyrotrons for Electron Cyclotron Heating and Microwave Diagnostics. Gyrotron microwave source research has concentrated on advanced source development for electron cyclotron resonance heating and microwave diagnostics in tokamaks. Mode competition in coaxial gyrotrons and multifrequency gyrotrons have been studied in collaboration with the FZK, Karlsruhe. A high-frequency quasi-optical gyrotron has been studied in collaboration with CRPP Lausanne. A gyrotron operating close to 1 THz (3rd harmonic of 280 GHz) has been considered for plasma diagnostics applications based on the Collective Thomson Scattering (CTS). A kinetic code was developed to model CTS and it was benchmarked with a fluid-code used in JET. The code can predict the scattered spectra in high temperature plasma where fluid models are not valid. Scattering measurements are planned for alpha particle diagnostics in fusion plasmas.

A fast Fokker-Planck code was developed to model electron cyclotron heating in reactor conditions. The stabilization of magnetohydrodynamic modes in ITER by using frequency-tunaeble gyrotron sources was analysed with the code. An INTAS project was co-ordinated during 1995-1997 to study modern frequency-tunaeble microwave power sources, e.g., gyrotrons, which improved the present diagnostics and heating schemes in fusion plasmas.

6.2.2 Plasma Confinement and Transport

ASDEX Upgrade Collaboration: Transitions from Low to High Confinement. Achieving a transition from low to high confinement, so-called L-H transition, is an essential step towards ignition or high-fusion gain operation of fusion reactors. The mechanism behind the transition is not, however, well understood. Collaboration between VTT Energy, HUT and Association IPP Garching started in 1996 to launch detailed simulation studies to explain the L-H transition characteristics in the ASDEX Upgrade at IPP. To understand the dynamics of the L-H transition, experimental data from ripple-trapped neutral beam-injected ions at the transition in ASDEX were first reproduced with

ASCOT. The results from ASCOT and from a specially developed Fokker-Planck code show that the charge exchange diagnostics can measure the radial electric field with an excellent time resolution. ASCOT has been used to test some well-known theory models of the L-H transition. Work is in progress to resolve stationary and self-consistent neo-classical ion transport fluxes at the transition region in order to reveal the role of neo-classical transport in the L-H transition.

Electron Density Profile Measurements and Particle Transport Studies with Multichannel Interferometer at the Wendelstein 7-AS. A new multichannel microwave interferometer was build and used in 1995-8 for the Wendelstein 7-AS stellarator in Garching. Within an agreement between HUT and IPP Garching, software for interpreting the signals from the measurements has been prepared. The code was used to study the electron density profile evolution in conventional discharges and during density oscillations produced by injecting a harmonically modulated gas feed to the plasma edge. Electron density oscillations were produced by modulating the gas feed to the plasma, and the propagating electron density perturbation was measured with the multichannel interferometer. In the experiments, the diffusion coefficients were found not to exceed the neo-classical level. An inward pinch, not predicted by the neo-classical theory, was detected at the plasma edge.

6.2.3 Plasma Engineering Projects

ITER Task: Dielectric Window Prototype for the ITER RF Transmission Line. VTT, HUT, IVO Technology Centre, and Rauma Materials Technology conducted an ITER Task during 1995-98 on rf-window development for the ITER vacuum transmission line (VTL) of ion cyclotron power. The goal was to obtain specifications for constructing two prototype vacuum windows for the ITER VTL being tested at Oak Ridge National Laboratory in the USA, and to present a design of the window. It was required that the window is compatible with the ITER conditions, withstands the strong dielectric heating and related thermal stresses, is resistant to breakdown with appropriate arc monitoring, can be remote handled, and can be manufactured by welding the ceramics to the conductor. Finite element calculations of the temperature, stress, and electric field were performed to find the conical-shaped dielectric geometry in a coaxial with proper materials. Accurate neutron flux calculation for the VTL with the MCNP4 code was performed for the ITER ion cyclotron system. Based on the neutron calculations the window is placed at the vacuum vessel feedthrough. A special remote handling scheme based on a window complex inserted inside a stainless steel casing has been invented. The choice of titanium as the conductor material was necessary to provide a design with water duct cooling in conductors without active gas cooling on ceramics. However, joining ceramics to titanium in this scale with ITER requirements for vacuum tightness has not so far been demonstrated. Initial new brazing experiments with full-size window prototypes at VTT have provided a promising scheme to meet the brazing needs.

The work continues with the production of the prototypes and design and the production of new vacuum windows for rf-launcher prototypes at high power density. The group has also prepared cost estimates of the antenna and transmission line parts of the ITER rf-heating systems in collaboration with IVO Technology Centre.

Central Solenoid Development for Spherical Tokamaks. The central solenoid is the most critical magnet component in tight aspect ratio tokamaks. The project to design and construct an appropriate water-cooled solenoid conductor for the Globus-M tokamak at the Ioffe Institute in Russia was started in 1994 as a Tekes project at VTT, HUT, and Outokumpu Poricopper. A manufacturing method has been invented for producing a 66-meter-long high-strength hollow conductor and it has been delivered to Russia, where it will be wound and tested in 1998. Outokumpu Poricopper provided a similar solenoid conductor also for the MAST tokamak in Association UKAEA Culham.

6.3 Fusion Reactor Materials - Objectives and Main Results

The objective of the fusion reactor materials research in the FFUSION programme is to carry out high-level research and development by applying novel manufacturing and testing methods. The work has been focused on advanced materials, advanced joining techniques, fracture mechanics, environmentally induced cracking and structural integrity in order to estimate radiation damage of fusion reactor components. Also studies in the fields of plasma facing materials and superconductors have been carried out. The research and development are performed in close collaboration with industry to encourage and increase the competitiveness of national companies to participate in the EU fusion programme. The fusion reactor materials research work is performed in the framework of ITER technology, Underlying Technology and European Blanket Programme-Structural Materials programmes in collaboration with Associations Risø, CEA and CRPP Lausanne.

6.3.1 Characterisation of Irradiated Cu and Cu-alloys

The current design for ITER utilizes Cu-alloys in the first wall and divertor structures. The function of the copper alloy in the first wall is to dissipate heat produced by plasma disruptions and it does not provide structural support for the first wall. However, the Cu-alloy for the divertor is designed also for structural support of the divertor cassette in addition to heat dissipation.

The objectives of the Cu alloy activities were (i) to determine the fracture toughness behaviour of both canditate CuAl25 IG0 and CuCrZr alloys in the ITER relevant temperature and neutron fluence ranges and (ii) to determine the mechanisms of elevated temperature fracture of the candidate copper alloys.

The main results are: (i) fracture toughness of CuCrZr alloy is clearly higher than that of CuAl25 IG0 alloy in ITER relevant temperature and neutron fluence ranges, (ii) fracture toughness of CuAl25 IG0 alloy is highly sensitive to temperature and neutron irradiation, (iii) miniaturized bend specimens of CuCrZr alloy give comparable fracture toughness results with standard size specimens, (iv) fracture toughness of CuAl25 IG0 alloy is highly strain rate sensitive and anisotropic indicating creep as a dominant fracture mechanism.

It was verified that the fracture toughness of both Cu-alloys decreased with increasing temperature up to 350 C. However, the fracture toughness of CuAl25 IG0 is much lower than that of CuCrZr alloy and decreases to a very low value already at temperature of 200 C. This apparent difference in fracture behaviour of the candidate Cu alloys is due to differences in microstructure. Reduction in fracture toughness of CuAl25 IG0 is observed after neutron irradiation to a dose level of 0.3 dpa on the contrary to CuCrZr alloy where only moderate effect of irradiation is observed as compared to unirradiated condition.

Significant reduction in fracture toughness of CuAl25 IG0 alloy is also observed with decreasing displacement rate at temperature of 200 C. Thus, fracture toughness is strain rate sensitive. This kind of time dependent fracture toughness behaviour indicates that at elevated temperatures creep mechanism dominate the crack growth in CuAl25 IG0 alloy.

Marked fracture toughness anisotropy of CuAl25 IG0 alloy plate is also observed due to fracture plane orientation and crack propagation direction. The fracture toughness along short transverse plane is significantly lower than that along longitudinal or transversal plane. The fracture toughness anisotropy decrease with increasing temperature.

Different size and type SEN(B) and C(T) specimens give similar fracture resistance curves for CuCrZr alloy when crack extension does not exceed 25-

30% of the initial ligament size. This result indicates that the ASTM standard requirements for allowable crack extension and J-integral values are conservative for the side grooved high constraint type of 10 mm and 3 mm thick SEN(B) test specimens and that reliably fracture toughness data can be generated by using miniaturized SEN(B) specimens. Specimen size effect was also verified with CuAl25 IG0, Ti-alloys and F82H modified steel. By applying mixed mode (I/II) loading it is verified that mode I fracture toughness is not a conservative value for ductile materials like F82H modified stainless steel and CuAl25 IG0 alloy.

6.3.2 Cu/SS Joining Technology and Characterisation

The present design of ITER primary wall modules is based on multimaterial concept with stainless steel as a structural material, copper alloy as a heat sink and beryllium as a plasma facing material. The candidate joining method for manufacturing these components is hot isostatic pressing (HIP). Joining methods for ITER first wall modules and other in-vessel components and characterization of the joint properties in non- and post-irradiated conditions is essential for reliable reactor design.

The objectives of the Cu/SS joining activities were (i) to develop advanced joining methods for copper alloys and stainless steel, (ii) to characterize the integrity and fracture toughness properties of the Cu/SS joints and (iii) to further develop testing methods for miniaturized Cu/SS joint specimens.

The main results is summarized in the following (i) verification that explosion welding (EXW) and HIP methods are viable methods to manufacture Cu/SS components, (ii) determination of general fracture behaviour and fracture toughness of Cu/SS joints at ITER relevant temperature and neutron fluence ranges, (iii) development of ultrasonic and eddy current methods for integrity assessment of ITER first wall mock-ups and (iv) verification of the fracture toughness test method for Cu/SS joints which utilizes miniaturized single edge notched bend SEN(B) specimens.

It has been verified that EXW and HIP can be used to produce good quality Cu-alloy to SS-joints. The metallurgy and mechanical properties of as received explosion welding (EXW) and HIP joints are different due to basic differences in bonding methods. However, after appropriate post weld heat treatment microstructure and mechanical properties of EXW and HIP joints are similar.

Mechanical testing of copper stainless steel joints is complicated due to differences in elastic and plastic properties of the constituent metals. It has been shown that fracture in many cases occur within copper alloy, however, at elevated temperatures and under creep condition fracture occurs along the joint interface. The fracture toughness of the Cu/SS joints is lower than that of the constituent copper alloys and a further reduction is observed after neutron irradiation to a dose level of 0.3 dpa. In the studied temperature range the precipitation hardened CuCrZr/SS joints showed higher fracture toughness when compared to dispersion strengthened CuAl25 IG0/SS joints. The very low fracture toughness of dispersion strengthened CuAl25 IG0/SS joints is partly due to strong anisotropy of the CuAl25 IG0 alloy. Multiple HIP thermal cycles was shown to have only a moderate effect on fracture toughness of the joints, however, at elevated temperatures fracture propagated along the joint interface.

The integrity of various Cu/SS joints and ITER primary wall mock ups manufactured by EXW, HIP or rheocast methods have been successfully evaluated using ultrasonic and eddy current techniques. VTT participated in EU Home Team test programme of ITER primary wall mock ups by characterizing the integrity of the industrially manufactured Cu/SS and Be/Cu/SS mock ups before and after high heat flux testing. The main results so far indicate that precipitation hardened CuCrZr/SS mock ups have larger operational margin since first failures were observed at 7 MW/m^2 compared to dispersion strengthened CuAl25/SS mock ups which failed at 5 MW/m^2. No failures were observed at 0.75 MW/m^2 after 13000 cycles in either type of mock-ups.

6.3.3 Behaviour of Hydrogen Isotopes in First Wall Materials

The aim of this project was focused on the hydrogen cycle and it was based on studies of the physical interactions between plasma and first-wall materials, i.e., on studying the hydrogen isotope (H,D) behaviour in metals and in diamond-like carbon (DLC) films.

A study of hydrogen migration between impurity layers in Ta, Ni, W and stainless steel AISI 316L was carried out. The effect of ion-induced damage and annealings in the H/He retention was investigated. A stopping power of 5-100 keV He-ions in Ta, Ni, W, AISI 316L, Cu, Ni, Mo and Cr was studied.

Carbon-based composites or DLC-films are potential plasma facing materials in fusion machines. Investigations were also focused on the development of a method for the production of co-deposited DLC-layers with variable H/D-concentration via DIARC plasma arc-discharge coating method for studies of erosion, migration, trapping and O_2 gas exposure removal, and on the development, characterization and production of test samples of DLC/SiC composite coatings for erosion studies and thick DLC/graphite coatings for re-deposition studies. A study of the trapping, de-trapping and migration of hydrogen isotopes in DLC films and carbon-based composite materials was carried out. The impurities in the coatings, their depth distribution, sample thickness, density and amount of diamond-like sp^3-bonds were measured.

The annealing behaviour of hydrogen and hydrogen containing He-precipitates was studied in implanted and co-deposited samples. The migration of H and D in implanted coatings is well described with a diffusion equation and exhibits good Arrhenius behaviour.

A set of films grown in a hydrogen atmosphere for migration studies was also made. The H-concentration in different depositions was varied by changing the pressure of the H-atmosphere between 0.06 and 0.6 mPa. Annealing experiments showed a decrease in the hydrogen concentration with increasing temperature, hydrogen release and migration to the interface.

6.3.4 Fusion Neutronics

Capacity for neutron and gamma flux calculations for fusion reactors was established at VTT Energy in 1993-95. The objective is to provide nuclear analysis for the ITER design and to support Finnish industry in the design and supply of components for ITER. This was to be done by calculating the radiation environment of various components, which is important for selecting materials and proving shielding. Initially, the work was done using S_N-codes, such as ANISN, contained in the REPVICS software created for radiation calculations in fission reactors.

The S_N method, in which discrete directions for neutrons are selected, is not very useful in a complicated geometry of fusion reactors. The Monte Carlo method is better in such a geometry, and the MCNP program with the FENDL-1 cross-section library has become an international standard in the field and it has been explicitly chosen for the ITER project.

The MCNP-calculations have mainly dealt with the neutron and gamma flux in and near equatorial ports containing the launching structures of rf-heating systems In addition, other calculations have been performed to verify and compare the results calculated by other teams.

The calculations for a port containing an ion cyclotron antenna also provide data on the flux at the vacuum window position. Thus, there has been good synergy between the work done for the ITER Team and that done to support the ion cyclotron vacuum window project in the FFUSION programme. In addition, neutronics calculations and nuclear analysis for the ITER lower hybrid launcher have led to major modifications to the early launcher design. It has been shown that all rf-heating systems require more shielding than was originally envisaged to meet the limit for the shutdown dose in the cryostat region. Some work has also been done to determine the approximate flux at various locations in the In-Vessel Viewing System.

Thus the original objectives have been well achieved and useful contacts with the ITER Team have been established. One scientist from VTT Energy has, during the period 1996-1998 spent about two to three months per year in the ITER Team in Garching under the Visiting Home Team Personnel Contracts.

6.4 Remote Handling and Viewing – Objectives and Main Results

Close collaboration between research institutes, universities and industry, which was one of the main objectives in the FFUSION programme, has been well established in the remote handling activities. IVO Technology Centre has a design responsibility in the IVVS project and Hytar Oy is actively involved in the development work of the water hydraulic tools for ITER divertor refurbishment.

6.4.1 In-Vessel Viewing System

The visual inspection of the interior of the ITER tokamak vacuum vessel will be periodically required to check for damage caused by plasma operations and for planning maintenance interventions. The In-Vessel Viewing System (IVVS) designed is based on rotating line scanning technology using linear arrays of optical fibres to send images from a distance to sensors (CCD cameras). This concept originally proposed by the JET Team has been further analysed and developed in the FFUSION programme.

The complete IVVS consists of 10 identical units installed on top of the ITER torus 36° apart. For viewing, the IVVS probes are inserted into the vessel through vertical ports using long bellows as the vacuum barrier and through key-lock mechanisms which shield the probes from neutron bombardment during plasma operation. The complete picture of the vessel interior is generated by rotating each probe 360°. During viewing, the illumination is provided by pulsed high-power laser beams that pass through the probes and are diffused by mirrors on the vertical viewing plane. The line images from the optical heads are first transferred to CCD camera chips with optical fibre arrays, then grabbed by an image processor and finally stored in the image memory for further analysis. The IVVS is designed to complete the viewing cycle of the whole vessel in 6 minutes.

Design Activity: A detailed design of a single IVVS probe complete with mechanics, laser illumination, fibre optics, control electronics and picture analysis has been carried out.

The mechanics include the insertion system, which moves and aligns the view probe, the scanning system for probe rotation, the key-lock mechanism for shielding the probe during plasma operation, and vacuum hardware for separation of the primary and secondary vacuum from the atmosphere.

The optical head of the probe consists of 16 optical modules covering a vertical field-of-view of 16 ° each. Their lens optics is based on the double-Gauss configuration using radiation-hardened Schott glass material. Up to 10 linear arrays of 1000 coherent optical fibres are positioned vertically, next to each other, in focal planes of each lens package. Each array is focused at a different object distance, which means that there is always one image line in focus when the probe is rotating.

The control and viewing system control the probe motion, collect image data from the CCD camera chips, form the image from raw data and show the ready images to the operator. The images are also saved in the image database for further inspection.

Prototype Activity: Essential parts of the IVVS have been developed to prototype stage. These include: 1) a full-scale mechanical prototype of the view probe and its scan mechanism; 2) an optical prototype with an illumination laser, one lens package, optical fibre arrays and a CCD-camera, and 3) a computer system for scan control, frame-grabbing and picture analysis.

The mechanical prototype has been fully assembled and tested. Further tests will take place in the fall 1998 when the optical prototype has been integrated into the system.

Future Considerations: The current IVVS design is suitable for the inspection of the vacuum vessel walls (blanket) only. The current specifications (Naka JCT Remote Handling Group, July 1997) set requirements also on divertor viewing, which increases the probe length by several meters. In addition, the whole IVVS system is to be located below the ITER bioshield. This implies that the probe, the insertion mechanism, the optical bench and the shielding system selected (key-lock, plug) need to be redesigned.

The IVVS project has been useful for the participating research organizations in providing access to the ITER organization. The expertise developed can also be applied to other areas of technology, where ultimate environmental conditions limits the use of more conventional methods. In fusion application, co-operation with JET Joint Undertaking should be developed because the JET machine would provide a most realistic environment for the IVVS system.

6.4.2 Water Hydraulic Tools for Divertor Refurbishment

Tampere University of Technology, Institute of Hydraulics and Automation (TUT/IHA) has been working in the FFUSION programme since 1994 in the field of ITER reactor divertor cassette remote handling maintenance. TUT/IHA has been working in close co-operation with the NET Team, ITER Joint Central Team, Association ENEA and NNC Limited.

The divertor is located on the bottom of the reactor vessel. The divertor, consisting of 60 divertor cassettes, operates as a collector for particles from the plasma. Due to harsh operation conditions, the divertor cassettes have to be replaced eventually. To minimize the amount of radioactive waste generated during maintenance on the divertor, the divertor cassette body is designed to be re-usable, but its plasma facing components are designed to be changed in a hot-cell. The divertor maintenance operations are verified in two test platforms in Brasimone, Italy. The divertor test platform is to demonstrate the divertor cassette replacement and transportation and the divertor refurbishment platform is for cassette disassembly and assembly of fresh parts, i.e., refurbishment.

The aim of TUT/IHA work was to provide information for the system design of the two platforms and for required maintenance operations on expertise areas of TUT/IHA. In particular, the aim was to provide knowledge on water-hydraulic technology and to design and deliver tool prototypes for some maintenance operations. The aim was also to acquire information on the nuclear field and experience of new, demanding water-hydraulic applications.

During 1996–97 TUT/IHA worked on divertor cassette refurbishment developing tools for the replacement of plasma-facing-components. Prototypes of the most important tools were manufactured and have been delivered to the divertor refurbishment test platform in Brasimone.

The 1996 work also included development work of water-hydraulic motion control, for which IHA established a two-degree-of-freedom test bench at the early phase of the project. The test bench has been used for studying properties of control systems for ITER. The same environment is also being used for the development of the teleoperation system.

At the beginning of 1998, TUT/IHA started working on development project for cassette component joining methods. TUT/IHA will provide hydraulic test facilities for large force requiring tests, perform joining tests and develop joining tools on the basis of tests.

In addition to the work that has been carried out directly on certain ITER applications, TUT/IHA has also studied some topics important for more sophisticated future remote handling applications. These are, for example, advanced valve and control technology and alternative methods for accurately controlling flow and pressure, and the teleoperation of hydraulic-driven devices by combining model-based teleoperation and force-feed back.

The main results of TUT/IHA projects in the FFUSION programme are:

- Increased knowledge of water-hydraulics advantages in nuclear environments, specially in ITER.
- Input for the ITER divertor test platform and divertor cassette mover design process.
- Delivered prototype tools for ITER divertor refurbishment platform in Italy.
- Design information for ITER divertor element joining development.
- Increased knowledge and experience of the control of water hydraulics. A new method for accurately controlling water-hydraulic actuators was also developed.
- Increased teleoperation experience and first steps towards an easy-to-install teleoperation system.
- New contacts and projects with the conventional nuclear industry

Future plans cover continuing the research work on water-hydraulic motion control and force control. The new control method developed for an ITER application will be developed further. New application areas for water hydraulics with improved control will be sought from ITER and the conventional nuclear industry. The knowledge acquired during ITER work will also be applied to more common industries.

The teleoperation of water-hydraulic systems by combining force and model-based principles is under further study, and work for ITER will be continued, at first on developing water-hydraulic tool systems for joining and joint replacing.

Appendix A:

FFUSION Projects and Tasks

The three research areas in the FFUSION programme consists of the following projects:

I Fusion Physics

- Fusion Plasma Engineering (FUS)
- Radio-Frequency Applications of Fusion Plasmas (PLA)

II Fusion Reactor Materials

- First Wall Materials (MAT)
- Ion Beam Studies on Plasma Facing Components (ION)
- Analytical Chemistry of Fusion Materials (ANA)
- Fusion Neutronics (NEU)
- Superconductor Development (SCD)

III Remote Viewing and Handling Systems

- In-Vessel Wieving System (IVVS)
- Water Hydraulic tools for Divertor Refurbishment (HYD)
- Teleoperation Techniques (TEL)
- Remote Manipulation (MAN)

Since 1995, the most of the work carried out in the FFUSION programme consists of the Physics and Technology Tasks of the EU Fusion Programme. The FFUSION projects cover the following Physics and Technology Tasks:

Fusion Plasma Engineering (FUS):

1. Vacuum Window Development for Ion Cyclotron Radio-Frequency Power Transmission Line (NET Task T361)
2. ICRF Vacuum Transmission Line – Dielectric Window Design (NET Contract A7-851CA)
3. Code Development for RF Modules in Transport Codes (JET Task, Tekes TA6)

4. ITER FDR Costing (NET Contract A7-851DT)

Radio-Frequency Applications of Fusion Plasmas (PLA):

1. The Role of Short Wavelength Waves during Heating and Current Drive in the Ion Cyclotron Range Frequencies (JET Task DAMD/Tekes/01)
2. Development and Experimental Evaluation of Theoretical Models in the Field of ICRF Heating (JET Task DAMD/Tekes/01)

First Wall Materials (MAT):

1. Cu/SS Joining Technology (NET Task T212)
2. Cu and Cu-Alloys Irradiation Testing (NET Task T213)
3. Titanium Alloys Irradiation Testing (NET Task BL16.5-2)
4. Detailed Investigation of CuAl25(IG1), it´s Joints with 316LN SS and Joints Testing Procedures (NET Task T213)
5. Aqueous Corrosion of 316L SS and Cu-Based Alloys (NET Task T217)
6. High Energy Beam Welding for Manufacture of Large Tokamak Containment Sectors (NET Task T301/3)
7. Non-Destructive Examination of Primary Wall Small Scale Mock-ups (NET Contract A6-456)

Ion Beam Studies on Plasma Facing Components (ION):

1. Tritium Permeation and Inventory (NET Task T227)
2. Plasma Facing Armour Materials (NET Task T221)

Analytical Chemistry of Fusion Materials (ANA):

1. Evaluation of Erosion / Re-deposition (NET Task T226a)
2. Tritium Permeability, Retention, Wall Conditioning/Clean-Up (NET Task DV7a)

Fusion Neutronics (NEU):

1. Support of Nuclear Analysis (NET Contract A6-404)
2. Nuclear Analysis of Equatorial Heating Ports (NET Contract A6-467)

Superconductor Development (SCD):

1. ITER NbTi Superconducting Wire Development (NET Task M11)
2. ITER Nb_3Sn Superconducting Wire Development (NET Task M2/1)

In-Vessel Wieving System (IVVS):

1. ITER In-Vessel Viewing System – IVVS (NET Task T328)
2. In-Vessel Viewing System (NET Contract A7-851CG)
3. Linear Array IVVS – Design of Prototype Systems and Demo Imaging System (NET Contract A7-851EB)

Water Hydraulic tools for Divertor Refurbishment (HYD):

1. Development of Tooling for Divertor (NET Contract A6-404)
2. Feasibility Study of Divertor Facility (NET Task T232.11)
3. Tools for ITER Divertor Refurbishment Platform (NET Task T308/6)

Socio-Economic and Safety Studies (SERF/SEAFP):

1. SEAFP-2 – Improved Containment Concepts – External Hazards (NET Contract A7-851DJ)
2. Identification and comparative evaluation of environmental impacts of fusion and other possible future energy production technologies

VTT Energy		**FFUSION Research and Industrial Projects in 1993 - 98 (1000 mk)**									28.9.1998	
Seppo Karttunen											Appendix-pro-ffusion.xls	
Project		**Institute/Company**	**Period**	**Tekes**	**Academy**	**VTT**	**HUT/TUT**	**UH**	**Industry**	**Euratom**	**Total**	**Partners**
FFUSION-Research Programme									**IVO/EFET**			
FFUSION co-ordination	001 HAL	VTT ENE	94/98	1 341		30				402	1 773	
Fusion Plasma Engineering	101 FUS	VTT ENE	93/98	5 394		2 183			305	1 771	9 653	IVO (ITER Cost)
RF-applications	102 PLA	HUT TF	93/98		1 991		5 314			1 923	9 228	
RF Vacuum Window	103 ICH	VTT ENE,MAT	96/97	320		200	15		987	178	1 700	IVO, HUT
Fusion Materials	201 MAT	VTT MAT,CHE	93/98	5 550		3 893				2 667	12 110	Hi Speed Tech, OKU
Fusion Neutronics	202 NEU	VTT ENE	94/98	630		404				324	1 358	
Analytical Chemistry	203 ANA	VTT CHE	94/95	250		144					394	DiarcTech, UH
Ion Beam Research	204 ION	UH AL	93/98	1 050		104		2 103		883	4 140	Diarc, VTT CHE
In-Vessel Viewing Sys	301 IVVS	VTT AUT,ELE,HUT	96/98	2 900		820	410		3 439	2 094	9 663	DEMO at IVO
Water Hydraulics	302 HYD	TUT IHA	95/98	2 565			2 764			1 985	7 314	Hytar
Telemanipulation	303 MAN	TUT CON	95	100			75				175	Plustech
Teleoperation	TOP	HUT AUT	95	250			50				300	Plustech
Sosio-economic studies	SERF	VTT ENE	98	200		100				100	400	
FFUSION Research Total				**20 550**	**1 991**	**7 878**	**8 628**	**2 103**	**4 731**	**12 327**	**58 208**	
Industry R&D Projects												
Exposive Welding		High Speed Tech	95/96	182					437	182	801	VTT MAT
Diamond-like Coatings		Diarc	95/98	696					765	569	2 030	VTT MAT, CHE, UH
Water Hydraulics		Hytar	95/98	733					733		1 466	TUT IHA
Superconductors		OKU-SC	96/98	918					1 836	917	3 671	
CATIA Service		PI-Rauma	95/97	221					221		442	
Fusion Safety SEAFP-2		IVO	97							147	147	
NET assignments		Prizztech	95/98	403					403	1 661	2 467	Plustech, PI-Rauma
Industry co-ordination		Prizztech	93/98	2 821					3 236		6 057	VTT ENE
Industry Total				**5 974**	**0**	**0**	**0**	**0**	**7 631**	**3 476**	**17 081**	

Project partners in EU Fusion Programme:

JET Joint Undertaking, UK
NET Team, IPP and FZK, Germany
CEA Cadarache, France

ENEA Frascati, Italy
Risö, Denmark

Appendix B:

Participating Institutes, Companies, Contacts and Research Personnel 1993-1998

B1 Research Institutes and Universities

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Appendix C:

Seminars and Meetings

The following Seminars and Meetings were organized by the FFUSION programme. The ITER Council and ITER Explorer´s´ Meetings were organized in collaboration with the European Commission, DG XII.

- 1st Finnish-Russian Symposium on Fusion Research and Plasma Physics at Helsinki University of Technology, Espoo, May 16-19, 1993.
- Seminar on European Fusion Research and International Fusion Reactor Project - ITER at VTT, Espoo, October 22, 1993.
- 3rd Finnish-Russian Symposium on Fusion Research and Plasma Physics in Sjökulla Kirkkonummi, May 10-11, 1994.
- 1st Annual FFUSION Programme Seminar in Sjökulla Kirkkonummi, September 5, 1994.
- 2nd Annual FFUSION Programme Seminar at Prizztech Oy, Pori, May 24, 1995.
- Meeting of the Co-ordinating Committee for Fast Wave Heating (CCFW) at VTT Energy, Espoo, June 26-27, 1995.
- International Workshop on Copper Magnet Design for the Central Core of Tight Aspect Ratio Tokamaks at Helsinki University of Technology, Espoo, September 19-20, 1995.
- Meeting on the INTAS Project “Application of Frequency-Tunable Microwave Sources for Plasma Heating and Diagnostics at Helsinki University of Technology, Espoo, October 30-31, 1995.

- 3rd Annual FFUSION Programme Seminar at Tampere University of Technology, Tampere, March 7-8, 1996.

- 12th Meeting of the Co-ordinating Committee for Lower Hybrid Heating and Current Drive (CCLH) at VTT Energy, Espoo, June 27-28, 1996.

- 5th Finnish-Russian Symposium on Fusion Research and Plasma Physics at Helsinki University of Technology, Espoo, November 4-5, 1996.

- 4th Annual FFUSION Programme Seminar at M/S Silja Symphony, Helsinki-Stockholm, May 20-21, 1997.

- 12th ITER Council Meeting and 4th ITER Explorers´ Meeting at Hotel Rosendal, Tampere, July 23-23, 1997.

- 5th Annual FFUSION Programme Seminar in Sjökulla Kirkkonummi, June 10, 1998.

- FFUSION Programme 1993-1998 Summary Seminar at Dipoli, Espoo, November 12, 1998

- 7th Finnish-Russian Symposium on Fusion Research and Plasma Physics at Helsinki University of Technology, Espoo, November, 1998.

Appendix D:

Graduate, Licentiate and Doctorate Theses

1. Mikko Alava, "On Mode Conversion to Electrostatic Waves in Ion Cyclotron Range Radiofrequency Heating of Fusion Plasmas", Doctorate Thesis, Acta Polytechnica Scandinavica, Applied Physics Series No. 189, Helsinki 1993, 32 pp + app.

2. Pekka Haussalo, “Study of Hydrogen Trapping at Precipitates in View of Fusion Reactor Materials,” Doctorate Thesis, Acta Polytechnica Scandinavica, Applied Physics Series No. 206, Helsinki 1996, 24 pp. + app.

3. Seppo Sipilä, “Monte Carlo Simulation of Charged Particle Orbits in the Presence of radiofrequency Waves in Tokamak Plasmas”, (1997) Doctorate Thesis, Helsinki University of Technology, Espoo 1997.

4. Seppo Sipilä, "Simulations of Charged Particle Orbits in a Tokamak", Licentiate Thesis, Helsinki University of Technology, Department of Technical Physics, Espoo 1993, 72 pp.

5. Mervi Mantsinen, "Simulations of Burning Tokamak Plasmas", Licentiate Thesis, Helsinki University of Technology, Department of Technical Physics, Espoo 1994, 111 pp.

6. Karin Rantamäki, “An Electrostatic Particle-in-Cell Model for a Lower Hybrid Grill”, Licentiate Thesis, September 1998, Helsinki University of Technology, 71 pp.

7. Mikko Juntunen, "Participation of Finnish Industry and Research in International Fusion Research Programmes", Diploma Thesis, Helsinki University of Technology, Department of Technical Physics, Espoo 1993, 77 pp. (in Finnish)

8. Karin Rantamäki, “Particle-in-Cell Simulations of Lower Hybrid Current Drive,” Helsinki University of Technology, Diploma Thesis, March 15, 1996, 69 pp.

9. Antti Daavittila, “Application of Lasers in Plasma Diagnostics,” Helsinki University of Technology, Diploma Thesis, 1996, 75 pp.

10. Timo Kiviniemi, “3D Fokker–Planck Code for Numerical Simulation of Tokamaks,” Helsinki University of Technology, Diploma Thesis, August 23, 1996, 68 pp.

11. Anssi Laukkanen, “The Effect of Asymmetric Loading on Fracture Toughness of Metallic Materials”, Diploma Thesis, 1997, Espoo, Helsinki University of Technology, 198 p.

12. Mika Pyykkönen, “Pienten koesauvojen säröpituuden mittaus murtumismekaanisissa kokeissa sähköisellä menetelmällä”, Diploma Thesis, 1997, Espoo, Helsinki University of Technology, 75 p. (in Finnish).

13. Samuli Saarelma, “Magnetohydrodynamic Equilibrium in Strongly Shaped Tokamak Plasmas”, Diploma Thesis, 1997, Helsinki University of Technology, 87 p.

14. Tuomas Tala, “Mode Transformation of Lower Hybrid Waves in Tokamaks”, Diploma Thesis, May 1998, Helsinki University of Technology, 61 pp.

Appendix E:

Publications, Reports and Patents 1993-1998

E1 Fusion Physics – Plasma Engineering

1.1 Publications in Scientific Journals – Fusion Physics

1. J.A. Heikkinen, S.K. Sipilä and T.J.H. Pättikangas, "Monte Carlo Simulation of Runaway Electrons in a Toroidal Geometry", Computer Physics Communications **76** (1993) 215–230.

2. J.A. Heikkinen, S.J. Karttunen, T.J.H. Pättikangas and S.K. Sipilä, "Runaway Losses in Current Ramp-Up with Lower Hybrid Waves", Nuclear Fusion **33** (1993) 887–894.

3. J.A. Heikkinen and R.R.E. Salomaa, "Diffusion by Amplitude Modulation in Hamiltonian Systems", Physica D **64** (1993) 365–378.

4. O. Dumbrajs, "Review of the theory of mode competition in gyrotrons" in the Book "Gyrotron Oscillators" Eds. Edgcombe, Taylor and Francis, London 1993, pp. 87–121.

5. J.T. Berndtson, J.A. Heikkinen, S.J. Karttunen, T.J.H. Pättikangas, and R.R.E. Salomaa, "Analysis of Velocity Diffusion of Electrons with Vlasov–Poisson Simulations," Plasma Physics and Controlled Fusion **36** (1994) 57–71.

6. J.A. Heikkinen and T.J.H. Pättikangas, "Effect of a Large-Amplitude Wave on Velocity Distribution of Particles in a Collisional Vlasov Plasma", Physical Review E **50** (1994) 2208–2216.

7. P. Bertrand, A. Ghizzo, S.J. Karttunen, T.J.H. Pättikangas, R.R.E. Salomaa and M. Shoucri, "Generation of Ultrafast Electrons by Simultaneous Stimulated Raman Backward and Forward Scattering", Physical Review E **49** (1994) 5656–5659.

8. O. Dumbrajs "Nonstationary theory of mode competition in gyrotrons with allowance for small inhomogeneity of the guiding magnetic field," Journal of Applied Physics **76** (1994) 5580–5585.

9. O. Dumbrajs and J.A. Heikkinen, "Fast Frequency-Step-Tunable Gyrotrons for Plasma Heating and Fusion Diagnostics", Fusion Technology **26** (1994) 561–565.

10. M.J. Alava, J.A. Heikkinen and T. Hellsten, "On the Origin of Parasitic Modes in Numerical Modeling of Wave Propagation in Plasmas", Journal of Computational Physics **114** (1994) 85–99.

11. M.J. Alava, J.A. Heikkinen, T. Hellsten, I. Pavlov, and O.N. Shcherbinin, "On the Modelling of Fundamental Cyclotron Absorption by Finite Larmor Radius Wave Equations", Physica Scripta **50** (1994) 275–282.

12. M.J. Alava, J.A. Heikkinen, I.P. Pavlov, and O.N. Shcherbinin, "The Effect of Alfvén Resonance on Ion Cyclotron Heating in a Small Aspect Ratio Tokamak", Zh. Tekh. Fiz. **64** (1994) 31–41.

13. S.J. Karttunen, T.J.H. Pättikangas and R.R.E. Salomaa, "Momentum Diffusion in Current Drive by Electron Plasma Waves", Plasma Physics and Controlled Fusion **36** (1994) 657–671.

14. T.K. Kurki-Suonio, M.J. Alava, S.K. Sipilä, and J.A. Heikkinen, "Monte Carlo Simulation of Edge Ion Dynamics in the Presence of Collisions and a Radial Electric Field", Contributions to Plasma Physics **34** (1994) 180–185.

15. G. M. Staebler, F. L. Hinton, J. C. Wiley, R. R. Dominguez, C. M. Greenfield, P. Gohil, T. K. Kurki-Suonio, and T. H. Osborne, "High and very high modes from energy, particle, and momentum transport models", Physics of Plasmas **1** (1994) 909-926.

16. S.K. Sipilä and J.A. Heikkinen, "Monte Carlo Simulation of Lower Hybrid Current Drive in Tokamaks", IEEE Transactions on Plasma Science **22** (1994) 260–266.

17. O. Dumbrajs "Eccentricity of the Electron Beam in a Gyrotron Cavity," International Journal of Infrared and Millimeter Waves **15** (1994) 1255–1262.

18. O. Dumbrajs and M. Thumm, "Gyrotrons for Technological Applications" International Journal of Electronics **76** (1994) 351–364.

19. G.S. Nusinovich, M.E. Read, O. Dumbrajs and K.E. Kreischer, "Theory of Gyrotrons with Coaxial Resonators", IEEE Transactions on Electron Devices **41** (1994) 433–438.

20. J.A. Heikkinen and S.K. Sipilä, “Alpha Particle Transport Driven Current in Tokamaks,” Physical Review E, Rapid Communication **51** (1995) 1655–1658.

21. M.J. Alava, J.A. Heikkinen, and T. Hellsten, “Investigation of the Effect of Alfvén Resonance Absorption on Fast Wave Current Drive in ITER,” Nuclear Fusion **35** (1995) 881–889.

22. I.P. Pavlov and J.A. Heikkinen, “Effect of Antenna Orientation and Plasma Anisotropy on the Directivity of Fast Wave Antenna Radiation,” Physics of Plasmas **2** (1995) 3573–3581.

23. J.A. Heikkinen and S.K. Sipilä, “Power Transfer and Current Generation of Fast Ions with Large-k_θ Waves in Tokamak Plasmas,” Physics of Plasmas **2** (1995) 3724–3733.

24. P. Bertrand, A. Ghizzo, S.J. Karttunen, T.J.H. Pättikangas, R.R.E. Salomaa and M. Shoucri, “Two-stage Electron Acceleration by Simultaneous Stimulated Raman Backward and Forward Scattering,” Physics of Plasmas **2** (1995) 3115–3129.

25. O. Dumbrajs, J.A. Heikkinen, K. Sarparanta, S.K. Sipilä, K. Novik, and A. Piliya, “Diagnostics of Density Fluctuations by Enhanced Scattering with Frequency-Tunable Microwave Power Sources,” International Journal of Infrared and Millimeter Waves **16** (1995) 307–315.

26. P. Gantenbein, E. Borie, O. Dumbrajs and M. Thumm, "Design of a High Order Volume Mode Cavity for a 1 MW/140 GHz Gyrotron", International Journal of Electronics **78** (1995) 771-787.

27. G.S. Nusinovich and O. Dumbrajs, "Two Harmonic Prebunching of Electrons in Multicavity Gyrodevices", Physics of Plasmas **2** (1995) 568-577.

28. G.S. Nusinovich, P. Latham and O. Dumbrajs, "Theory of Relativistic Gyroklystrons”, Physical Review E **52** (1995) 998-1012.

29. O. Dumbrajs and J. Koponen, "Magnetic Field Tapering in the KfK Coaxial Gyrotron", International Journal of Infrared Millimeter Waves **16** (1995) 473-477.

30. O. Dumbrajs, G.S. Nusinovich and B. Levush, "Wave Interaction in Gyrotrons with Off-Axis Electron Beams", Physics of Plasmas **2** (1995) 4621-4630.

31. J.A. Heikkinen and T.P. Kiviniemi, “Ion Cyclotron Minority Ion Heating in the Presence of Lower Hybrid Waves,” Physics Letters **A214** (1996) 53–58.

32. J.A. Heikkinen and S.K. Sipilä, “Current Driven by Lower Hybrid Heating of Thermonuclear Alpha Particles in Tokamak Reactors,” Nuclear Fusion **36** (1996) 1345–1355.

33. J.A. Heikkinen, T.P. Kiviniemi, M.J. Mantsinen, A. Saveliev, L.-G. Eriksson, T.J.H. Pättikangas, S.K. Sipilä and A. Piliya, “Analysis of Fast Minority Ion Distribution and Current Generation for ICRF and LH Heating,” Plasma Physics and Controlled Fusion **38** (1996) 2063–2078.

34. G. Manfredi, M. Shoucri, I. Shkarofsky, A. Ghizzo, P. Bertrand, E. Fijalkow, M. Feix, S. Karttunen, T. Pättikangas and R. Salomaa, “Collisionless Diffusion of Particles and Current across a Magnetic Field in Beam Plasma Interaction,” Fusion Technology **29** (1996) 244–260.

35. J.A. Heikkinen and O. Dumbrajs, “Kinetic Model of Collective Scattering off Fast Ion Generated Electromagnetic Fluctuations in Magnetized Vlasov Plasma,” Physics of Plasmas **3** (1996) 696–698.

36. M.E. Read, G.S. Nusinovich, O. Dumbrajs, G. Bird, J.P. Hogge, K. Kreischer and M. Blank, “Design of a 3 MW 140 GHz Gyrotron with a Coaxial Cavity,” IEEE Transactions on Plasma Science **24** (1996) 586-595.

37. G.S. Nusinovich, B. Levush and O. Dumbrajs, “Optimization of Multistage Harmonic Gyrodevices,” Physics of Plasmas **3** (1996) 3133-3144.

38. G.S. Nusinovich and O. Dumbrajs, “Theory of Gyro-Backward Wave Oscillators with Tapered Magnetic Field and Waveguide Cross Section,” IEEE Transactions on Plasma Science **24** (1996) 620-629.

39. A.T. Dyson, A.E. Dangor, A.K.L. Dymoke-Bradshaw, T. Ashfar-Rad, P. Gibbon, A.R. Bell, C.N. Danson, C.B. Edwards, F. Amiranoff, G. Matthieusent, S.J. Karttunen and R.R.E. Salomaa, “Observations of Relativistic Plasma Waves Excited by a 1.064 μm and 1.053 μm Laser Beat,” Plasma Physics and Controlled Fusion **38** (1996) 505–525.

40. J.A. Heikkinen and S.K. Sipilä, “Monte Carlo Simulation of Minority Ion Bootstrap Current by Off-Axis Ion Cyclotron Heating in Tokamaks”, Nuclear Fusion **37** (1997) 835-849.

41. J.A. Heikkinen, M.A. Irzak and O.N. Shcherbinin, “Modeling of Folded Waveguide Antennas”, Nuclear Fusion **37** (1997) 1477-1495.

42. O. Dumbrajs, “A Multifrequency Gyrotron for Plasma Heating and Diagnostics”, International Journal of Infrared and Millimeter Waves **18** (1997) 2111-2115.

43. J.A. Heikkinen, W. Herrmann and T. Kurki-Suonio, “The Effect of a Radial Electric Field on Ripple-Trapped Ions Observed by Neutral Particle Fluxes”, Physics of Plasmas **4** (1997) 3655-3662.

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