4th IAEA Technical Meeting on

ECRH PHYSICS and TECHNOLOGY for ITER

Vienna, 6 – 8 June, 2007

BOOK of ABSTRACTS
and
PROGRAMME

4th Technical Meeting on ECRH Physics and Technology for ITER

June 6 - 8, 2007, Vienna, Austria

PROGRAMME OVERVIEW

TIME	Wednesday, June 6	TIME	Thursday, June 7	Friday, June 8
09:30 - 09:50	**Welcome:** **A. Malaquias**		**GYROTRONS I** **Chair: M. K. A. Thumm**	**EC WAVE PHYSICS II** **Chair: Giruzzi**
	EC EXPERIMENTS I **Chair: S. Cirant**	09:00 - 09:25	G. G. Denisov	Sergiy Pavlov
09:50 - 10:15	A. Manini	09:25 - 09:50	K. L. Felch	A. K. Ram
10:15 - 10:40	R. J. La Haye	09:50 - 10:15	K. Sakamoto	V. L. Vdovin
10:40 - 11:10	*Break*	10:15 - 10:45	*Break*	*Break*
11:10 - 11:35	J. Bucalossi		**GYROTRONS II** **Chair: R. J. Temkin**	**ITER SYSTEM III** **Chair: Litvak**
11:35 - 12:00	S. Alberti	10:45 - 11:10	B. A. Piosczyk	A. Serikov
12:00 - 13:55	*Lunch*	11:10 - 11:35	S.L. Rao	D. Strauss
	EC EXPERIMENTS I/EC WAVE PHYSICS I **Chair: Saibene**	11:35 - 12:00	V. Erckmann	R. W. Heidinger
13:55 - 14:20	G. Giruzzi	12:00 - 12:25	G. G. Denisov	M. A. Henderson
14:20 - 14:45	N. B. Marushchenko	12:25 - 13:40	*Lunch*	*Lunch*
14:45 - 15:15	*Break*		**ITER EC SYSTEM II** **Chair: W. Kasparek**	**EC EXPERIMENTS II** **Chair: Erckmann**
	ITER EC SYSTEM I **Chair: K. Sakamoto**	13:40 - 14:05	A. Moro	A. Lazaros
15:15 - 15:40	M. A. Henderson	14:05 - 14:30	P. Platania	F. Volpe (presented by R. J. La Haye)
15:40 - 16:05	K. Takahashi	14:30 - 14:55	E. Poli	B. A. Hennen
16:05 - 16:30	W. A. Bongers	14:55 - 15:20	G. Ramponi	D. H. Wagner
16:30 - 16:55	T. P. Goodman	15:20 - 15:50	*Break*	*Break*
16:55	**End of Session**		**TRANSMISSION LINES I** **Chair: A. G. A. Verhoeven**	**SUMMARY SESSION** **Chair: Luce**
17:00	IAC Meeting	15:50 - 16:15	B. M. Plaum	Summaries and Discussion
19:30	Informal Banquet at a Viennese Heuriger near Beethoven's home	16:15 - 16:40	D. A. Rasmussen	
		16:40 - 17:05	R. J. Temkin	
		17:05 - 17:30	W. Kasparek	
		17:30	**End of Session**	**Closing**

TOPICS

A. EC wave physics: current understanding and extrapolation to ITER.

B. Application of EC waves to confinement and stability studies, including active control techniques for ITER.

C. Transmission systems/Launchers: state of the art and ITER relevant techniques.

D. Gyrotron development towards ITER needs.

E. System integration and optimisation for ITER.

INTERNATIONAL ADVISORY COMMITTEE

Bora, D.	India
Cirant, S.	Italy
Erckmann, V.	Germany
Giruzzi, G.	France
Litvak, A.G.	Russia
Luce, T.C. (CHAIR)	USA
Sakamoto, K.	Japan
Thumm, M.	Germany
Tran, M.	Switzerland
Verhoeven, A.G.A.	Netherlands

LOCAL ORGANISER AND SCIENTIFIC SECRETARY

Mr. Artur Malaquias
A.Malaquias@iaea.org
NAPC Physics Section
International Atomic Energy Agency
Office A2309
Wagramer Str. 5, P.O. Box 100, A1400 Vienna, Austria
Tel.: +43 1 2600-21707, FAX: +43 1 26007-21707

Supporting Secretariat:

Ms. Cecilia Devia Torres
C.Devia-Torres@iaea.org
NAPC Physics Section
International Atomic Energy Agency
Office A2374
Wagramer Str. 5, P.O. Box 100, A1400 Vienna, Austria
Tel.: +43 1 2600-21755, FAX: +43 1 26007

Wednesday, June 6

09:30	Opening	Malaquias, A.	IAEA	

EC Experiments I   Chair: S. Cirant

09:50 + 25'	Plasma Stability Enhancement using EC-RH/CD in ASDEX Upgrade	Manini, A.	Max-Planck-Institut für Plasmaphysik, Garching, Germany	7
10:15 + 25'	Requirements for Alignment of ECCD for NTM Stabilization in ITER	La Haye, R. J.	General Atomics, San Diego, CA, USA	8
10:40 + 30'	30' Break			
11:10 + 25'	First experiments of plasma start-up assisted by ECRH on Tore Supra	Bucalossi, J.	Centre d'études de Cadarache, Saint-Paul-lez-Durance, France	9
11:35 + 25'	Experiments with Real-time controlled ECW experiments on the TCV Tokamak	Alberti, S.	Centre de Recherches en Physique des Plasmas, Lausanne, Switzerland	10
12:00 + 115'	115' Lunch			

EC Experiments I/EC Wave Physics I Chair: Saibene

13:55 + 25'	Integrated Modelling of ITER Scenarios with ECCD	Giruzzi, G.	Centre d'études de Cadarache Saint-Paul-lez-Durance, France	11
14:20 + 25'	Electron Cyclotron Current Drive Predictions for ITER: comparison of different models	Marushchenko, G.	Max-Planck-Institut für Plasmaphysik (IPP), Greifswald, Germany	12
14:45 + 30'	30' Break			

ITER EC System I Chair: Sakamoto

15:15 + 25'	Design status of the ITER upper port launcher	Henderson, M. A.	Centre de Recherches en Physique des Plasmas, Lausanne, Switzerland	13
15:40 + 25'	Development of ITER Equatorial EC Launcher for Reliability Improvement	Takahashi, K.	Japan Atomic Energy Agency, Naka, Ibaraki, Japan	14
16:05 + 25'	Recent developments of the Upper port ECH&CD launcher systems for ITER based on the remote steering concept	Bongers, W. A.	FOM Institute for Plasma Physics "Rijnhuizen", The Netherlands	15
16:30 + 25'	Ten Years of Experience in Integrated Control of the Multi-Megawatt ECW system on the TCV Tokamak	Goodman, T. P.	Centre de Recherches en Physique des Plasmas, Lausanne, Switzerland	16
16:55	End of session			
17:00	IAC Meeting			
19:30	Informal Banquet at a Viennese Heuriger near Beethoven's home			

Thursday, June 7

Gyrotrons I Chair: Thumm

09:00 + 25'	Development in Russia of High Power Gyrotrons for Fusion	Denisov, G. G.	Institute of Applied Physics of RAS, Nizhny Novgorod, Russian Federation	17
09:25 + 25'	Operating experience on six, 110 GHz, 1 MW gyrotrons for ECH applications	Felch, K. L.	Communications and Power Industries, Palo Alto, CA, USA	18
09:50 + 25'	Demonstration of 1MW high efficiency oscillation on 170 GHZ CW Gyrotron	Sakamoto, K.	Japan Atomic Energy Agency (JAE), Naka, Ibaraki, Japan	19
10:15 + 30'	30' Break			

Gyrotrons II Chair: Temkin

10:45 + 25'	Status of the 2 MW, 170 GHz Coaxial Cavity Gyrotron for ITER	Piosczyk, B. A.	Forschungszentrum Karlsruhe, Karlsruhe, Germany	20
11:10 + 25'	Gyrotron Source System for ITER Plasma start up	Rao, S. L.	Institute for Plasma Research Near Indira Bridge, Bhat, Gandhinagar, India	21
11:35 + 25'	Advanced Gyrotron Collector Sweeping with smooth power distribution	Erckmann, V.	Max-Planck-Institut für Plasmaphysik (IPP), Greifswald, Germany	22
12:00 + 25'	Window Development at IAP?	Denisov, G. G.	Institute of Applied Physics of RAS, Nizhny Novgorod, Russian Federation	
12:25 + 75'	75' Lunch			

ITER EC System II Chair: Kasparek

13:40 + 25'	Beam characteristics including general astigmatism effects in the Remote Steering ITER ECRH Upper Launcher	Moro, A.	Istituto di Fisica del Plasma, Milano, Italy	23
14:05 + 25'	Numerical calculations of beam patterns for the ITER ECRH Upper Launcher	Platania, P.	Istituto di Fisica del Plasma, Milano, Italy	24
14:30 + 25'	Performance Evaluation of the Remote-Steering Option for the ITER EC Upper Launcher	Poli, E.	Max-Planck-Institut für Plasmaphysik, München, Germany	25
14:55 + 25'	Physics analysis of the ITER ECW system for an optimized performance	Ramponi, G.	Istituto di Fisica del Plasma, Milano, Italy	26
15:20 + 30'	30' Break			

Transmission Lines I Chair: Verhoeven

15:50 + 25'	Optimum Corrugations for Low-Loss Square and Cylindrical Waveguides	Plaum, B. M.	Institut für Plasmaforschung der Universität Stuttgart, Stuttgart, Germany	27
16:15 + 25'	Design of the ITER Electron Heating and Current Drive Waveguide Transmission Line	Rasmussen, D. A.	US ITER Project Office Oak Ridge National Laboratory Oak Ridge, TN, USA	28
16:40 + 25'	Estimation of the loss in the ECH Transmission lines for ITER	Temkin, R. J.	Plasma Science and Fusion Center, MIT Building, Cambridge, MA, USA	29
17:05 + 25'	FADIS, a fast switch and combiner for high-power millimeter wave beams.	Kasparek, W.	Institut für Plasmaforschung der Universität Stuttgart, Stuttgart, Germany	30
17:30	End of session			

Friday, June 8

EC Wave Physics II Chair: Giruzzi

09:00 + 25’	A general method for relativistic plasma dielectric tensor evaluation	Pavlov, Sergiy	Institute of Plasma Physics, National Sciences Center "Kharkov", Institute of Physics and Technology, Kharkov, Ukraine	31
09:25 + 25’	Relativistic Effects in Electron Cyclotron Resonance Heating and Current Drive	Ram, A. K.	Plasma Science and Fusion Center, MIT Building, Cambridge, MA, USA	32
09:50 + 25’	Electron cyclotron heating modelling in large tokamaks and ITER with 3D full wave code	Vdovin, V. L.	RRC Kurchatov Institute, Nuclear Fusion Institute, Moscow, Russian Federation	33
10:15 + 30’		30’ Break		

ITER System III Chair: Litvak

10:45 + 25’	Nuclear analyses for the ITER ECRH launcher	Serikov, A.	Forschungszentrum Karlsruhe Association, Karlsruhe, Germany	34
11:10 + 25’	Thermal and electromagnetic study of the UPP for the ECRH in ITER	Strauss, D.	Forschungszentrum Karlsruhe Association, Karlsruhe, Germany	35
11:35 + 25’	Structural system of the ECH Upper Port Plug for ITER	Heidinger, R. W.	Forschungszentrum Karlsruhe Association, Karlsruhe, Germany	36
12:00 + 25’	Interface issues associated with the ITER ECH system	Henderson, M. A.	Centre de Recherches en Physique des Plasmas, Lausanne, Switzerland	37
12:25 + 65’		65’ Lunch		

EC Experiments II Chair: Erckmann

13:30 + 25’	The advantage of early application of electron cyclotron waves for the suppression of tearing modes: Assessment for ASDEX Upgrade and results from TEXTOR	Lazaros, A.	School of Electrical and Computer Engineering, Athens, Greece	38
13:55 + 25’	Locked Neoclassical Tearing Mode Control on DIII-D by ECCD and Magnetic Perturbations	Volpe, F. (presented by La Haye, R. J.)	General Atomics, San Diego, CA USA	39
14:20 + 25’	Modeling and control for fusion plasma stabilization by means of a mechanical ECRH launcher at TEXTOR	Hennen, B. A.	Eindhoven University of Technology, Eindhoven, Netherlands	40
14:45	Status of the New Multi-Frequency ECRH System for ASDEX Upgrade	Wagner, D. H.	Max-Planck-Institut für Plasmaphysik, Garching, Germany	41
15:10 + 30’		30’ Break		

Summary Session Chair: Luce

15:40 + 110’	Summaries and Discussion
17:30	End of session

ABSTRACTS

Plasma Stability Enhancement Using EC-RH/CD in ASDEX Upgrade

A. Manini, G. Gantenbein*, N. Hicks, F. Leuterer, M. Maraschek, R. Neu, J. Stober, W. Suttrop, D. Wagner, H. Zohm and the ASDEX Upgrade Team
Max-Planck-Institut für Plasmaphysik, EURATOM Association, Boltzmannstrasse 2, 85748 Garching, Germany
**Forschungszentrum Karlsruhe, PO Box 3640, 76021 Karlsruhe, Germany*

To increase both stability and performance of fusion plasmas, it is very important to be able to actively control its magneto-hydro-dynamic (MHD) stability, in particular sawteeth and neoclassical tearing modes (NTMs). Electron cyclotron resonance heating and current drive (EC-RH/CD) is an optimal tool due to the high steering flexibility and its very localised power deposition (P_{EC}) and driven current density (j_{CD}) profiles. Recently, in ASDEX Upgrade (AUG), a significant effort has been made to optimise such control using the (old) system ECRH-1. The modification of stability and behaviour of both sawteeth and NTMs with local EC-RH/CD are presented. The sawtooth stability depends on the local shear at q=1 and it can be modified by local co-/ctr-ECCD around q=1. The effects on the sawtooth period between wide and narrow CD widths are compared, showing that strongest effects are obtained with narrow co-ECCD. It is also shown that the effects of wide ctr-ECCD are dominated by the P_{EC} contribution, while the ones with narrow deposition by the j_{CD} contribution. Sawteeth can trigger NTMs, therefore, by controlling sawteeth, the excitation of NTMs can be influenced as well. Once NTMs are triggered, they can be fully stabilised with co-ECCD at the resonant surface. Detailed experiments on the optimisation of NTM suppression are presented. It is shown that the suppression efficiency is enhanced by either reducing the CD width (d), or, if the CD width is larger than the island size ($2d>w$), by modulating the ECCD power in phase with the O-point of the rotating island. These results are crucial for large next-step fusion devices, such as ITER, where $2d>w$ is expected to be unavoidable during NTM suppression. In 2007, AUG will operate with the first wall fully covered with tungsten. Under these conditions, it has been seen that certain scenarios are particularly subject to impurity accumulation in the plasma core. Since adding central heating reduces this accumulation, the combination of the ECRH-1 and of the (new) ECRH-2 systems, together providing about 2.5MW of pure electron heating, should significantly enlarge the AUG operational space. Finally, a real-time feedback system capable of steering the EC-RH/CD deposition during a discharge is presently being developed. The current status of the project will be outlined.

Requirements for Alignment of ECCD for NTM Stabilization in ITER*

R.J. La Haye, D.A. Humphreys, R. Prater, E.J. Strait and A.S. Welander

General Atomics, P.O. Box 85608, San Diego, California 92186-5608 USA

email: lahaye@fusion.gat.com, Phone: (858) 455-3134, Fax: (858) 455-4156

A recent change in the electron-cyclotron current drive (ECCD) launcher scheme in ITER from "remote" to "front" steering has narrowed the expected ECCD considerably [1], making the stabilization of neoclassical tearing modes (NTMs) — with or without modulation of the ECCD — much more certain [2]. Evaluation of the required EC power for either the m/n=3/2 or 2/1 modes, assuming perfect alignment of the peak ECCD on the rational surface in question, indicates that the proposed 20 MW is adequate [3]. However, the narrower ECCD makes the alignment more challenging.

Experiment shows that the marginal island w_{marg} for NTM stabilization is about twice the ion banana width [3], and is only 1–2 cm in ITER. With front steering, the ECCD is still relatively broad, with current deposition full width half maximum $\delta_{ec} \sim (5/3)\ w_{marg}$. This places strict requirements on ECCD alignment with the cw ECCD effectiveness dropping to zero for misalignments as small as ~ 2 cm.

DIII-D ECCD alignment techniques have proceeded from: (1) adjusting the fixed toroidal field BT in 0.01 T steps from shot to shot to maximize the initial island decay rate, (2) applying "search and suppress" real-time control to find and lock onto optimum alignment (adjusting BT or shifting the plasma major radius in equivalent 1 cm steps), and (3) in the absence of an island (and/or after completely suppressing an island) using real-time EFIT MHD reconstructions with the motional Stark effect diagnostic to accurately locate and adjust the location of the rational surface <u>and</u> to correct the current drive location for refraction. The latter method allowed DIII-D to run higher stable beta without either a 3/2 or a 2/1 mode ever occurring.

The DIII-D alignment results are used to confirm models for the effect of misalignment on the ECCD effectiveness and are then applied to ITER. Tolerances for misalignment will be presented to establish criteria for both the alignment (by moving mirrors) in the presence of an island, and for the accuracy of real-time ITER MHD equilibrium reconstruction.

[1] M.A. Henderson, 21st IAEA Fusion Energy Conf., Chengdu, China, 2006, IT/P2-15.
[2] R.J. La Haye, et al., 21st IAEA Fusion Energy Conf., Chengdu, China, 2006, EX/P8-12.
[3] R.J. La Haye, et al., Nucl. Fusion **46**, 451 (2006).

---

*Work supported by the U.S. Department of Energy under DE-FC02-04ER54698.

First Experiments of Plasma Start-up Assisted by ECRH on Tore Supra

J. Bucalossi, E. Trier, P. Hertout, M. Lennholm and F. Saint-Laurent

Association EURATOM-CEA, CEA/DSM/DRFC, CEA Cadarache

13108 St Paul Lez Durance, France

ECRH pre-ionisation and assisted start-up will be necessary in ITER due to the low electric field available (≤0.3 V/m). Even if this method has been demonstrated in a large number of tokamaks and is extensively used in stellerators, further experiments could give useful input to start-up simulations for ITER. This paper reports on the results recently obtained on ECRH assisted start-up on Tore Supra and discusses the experiments which are planned for the forthcoming campaigns.

The tokamak Tore Supra is a superconducting device (R = 2.4 m, a=0.72 m) which has been designed to address the physics and technology of long duration plasma discharges. The additional heating is exclusively provided by RF systems (9 MW IRCH, 4 MW LHCD, and 0.8 MW ECRH). The ECRH/ECCD system is based on two 118 GHz gyrotrons. The power is injected into the plasma as Gaussian beams by an antenna located on the low field side which, using actively cooled mirrors inside the Tore Supra vacuum vessel, allows extensive control of both poloidal and toroidal injection angles.

During the last experimental campaign, the ECRH system has been used for the first time to assist the plasma start-up. The usual start-up voltage on Tore Supra is 25 V (1.7 V/m) while the toroidal field in the centre of the vessel is 3.8 T and the deuterium prefill pressure is around 10^{-2} Pa. Due to the pre-magnetisation of the iron core, the minimum stray field value in the vacuum vessel is in the 5 mT range, which leads to toroidal connection lengths lower than 300 m. In these conditions, attempts have been made to lower the start-up voltage. Sustained breakdowns were obtained up to ~0.45 V/m without external assistance. Besides, 350 kW of ERCH have been applied during 50 ms (1 gyrotron in O-mode) at an intermediate voltage (0.8 V/m). Pre-ionisation was successfully achieved and ~30 kA of current were established in 10 ms. The plasma started on the outboard limiter (R~3 m), while the EC resonance stood in the high field side region (R~2.18 m). Note that, in the same condition and without ECRH the plasma start-up is delayed by ~30 ms.

Next experiments will be devoted to lower voltage studies (<0.45 V/m), conditions where ionisation is not obtained without ECRH, to variation of the magnetic null quality (lower pre-magnetisation and without pre-magnetisation) and null location, and to variation of the prefill pressure. Finally, experiment at reduced magnetic field (~2T) with second harmonic resonance inside the vessel and X-mode waves will be attempted.

Experiments With Real-time Controlled ECW on the TCV Tokamak

S. Alberti[1], G. Arnoux[2], J. Berrino[1], Y.Camenen[1], S. Coda[1], B.P. Duval[1], T.P. Goodman[1], J-M. Moret[1], J.I. Paley[1], L. Porte[1] and TCV team
[1]*Ecole Polytechnique Fédérale de Lausanne (EPFL), Centre de Recherches en Physique des Plasmas, Association EURATOM-Confédération Suisse, CH-1015 Lausanne, Switzerland*
[2]*Départment de Recherches sur la Fusion Contrôlée, Association EURATOM-CEA, CEA/Cadarache, 13108 Saint Paul-lez-Durance Cédex, France*

e-mail contact of the first author: stefano.alberti@epfl.ch

The ECW system on the TCV tokamak consists of six gyrotrons (82.7GHz/0.5MW/2s) used for X2-ECH/ECCD and three gyrotrons (118GHz/0.5MW/2s) used for X3-ECH in a top-launch configuration. The X3 system broadens the operational space on TCV with the possibility of heating plasmas at high density, well above the cutoff density of the X2 system (X2 cutoff at $n_e = 4.2\ 10^{19}m^{-3}$). From its inception the ECW system has been designed to allow real-time control (RTC) on a variety of actuators such as the inclination of the last mirror on each launcher and/or the injected RF power. Some of these RTC actuators of the ECW system have been used with the TCV hybrid (digitally controlled) analogue controller based on matrix multiplication of signals and PID controllers.

In this paper three different experiments using the RTC capabilities of the ECW system are reported. The first experiment is related to the RTC of the X3 mirror angle in the top-launch configuration aimed at maximising the single-pass absorption of the X3 wave. Secondly, using X2-ECCD in a fully non-inductive current scenario with no external loop voltage a feedback control loop has been successfully implemented to control the plasma current in real time using the ECW power actuator. Finally, using a real time plasma elongation observer and both the ECW power and the deposition radius (via the mirror angles) as actuators, the plasma elongation has been varied in a controlled way without any change in the shaping magnetic fields; in particular, this has allowed the stable sustainment of highly-elongated, low-current plasmas that are vertically unstable in Ohmic conditions. An Advanced Digital Plasma Control System replacing the hybrid analogue controller is presently being installed and commissioned on TCV. The main features of this system will be outlined.

Integrated Modelling of ITER Scenarios with ECCD

G. Giruzzi, J.F. Artaud, V. Basiuk, J. Garcia, F. Imbeaux, M. Schneider

Association Euratom-CEA sur la Fusion, CEA/DSM/DRFC, CEA/Cadarache,
13108 St. Paul-lez-Durance, France

The main application of EC waves in ITER will be the control of NTM by means of the top launchers. However, virtually every ITER scenario would profit from appropriate use of ECCD by means of the equatorial launcher. In particular, hybrid and steady-state scenarios require off-axis current drive in order to either keep the safety factor above 1 or produce negative shear in a large part of the plasma cross-section. In this type of applications, alignment of the current sources and self-consistency of current and temperature profiles are critical issues, which can only be addressed by integrated modelling. To this end, the CRONOS suite of codes has been applied to the simulation of these scenarios. CRONOS integrates, in a modular structure, general 2-D magnetic equilibria, radiation and particle losses, several heat, particle and impurities transport models, as well as heat, particle and momentum source modules, associated, e.g., with neutral beams, radio-frequency waves, pellet ablation, etc. The ECCD module includes toroidal ray-tracing and linear computation of the driven current, which is generally adequate for ITER parameters.

This presentation will give a short description of the CRONOS suite of codes, followed by results of simulations of ITER hybrid scenarios assisted by ECCD. Use of ECCD in steady-state scenarios will also be addressed.

Electron Cyclotron Current Drive Predictions for ITER: Comparison of Different Models

N.B. Marushchenko[†], H. Maassberg, C.D. Beidler, Yu. Turkin

Max-Plank-Institut für Plasmaphysik, EURATOM-Association, Greifswald, Germany

Due to its high localization and operational flexibility, Electron Cyclotron Current Drive (ECCD) is envisaged for stabilizing the Neoclassical Tearing Mode (NTM) in tokamaks and correcting the rotational transform profile in stellarators. While the spatial location of the electron cyclotron resonant interaction is usually calculated by the ray-tracing technique, numerical tools for calculating the ECCD efficiency are not so common. Two different methods are often applied: i) direct calculation by Fokker-Planck modelling, and ii) by the adjoint approach technique. In the present report we analyze and compare different models used in the adjoint approach technique from the point of view of ITER applications.

The numerical tools for calculating the ECCD efficiency developed to date do not completely cover the range of collisional regimes for the electrons involved in the current drive. Only two opposite limits are well developed, collisional and collisionless. Nevertheless, for the densities and temperatures expected for ECCD application in ITER, the collisionless limit model (with trapped particles taken into account) is quite suitable. We analyze the requisite ECCD scenarios with help of the new ray tracing code TRAVIS with the adjoint approach implemented. The (adjoint) Green's function applied for the current drive calculations is formulated with momentum conservation taken into account; this is especially important and even crucial for scenarios, in which mainly bulk electrons are responsible for absorption of the RF power. For comparison, the most common "high speed limit" model in which the collision operator neglects the integral part and which is approximated by terms valid only for the tail electrons, produces an ECCD efficiency which is an underestimate for some cases by a factor of about 2. In order to select the appropriate model, a rough criterion of "high speed limit" model applicability is formulated. The results are verified also by calculations with help of a bounce-averaged Fokker-Planck code (here, bounce averaging is equivalent to the collisionless approach) with the linearized collision operator, i.e. with momentum conservation taken into account.

[†] e-mail: nikolai.marushchenko@ipp.mpg.de

Design Status of the ITER Upper Port Launcher

M.A. Henderson[1], R. Chavan[1], R. Bertizzolo[1], A. Bruschi[2], D. Campbell[3], S. Cirant[2], A. Collazos[1], I. Danilov[4], F. Dolizy[1], J. Duron[1], D. Farina[2], K. Kleefeldt[4], R. Heidinger[4], J.-D. Landis[1], E. Poli[5], G. Ramponi[2], G. Saibene[3], F. Sanchez[1], O. Sauter[1], A. Serikov[4], H. Shidara[1], P. Spaeh[4], V.S. Udintsev[1], H. Zohm[5]

[1] CRPP, EURATOM – Confédération Suisse, EPFL, CH-1015 Lausanne Switzerland
[2] Istituto di Fisica del Plasma, EURATOM- ENEA- CNR, 20125 Milano, Italy
[3] EFDA Close Support Unit, Boltzmannstrasse 2, D-85748 Garching, Germany
[4] Forschungszentrum Karlsruhe, EURATOM-FZK, D-76021 Karlsruhe, Germany
[5] Max Planck-Institute für Plasmaphysik, EURATOM-IPP, D-85748 Garching, Germany

First Author e-mail: mark.henderson@epfl.ch

The purpose of the ITER ECRH upper port antenna (or launcher) will be to drive current locally to stabilise the NTMs (depositing ECCD inside of the island which forms on the q=3/2 or 2 rational magnetic flux surfaces) and control the sawtooth instability (deposit ECCD near the q=1 surface). The launcher should be capable of steering the focused beam deposition location across the resonant flux surface over the range in which the q=1, 3/2 and 2 surfaces are expected to be found, for the various plasma equilibria susceptible to the onset of NTMs and sawteeth. ITER's present reference design uses a front steering (FS) concept, with the moveable mirror close to the plasma. Two separate mirrors are used to decouple the focussing and steering aspects resulting in an optimized optical configuration providing a well focused beam over a large steering range. The launcher is capable of steering eight 2MW beams in all of the four allocated upper port plugs.

The critical component of the FS launcher is the steering mechanism, which will be a frictionless and backlash free mechanical system based on the compliant deformation of structural components to avoid the in vessel tribological difficulties. An inert gas pressure controlled bellows system provides accurate angular positioning of the steering mirror. The entire launcher (mm-wave components) can be designed fail-safe in that if a given subsystem fails, it can be isolated and ITER can continue operation. In addition, in-situ leak testing of critical components (steering mechanism, cooling, diamond window, etc.) is envisioned to insure proper functioning and avoidance of disrupting ITER operation. Details of the FS launcher design relating to physics performance, mm-wave optical design, thermohydraulic, electromagnetic and integration of launcher components into the port plug will be discussed.

Development of ITER Equatorial EC Launcher for Reliability Improvement

K. Takahashi, K. Noriyuki, K. Kajiwara, A. Kasugai and K. Sakamoto
Japan Atomic Energy Agency, Naka, Ibaraki, 311-0193 Japan
e-mail : takahashi.koji@jaea.go.jp

Establishing the conceptual design of the ITER equatorial EC launcher, design studies and development of the launcher components such as a front shield, a steering mirror, a torus window and etc… were carried out. For the next stage, the practical and reliable design of the launcher is essential. In order to achieve the task, the trial fabrication of the component, such as a steering mirror, drive mechanism for the mirror and waveguide components, high power RF experiment of a dog-leg structure transmission line and the windows and so on has been carried out. The steering mirror prototype including the spiral cooling tubes was fabricated under the present design. Hot isostatic press (HIP) technique for the bonding between the Cu-alloy mirror body and the stainless steel (SS) cooling tubes inside was applied. The test of water flow was so far carried out and the expected flow rate was successfully obtained. The test of the drive mechanism for the mirror was also carried out under 100ºC and vacuum, the relevant ITER circumstance. It was found that the LM actuator was stuck due to the unexpected load in the link structure. Then, the link has been removed and the actuator structure has been modified. The SS corrugated waveguide prototype for the launcher that had several cooling holes along the waveguide axis was fabricated. The corrugation surface was coated by Cu so as to minimize the ohmic loss at the inner surface. The RF test with 0.7MW was carried out. The cooling water was saturated and the loss was approximately estimated as 0.34kW that corresponds to the surface heat load of 1.8kW/m^2 at 1MW transmission. It is recognized that 1MW transmission is possible in the SS/Cu waveguide. The further analytical studies of the launcher structure have also been carried out. The quasi-optical (QO) transmission layout has been considered instead of the waveguide in the present design. The advantage of this option is possibly reducing the cost. According to the preliminary analysis of beam propagation, it is verified that transmission loss at the free space region is about 3%, which is acceptable level. The electromagnetic (EM) analyses of the front shield and the steering mirror have been progressed. It was found that the size reduction or some slits were necessary for the largest shield module to withstand the EM force.

Recent Developments of the Upper Port ECH&CD Launcher Systems for ITER Based on the Remote Steering Concept

W.A. Bongers[1], M.F. Graswinckel[1], M.A. van den Berg[1], A. Bruschi[3], I. Danilov[4], A.J.H Donné[1], B.S.Q. Elzendoorn[1], A.P.H. Goede[1], R. Heidinger[4], M.A. Henderson[6], O.G. Kruijt[1], B. Kruizinga[2], N. Lopes Cardozo[1], A. Moro[3], E. Poli[5], D.M.S. Ronden[1] and A.G.A. Verhoeven[1]

[1]*FOM-Institute for Plasma Physics Rijnhuizen, Association EURATOM-FOM, Nieuwegein, The Netherlands, http://www.rijnhuizen.nl/,* [2]*TNO, Delft, The Netherlands*
[3]*Istituto di Fisica del Plasma, CNR, Ass. EURATOM-ENEA-CNR, Milano, Italy*[4]
[4]*FZK, Karlsruhe, Germany,* [5]*IPP, Garching,* [6]*CRPP, Lausanne*

Abstract. ITER employs Electron Cyclotron Waves (ECW) for heating of the plasma and to stabilize MHD modes including the neoclassical tearing modes (NTM). To cover all the significant ITER plasma scenarios, the upper port ECW launcher needs a beam steering mechanism to reach the magnetic q-surfaces where NTM instabilities can grow. Two alternative approaches are being studied in Europe under EFDA tasks: Front Steering (FS), which has rotating mirrors situated near the plasma, and Remote Steering (RS), which has the mirror steering mechanism situated in the secondary vacuum. The steering mirrors in the RS system project the mm-wave beam onto a Square Corrugated Waveguide (SCW) of a length such that the input beam profile is imaged at the output, producing a steered output beam. The advantage of RS is that the steering mechanism is shielded from direct plasma exposure and neutron bombardment and it also allows simpler maintenance. One of the drawbacks of RS is that the output steering range is inversely coupled to the focusing. The optimization of the quasi-optical system is done for a short (4.3 m) and a long (9.1 m) SCW. The results are evaluated for their effectiveness in NTM stabilization by calculating the parameter η which represents the ratio between the EC wave driven current and the bootstrap current, preferably η should exceed 1.2. The performance is also evaluated in terms of beam focusing properties and power loading on the mirrors. The η achieved so far meets the requirements in the ITER reference scenarios for q=2/1 and partly for q=3/2. Additional tools under investigation for optimization are: non spherical mirror curvatures, tapered SCWs and the use of general astigmatic beams.

This work has been carried out as part of the EFDA work program Task TW6-TPHE-ECHULB for the development of the ITER ECRH Upper Port Launcher system, in collaboration with the Euratom Associations IFP Stuttgart, IPP Garching, FZK, IFP/CNR and CRPP.

Ten Years of Experience in Integrated Control of the Multi-Megawatt ECW System on the TCV Tokamak

T.P. Goodman, S. Alberti, S. Coda, B.P. Duval, D. Fasel, M.A. Henderson, J-P. Hogge, P-F. Isoz, P. Lavanchy, J. Lister, B. Marletaz, Ph. Marmillod, J-M. Moret, L. Porte and TCV team
Ecole Polytechnique Fédérale de Lausanne (EPFL), Centre de Recherches en Physique des Plasmas, Association Euratom-Confédération Suisse, CH-1015 Lausanne, Switzerland

e-mail contact of the first author: timothy.goodman@epfl.ch

The ECW system on the TCV tokamak consists of six gyrotrons (82.7GHz/0.5MW/2s) used for X2 and electron Bernstein wave (EBW) ECH/ECCD with individual low-field-side launchers. Three additional gyrotrons (118GHz/0.5MW/2s) are used for X3-ECH in a top-launch configuration to provide central heating of high-density plasmas, at nearly 3 times the cutoff density of X2 (cutoff at n_e = 4.2 $10^{19}m^{-3}$). The first two X2 gyrotrons were put into operation in April and November of 1996 with one more following in June, 1997. The full complement of 6 X2 gyrotrons operated simultaneously in TCV in December, 1999. With the acceptance of the third X3 in October, 2003, the full power of 9 gyrotrons (4.2MW injected) was available for experiments. From its inception the ECW system has been designed to allow the highest possible degree of automation, integration and flexibility in the experimental program. Each of the X2 and X3 subsystems is routinely individually operated by one person. Only a modest integration effort would be required to permit 9 gyrotron operation by one operator. This high level of automation in the system allows the operator to adapt to changes in the experimental program "on the fly" while maintaining a high success rate. An added benefit is that it allows the physicists to bring their expertise in EC-wave physics to bear on the day-to-day experimental programs to which they would otherwise be associated purely as gyrotron operators.

This paper will present an objective evaluation of the use of the most powerful and flexible EC system in operation today. In it we stress that the modularity of the system, the universality of the plant control, the ease with which the control can be modified and improved, and the close collaboration between technicians, engineers and physicists have each been crucial to the successes of over a decade of reliable, adaptable and user-friendly multi-megawatt EC operation on TCV. The TCV ECW system therefore appears to be an appropriate model upon which future installations, such as that of ITER, can be based.

Development in Russia of High Power Gyrotrons for Fusion

G,G,Denisov[1], A.G.Litvak[1], V.E.Myasnikov[2], E.M.Tai[2], V.I.Ilin[3].

[1] *Institute of Applied Physics Russian Academy of Sciences,*
46 Ulyanov Street, Nizhny Novgorod, 603950, Russia
[2] *GYCOM Ltd., 46 Ulyanov Street, Nizhny Novgorod, 603155, Russia,*
[3] *Russian Research Centre "Kurchatov Institute", Kurchatov square, 1, Moscow, 123182, Russia*

den@appl.sci-nnov.ru ; tel.: +7 8312 369920; fax.: +7 8312 362061

Electron cyclotron systems of fusion installations are based on powerful millimetre wave sources – gyrotrons, which are capable to produce now microwave power up to 1 MW in very long (hundreds of seconds) pulses. The paper presents the latest achievements in development at IAP/GYCOM of MW power level gyrotrons for fusion installations. Among them are a new versions of 170 GHz gyrotron for ITER and multi-frequency (105-140 GHz) gyrotron for Asdex-Up. The gyrotrons are equipped with diamond CVD windows and depressed collectors.

The most efforts were spent for development of ITER gyrotron. The tests are carried out at specially prepared test stand in Kurchatov Institute. The following gyrotron output parameters were demonstrated so far in many pulses: 1MW/30 sec and 0.64 MW/300 sec. Also a gyrotron with a higher power -1.5 MW was designed and tested in short pulses. The tests continue.

In two tested long-pulse dual-frequency gyrotrons, power in the output Gaussian beam exceeding 0.9MW at 140GHz and 0.7MW at 105GHz was attained at specified 10-s pulse duration. The multi-frequency gyrotron should operate at least at four frequencies in the frequency range 105GHz-140 GHz. Two window concepts for the gyrotron are considered: Brewster window and two-disc adjustable window.

Last years significant efforts were done by IAP/GYCOM in order to solve the whole scope of problems associated with the use of CVD diamond windows in gyrotrons: growing of discs, their cutting and polishing, and then high-temperature brazing and mounting to a tube. Two setups for growing diamond discs have been put into operation. The first discs grown at IAP have acceptable mechanical and electrical parameters. The IAP/GYCOM discs have been successfully brazed at near 800°C temperature to metal constructions and tested with high-power gyrotrons.

Operating Experience on Six, 110 GHz, 1 MW Gyrotrons for ECH Applications

K. Felch, M. Blank, P. Borchard, S. Cauffman, P. Cahalan, and H. Jory
Communications and Power Industries, 607 Hansen Way, Palo Alto, CA 94304 USA
J. Lohr and Y. Gorelov
General Atomics, P.O. Box 85608, San Diego, CA 92186 USA

Since 1999, CPI has shipped six, 110 GHz, 1 MW, 10-s pulsed gyrotrons to General Atomics for use in electron cyclotron heating (ECH) experiments on the DIII-D tokamak. As a result of extensive testing of these gyrotrons at CPI and General Atomics, a wealth of information has been obtained regarding the long-term performance of the original electrical and mechanical design of the tubes. This paper will summarize the operating experience on the different gyrotrons and give observations relating to each of the main design features.

The key design areas of the gyrotron are the electron gun, interaction cavity, output coupler, electron beam collector and output window. In the electron gun, no high-voltage arcing or current emission problems have been observed. Regarding the interaction cavity, all of the gyrotrons have achieved 1 MW at the nominal 80 kV, 40 A beam parameters except one, which was somewhat low in power. Whether this deficiency was due to a cavity or an electron beam problem has not been determined. There have been no cavity failures due to cyclic fatigue or overheating. For the output coupler, losses due to diffraction effects and other inefficiencies in the transformation from the cavity mode to the Gaussian beam have been about 6-7%. The efficiency of coupling the output power from the gyrotron into an HE_{11} mode in 31.75-mm-diameter corrugated waveguide is 93-95%. All three of the first set of gyrotrons sustained damage to the electron beam collector after prolonged operation. Cyclic fatigue damage due to sweeping the beam axially in the collector by modulating the collector magnet coil was the primary cause of the collector difficulties. To alleviate these problems, sweep rates were increased from 4 Hz to 5 Hz and maximum power densities were lowered by increasing the amplitude of the sweep and by raising the position of the spent beam in the collector. Two window failures have been experienced on the 110 GHz gyrotrons. One of the failures was caused by corrosion of a low-temperature, aluminum-based braze. Following the corrosion problem, all subsequent windows were brazed using a high-temperature braze process that was resistant to corrosion in water. However, one of the first windows using the high-temperature braze failed due to the presence of a lossy surface layer that was apparently deposited during the braze process. Improvements to the braze process and vigilant checking of the window loss properties at each step of the fabrication process have eliminated any further difficulties with the window.

Demonstration of 1MW High Efficiency Oscillation on 170 GHz CW Gyrotron

K. Sakamoto, A. Kasugai , K. Kajiwara, K. Takahashi and K. Noriyuki
Japan Atomic Energy Agency, Naka, Ibaraki, 311-0193 Japan
e-mail : sakamoto.keishi@jaea.go.jp

On the EC H&CD system of ITER, the demonstration of 1 MW 170 GHz CW gyrotron has been the most critical issue, and the extensive developments have been carried out at many institute and companies. Here, we report the first stable oscillation of 1 MW at 170 GHz gyrotron that is applicable for EC H&CD system on ITER. The oscillation mode is TE31,8 with a cylindrical cavity, and is converted to the Gaussian line beam using a high efficiency mode converter of ~97.5 %. With a beam voltage of 72 kV and a beam current of 38.5 A, the output of greater than 1MW was obtained for 800 sec. As the depressed collector voltage is ~25kV, the efficiency was 55 %, which exceeds 50 % which is the designed value of the EC H&D system. The high efficiency oscillation was established in the so called “hard self-excitation region”. The measured stray radiation in the gyrotron was 2 % of the output power. Up to now, the 1 hour operation was also obtained at 0. 6MW output power, which corresponds to 2.15 GJ of the output energy. During the operation the vacuum in the gyrotron was kept at ~10^{-6} Pa and that decreased after 40 min., which shows the possibility of full-CW operation. The output power couples with HE11 mode using a matching optics unit with the efficiency of 96 %, and 95 % of the output power was transmitted to the dummy load after 7 m transmission with the corrugated waveguide with two miter bends. This coupling efficiency satisfies the ITER criteria. As a conclusion, the result assure to provide EC H&CD system for ITER.

Status of the 2 MW, 170 GHz Coaxial Cavity Gyrotron for ITER

B. Piosczyk[1], F. Albajar[2], S. Alberti[3], P. Benin[4], W. Bin[5], T. Bonicelli[2], S. Cirant[5], G. Dammertz[1] O. Dumbrajs[6], D. Fasel[3], J. Flamm[7], G. Gantenbein[1], E. Giguet[4], T. Goodman[3], J.P. Hogge[3], S. Illy[1], J. Jin[1], C. Lievin[3], O. Prinz[1], T. Rzesnicki[1], M. Schmid[1], M. Thumm[1,7], M.Q. Tran[3]

[1]Forschungszentrum Karlsruhe (FZK), Association EURATOM-FZK, IHM, Postfach 3640, D-76021 Karlsruhe, Germany,
[2]European Fusion Development Agreement (EFDA), D-85748 Garching, Germany
[3]Centre de Recherche en Physique des Plasmas (CRPP), Association Euratom-Confédération Suisse, EPFL Ecublens, CH-1015 Lausanne, Suisse
[4]Thales Electron Devices (TED), F-78141 Vélizy-Villacoublay, France
[5]Instituto di Fisica del Plasma, EURATOM-ENEA-CNR Association, 20125 Milan, Italy
[6]Helsinki University of Technology, Association EURATOM-TEKES, FIN-02150 Espoo, Finland
[7]Universitaet Karlsruhe, IHE, D-76131 Karlsruhe, Germany
e-mail: bernhard.piosczyk@ihm.fzk.de

A 170 GHz coaxial cavity gyrotron with 2 MW output power in continuous wave (CW) operation is under development in cooperation between European research centres together with European industry. A first industrial prototype of such a gyrotron has already been fabricated and delivered to CRPP Lausanne, where a suitable test facility has been constructed. Due to a delay in fabrication the delivery of the gyrotron magnet is expected in May 2007. Thus experimental tests are expected for the second half of this year.

In parallel to the industrial activities, experimental operation with a short pulse (~ few ms) 170 GHz coaxial gyrotron ("pre-prototype") which uses the same main components as designed for the industrial tube has been continued. The mechanism of parasitic low frequency (LF) oscillations around 260 MHz has been identified. Based on this identification, small modifications of the geometry of the coaxial insert have been made. As a result the starting current for the LF oscillations has been increased by a factor of about 3 causing a strong reduction of the LF amplitude. Measurements with a prototype of a microwave load, which has been designed and fabricated for operation with the 2 MW prototype tube, have been performed. In addition to the distribution of the microwave power absorbed on the wall, the amount of power reflected back into the gyrotron has been measured and its influence on gyrotron performance has been investigated. The performance of the quasi optical (q.o.) RF output system presently installed in the industrial prototype tube is insufficient, mainly because of the low Gaussian content of the RF output beam. As a first step a new launcher with a different wall corrugation and a new adapted phase correcting mirror has been designed and fabricated. According to simulations an increase of the Gaussian content to about 87% is expected. This q.o. RF output system has been installed in the pre-prototype tube for performing hot measurements.

This work was carried out within the framework of EFDA

Gyrotron Source System for ITER Plasma Start Up

S.L.Rao, K.Sathyanarayana, Mahesh K., Ujjwal K.Baruah , Ritesh S., Rajnish K.,

Institute For Plasma Research
Bhat,Gandhinagar,INDIA

Abstract

For a reliable plasma current start-up with a limited toroidal electric field of ~0.3V/m, ITER would require Electron Cyclotron Heating (ECH) assistance during this phase. An ECH Start-up system with an installed capacity of 3MW RF power, at a frequency of ~127 GHz with a maximum pulse length of 10 s, is being envisaged for ITER Plasma Start-up system.
The Indian Participating Team (IN PT) is currently working on the details of the gyrotron source including auxiliary power supplies, High voltage power supplies, protections & controls for the ECH Start-up system. The specified gyrotron sources are expected to be commercially available involving certain development on the part of supplier to re adopt the proven technologies to a new design suitable for the specified frequency. Diode type tube configuration would be preferred, as this would allow a simpler High voltage power supply configuration. The required HVPS would be based on PSM technology and one HVPS would be driving all the three start up gyrotrons in parallel. The required auxiliary power supplies like the Ion pump power supplies, Magnet power supplies, filament power supplies and the gyrotron tanks would be procured as per the basic designs and /or tube specifications. A VME based Data Acquisition and Control system would be built with various fast (critical) & slow interlocks for the safe operation of the tubes. For the critical faults, the HVPS would be removed within a time scale of 10 μs. Various parameters like rf, vacuum, cooling and DC parameters would be monitored and/or set remotely. Integrated testing of the gyrotron source system into a calorimetric water load is planned at IN-PT site. The paper highlights the details of the integrated gyrotron system as planned by the Indian participating team.

Advanced Gyrotron Collector Sweeping with Smooth Power Distribution

V. Erckmann[1], M. Schmid[2], H.P. Laqua[1], G. Dammertz[2], S. Illy[2], H. Braune[1], F. Hollmann[1], G. Michel[1], F. Noke[1], F. Purps[1],
ECRH groups at IPP Greifswald[1], FZK Karlsruhe[2], and IPF Stuttgart[1].

[1]*Max-Planck-Institut für Plasmaphysik (IPP), EURATOM-Association, D-17491 Greifswald, Germany*
[2]*Forschungszentrum Karlsruhe, Association EURATOM-FZK, IHM, D-76021 Karlsruhe, Germany*
[3]*Institut für Plasmaforschung, Universität Stuttgart, Pfaffenwaldring 31, D-70569 Stuttgart, Germany*

e-mail: volker.erckmann@ipp.mpg.de

Present day high power cw gyrotrons operate with an rf-power of typically 1 MW and efficiencies of 35-50 % [1,2], thus typically 1-2 MW power remains in the spent electron-beam and must be dumped in the gyrotron-collector. **V**ertical magnetic **F**ield **S**weeping (VFS) is a well-established method and commonly used to bring the local collector power density down to acceptable values. A less established method is '**T**ransverse **F**ield **S**weeping' (TFS), which was invented in Russia [3] and further investigated at FZK [4]. TFS proved to have a power capability comparable to VFS. The electron beam power distribution obtained with either method, however, shows an unfavorable profile with pronounced maxima and the collectors operate at the technical limits. As these maxima determine the overall collector capability, we have investigated a new collector sweeping method, which generates a homogenous power deposition profile thus enhancing the total heat load capability significantly. The experiments were performed using a commercial THALES Gyrotron TH 1507 (140 GHz, 1 MW, cw), recent results are reported. The application of the new sweeping method may be attractive for existing tubes, because they can be operated with higher safety margin (higher lifetime), as well as for future gyrotrons with higher rf-power (e.g. the next step ITER gyrotron), because existing collector designs may already satisfy the additional power demands.

[1] G. Dammertz, S. Alberti, A. Arnold, E. Borie, et al., *IEEE Trans. Plasma Science 30 (2002) 808-818*
[2] K. Felch, M. Blank, P. Borchard, P. Cahalan, S. Cauffman, T.S. Chu and H. Jory, *Journal of Physics: Conference Series 25 (2005) pp. 13-23.*
[3] private communication
[4] M. Schmid, S. Illy, G. Dammertz, V. Erckmann, M. Thumm: Transverse Field Collector Sweep System for high power CW Gyrotrons, *to be published in* Fusion Science and Technology

Beam characteristics including general astigmatism effects in the Remote Steering ITER ECRH Upper Launcher

A. Moro[1], W.A. Bongers[2], A. Bruschi[1], M.F. Graswinckel[2], E. Poli[3], D.M.S. Ronden[2], A.G.A. Verhoeven[2]

[1] *Istituto di Fisica del Plasma, CNR, Ass. EURATOM-ENEA-CNR, Milano, Italy*
[2] *FOM-Institute for Plasma Physics Rijnhuizen, Nieuwegein, The Netherlands*
[3] *IPP-Garching, Max-Planck-Institut für Plasmaphysik, Garching, Germany*

Abstract. General astigmatism treatment has been included in the description of the resulting beams for the latest design concepts proposed for the Remote Steering (RS) Launcher. The inclusion of general astigmatism is intended to provide the key parameters of the resulting beams and perform realistic beam tracing calculations in the plasma. It is in fact well known that the many requirements foreseen for the ITER ECRH Upper launcher force the beams to be injected in the plasma with a sufficient steering capability aiming at the stabilization of neoclassical tearing modes (NTM) through localized deposition of EC waves. To do this, double curvature mirrors are used in the latest RS quasi-optical systems. The mirrors are oriented at an angle with respect to each other and with respect to the directions of astigmatism of the incoming launched beam. The resulting beams are found to be generally astigmatic beams [1]. The correct description of the beam parameters resulting from this kind of complex launching system can in principle be used to optimize the RS quasi-optical system (in terms of localized heating and current drive efficiency), acting for example on the curvatures of the last mirror. The curvature in the direction determined by the steering plane should not be considered a free parameter, since the necessary output steering range has to be granted, but the curvature in the direction orthogonal to the steering plane could be modified to improve the performances of the launcher. Starting from the design proposed [2], the analysis presented here includes beam optics calculations. This approach was agreed within affiliate institues to optimise the efficiency of the RS Upper launcher design. The work was carried out under EFDA Task TW6-TPHE-ECHULB.

[1] A. Moro, A. Bruschi et al, *Sources of aberrations in the optics for the Remote sterring ITER ECRH Upper launcher*, Proc. 14th Joint Workshop on Electron Cyclotron Emission and Electron Cyclotron Resonance Heating, Santorini, Greece, 9-12 May 2006.

[2] M.F. Graswinckel, W.A. Bongers, et al., *Electron Cyclotron Resonance Heating Based On Remotely Steered Millimeter Waves Launched from the ITER Upper Port*, Fusion Science and Technology, to be published.

Numerical Calculations of Beam Patterns for the ITER ECRH Upper Launcher

P.Platania and C.Sozzi

Istituto di Fisica del Plasma, CNR, Ass. EURATOM-ENEA-CNR, Milano, Italy

The design of complex, multiple mirrors quasi-optical launchers in ECH&CD systems in fusion grade devices is constrained by severe requirements in terms of high power handling capability, extended steering range and room availability. In these cases, often diffractive effects and aberrations of the beams become important, significant side lobes and beam asymmetries arise [1] and the single mode Gaussian description is not longer sufficient. An efficient way to calculate the beam pattern in vacuum avoiding restrictive assumptions is using a numerical tool in which the detailed characteristics of the reflectors surfaces can be introduced and the resulting field propagation in vacuum can be computed including all the relevant effects. The description of the beam resulting from Physical Optics calculations takes into account the relevant causes of deformation and non gaussianity, such as aberration, beam truncation, thermal deformation of the mirrors and existence of surrounding structures [2].

In this work we discuss the application of the GRASP® electromagnetic code to the case of the ITER ECRH Upper Launcher. In the process of studying the launcher geometry of the Front Steering option we performed a truncation study to evaluate the beam pattern distortions with respect to mirror dimensions. For the Remote Steering, we simulated the so-called "Dogleg" layout to calculate the beam profiles at deposition location. In these studies, GRASP® has been proved to be a very efficient way to test the subsequent updates in the refinement phase of the model once the basic geometry has been implemented. The work was carried out under EFDA Task TW6-TPHE-ECHULB.

[1] A.Bruschi, S.Cirant, A.Moro, P.Platania, C.Sozzi, *Advanced Optics for the Remote Steering ITER ECRH Upper Launcher* Journal of Physics: Conf Series 25, 112-119, (2005).

[2] P.Platania, C. Sozzi, *Detailed beam pattern calculations for the ITER ECH&CD Upper Launcher* to appear on Fusion Science and Technology, special issue on ECH (2007)

Performance Evaluation of the Remote- Steering Option for the ITER EC Upper Launcher

E. Poli[1], W. A. Bongers[2], A. Bruschi[3], D. Farina[3], M. F. Graswinckel[2], M. A. Henderson[4], A. Moro[3], G. Ramponi[3], G. Saibene[5], A. G. A. Verhoeven[2], H. Zohm[1]

[1]*IPP-Garching, Max-Planck-Institut für Plasmaphysik, Garching, Germany*
[2]*FOM Institute for Plasma Physics Rijnhuizen, Nieuwegein, The Netherlands*
[3]*Istituto di Fisica del Plasma, CNR, Ass. EURATOM-ENEA-CNR, Milano, Italy*
[4]*CRPP, EURATOM –Confédération Suisse, EPFL, Lausanne, Switzerland*
[5]*EFDA Close Support Unit, Garching, Germany*

One of the main goals of the ITER Electron-Cyclotron (EC) Upper Launcher is the stabilisation of Neoclassical Tearing Modes (NTMs). The fulfilment of this objective is connected with the possibility of generating a current inside the island by EC waves, thus replacing the missing bootstrap current driving the mode. To achieve this, the EC current profile should be as narrow as possible, to increase the current density in the island region, thus reducing the total injected power necessary to suppress the mode.

A design option for the ITER Upper Launcher is based on the so-called Remote Steering (RS) concept [1], in which the steering mechanism is placed behind the waveguides, thus avoiding direct neutron bombardment from the plasma. In this paper, the performance of the latest RS layout is evaluated. Generally astigmatic beams [2] are employed to minimise the beam broadening in the plasma due to diffraction, which has a negative impact on the width of the current profile.

The stabilisation efficiency is estimated in terms of the figure of merit $\eta_{NTM}=J_{CD}/J_{bs}$ (J_{CD} being the EC driven current and J_{bs} the equilibrium bootstrap current at a given surface), for (2,1) and (3,2) modes for three reference ITER scenarios. The criterium for complete mode suppression is given as $\eta_{NTM} > 1.2$. The EC current profile is calculated by means of the TORBEAM code [3], which allows a straightforward modelling of generally astigmatic beams. The performance of a "long waveguide" design of the RS launcher is also discussed.

References

[1] A. G. A. Verhoeven *et al.*, J. Phys. Conf. Series **25**, 84 (2005).

[2] A. Bruschi *et al.*, J. Phys. Conf. Series **25**, 112 (2005); A. Moro, A. Bruschi et al, Proc. 14th Joint Workshop on Electron Cyclotron Emission and Electron Cyclotron Resonance Heating, Santorini, Greece, 2006.

[3] E. Poli *et al.*, Comp. Phys. Comm. **136**, 90 (2001).

Physics Analysis of the ITER ECW System for an Optimized Performance

G. Ramponi[1], D. Farina[1], M. A. Henderson[2], E. Poli[3], O.Sauter[2], G. Saibene[4], H. Zohm[3], C.Zucca[2], H.Takahashi [5]

[1]Istituto di Fisica del Plasma, CNR, Ass. EURATOM-ENEA-CNR, Milano,Italy
[2] CRPP, EURATOM –Confédération Suisse, EPFL, CH-1015, Lausanne, Switzerland
[3] IPP-Garching, Max-Planck-Institut für Plasmaphysik, D-85748 Garching, Germany
[4] EFDA Close Support Unit, Boltzmannstrasse 2, D-85748 Garching, Germany
[5]Naka Fusion Institute,Japan Atomic Energy Agency,Naka, Ibaraki,311-0193,Japan

The ITER ECW system consists of up to 24, 170 GHz gyrotrons, 1 to 2 MW each, connected to one launcher situated in one equatorial port (EL) and four launchers in the upper ports (UL), for a nominal injected power of 20MW. The two launchers systems, designed to inject mm-waves at various locations in the plasma, have different current drive characteristics: beams launched from the EL, having toroidal steering, give broad j_{CD} profiles and higher total driven current, good for central deposition and current profile control, while the UL, where the beams are steered in the poloidal direction, gives narrow j_{CD} profiles, good control of MHD activity such as NTM and sawteeth. Up to the end of 2005 the main goal of the physics analysis has been to evaluate the performance of the UL both in terms of the required steering range and of the figure of merit for Neoclassical Tearing Modes (NTMs) stabilization ($\eta_{NTM}=J_{CD}/J_{bs}$), for (2,1) and (3,2) modes for three reference ITER scenarios and for two launcher designs, the Front Steering (FS) and the Remote Steering (RS). Since the FS offers appreciable reserve in η_{NTM}, a new variant of the UL has been designed that improves the capabilities of the system with respect to sawteeth stabilization, by adopting different deposition ranges for upper and lower mirrors while still maintaining acceptable η_{NTM} over the full range. A possible synergy between the UL and EL has been pointed out, with the ultimate goal of providing an enhanced physics program in ITER. In this work it is shown that, by adding to present design of the EL the possibility to drive counter-current in addition to the existing co-current capability as well as modifying the poloidal tilt angle for the top and bottom steering mirrors, the flexibility of the system is increased and heating and current drive may be decoupled, increasing the capability for the control of the discharge and enhancing the operational domain. The possibility to avoid sawteeth by raising the q profile above q=1 in a variant of Scenario 2 and the large flexibility in controlling q_0 in the advanced Scenario 4 are shown. For the UL, the performance for NTM and sawteeth control by injecting the R.F. beams from a lower location are compared with that achieved with the present port location, both at nominal and at reduced magnetic field.

*) This work has been carried out under EFDA Technology Task TW6-TPHE-ECHULA

Optimum Corrugations for Low-loss Square and Cylindrical Waveguides

B. Plaum, E. Holzhauer, M. Grünert, H.Kumric

Institut für Plasmaforschung, Universität Stuttgart, Pfaffenwaldring 31, D-70569 Stuttgart, Germany
plaum@ipf.uni-stuttgart.de

Circular corrugated waveguides are widely used for transmission due to low loss propagation of the HE11 mode. Rectangular corrugated waveguides can be used for compact remote-steering antennas and power splitters. For high-power applications, the ohmic loss becomes important.

The experience however has shown, that analytical formulas for the losses often are too optimistic. This is mostly due to the fact, that they calculate the fields and losses by modelling the corrugated wall as a surface with anisotropic impedance. Thus, the fields and wall-currents inside the grooves, which contribute to the losses, are neglected. Furthermore, most calculation methods only work for rectangular corrugation profiles.
The new approach is based on the fact, that the propagation in a waveguide is similar to the periodic reflection of a plane wave at a planar wall.

The reflection losses are calculated using a 2-dimensional FDTD code. By the implementation of periodic boundary conditions, it becomes sufficient to consider a single corrugation period. The wall is modelled as a plasma with a plasma-frequency far above the microwave frequency. This results in a skin-depth in the order of a few grid-distances. By integrating the squared plasma current over the whole wall, one gets an expression, which is proportional to the dissipated power. The final calibration is then done with the well known reflection losses at a smooth wall under perpendicular incidence. The method works for arbitrary corrugation profiles and for both polarizations.

An additional feature of the code is the calculation of the phase of the reflected wave. By comparing the phase shifts for both polarizations, one can characterize a corrugation profile with respect the propagation of balanced hybrid modes or design optimum corrugation profiles for polarizers.

The paper presents the calculation method as well as the results for several typical groove profiles and comparisons with measurements.

Design of the ITER Electron Cyclotron Heating and Current Drive Waveguide Transmission Line

D. A. Rasmussen, T.S. Bigelow
Oak Ridge National Laboratory, PO Box 2008, Oak Ridge, TN, 37831, USA
Email: Rasmussenda@ornl.gov

M. A. Shapiro, J. R. Sirigiri, and R. J. Temkin
MIT Plasma Science and Fusion Center, MIT Building NW16-186, 167 Albany Street, Cambridge, MA 02139, USA

H. Grunloh
General Atomics, PO Box 85608, San Diego, CA 92186, USA

The ITER ECH transmission line components for the 24 MW 170 GHz cw heating and current drive system and the 127.5 GHz startup system are the responsibility of the US ITER team. The determination of specific performance requirements for the 63.5 mm diameter line and the initial design of components and layout between the gyrotrons and the launchers is underway[1,2]. Similar corrugated waveguide systems have been built and installed on several fusion experiments; however, none have operated at the high frequency and long-pulse required for ITER. Individual prototype components are being tested at low power to estimate ohmic and mode conversion losses[3]. A limited set of components have been or will be tested at 170 GHz with long-pulse high power at JAEA[4,] In order to develop and qualify the ITER components prior to procurement of the full set of 24 transmission lines, a 170 GHz high power test of a complete prototype transmission line is planned. Testing of the transmission line at 1-2 MW can be performed with a modest power (~0.5 MW) tube with a low loss (10-20%) resonant ring configuration that is being tested at ORNL[5]. Procurement of a suitable 170 GHz gyrotron is expected to take up to two years; however, a 140 GHz long pulse 400 kW gyrotron can be used in the initial round of tests.

[1] R. Olstad, et al, Journal of Physics: Conf. Series **25** (2005) 166-172
[2] T. S. Bigelow at al, to be published 17th Topical Conference on RF Power in Plasmas, Clearwater, FL, AIP (2007)
[3] S. T. Han, et al "Estimate of the Loss in the ECH Transmission Lines for ITER" (This workshop)
[4] M. Saigusa et al, Fusion Engineering and Design **74** (2005) 473-478
[5] T.S. Bigelow, Proc. 22nd Int. Conf. on IRMM Waves, Wintergreen, VA, IEEE (1997)

Estimation of the Loss in the ECH Transmission Lines for ITER

S. T. Han, M. A. Shapiro, J. R. Sirigiri, R. J. Temkin and P. P. Woskov
MIT Plasma Science and Fusion Center, MIT Building NW16-186, 167 Albany Street, Cambridge, MA 02139, USA Email: temkin@mit.edu

D. A. Rasmussen, Oak Ridge National Laboratory, PO Box 2008, Oak Ridge, TN, 37831

We present calculations of the loss in the ITER ECH transmission lines, including both the ohmic loss and the mode conversion loss of all of the components. Since the power generated by the gyrotron system is specified at 24 MW and the power delivered to the plasma should be at least 20 MW, the allowed loss in the lines must be held to a low value, no more than 17%. This value of loss, 17%, is also close to the theoretical loss value estimated by the ITER team, indicating that there is a challenge in achieving such a low loss value. Because some of the loss estimates are based on approximate theories, there is uncertainty in these estimates. Our estimate is a loss of about 11% for the transmission lines leading to the upper launcher. There is an additional 4% loss in the upper launcher itself. The total loss is thus 15% for the present estimate, slightly lower than the ITER Team estimate. For the transmission line to the equatorial plane launcher, the estimated total loss is 13% for the present calculation vs. 15% for the ITER Team. These estimates allow us to divide the loss budget between the two portions of the transmission line, since different parties are responsible for the different portions of the line.

The transmission line losses are estimated here by approximate theories. Improved theories and measurement techniques are needed and are under development. We will present results on the estimation of the mode conversion in an oversized miter bend using the computer code HFSS. The results show the excitation of higher order modes at the bend. The results are compared with pyroelectric camera measurements of mode conversion at a miter bend in a demonstration transmission line at MIT. Experimental results on the measurement of low microwave loss using a vector network analyzer and a radiometer will be presented and the results compared with theory. The present results will also be compared, where possible, with results from high power testing at 170 GHz at JAEA in Japan.

FADIS, a Fast Switch and Combiner for High-power Millimetre Wave Beams

W. Kasparek[1], M. Petelin[2], D. Shchegolkov[2], V. Erckmann[3], B. Plaum[1], A. Bruschi[4],
ECRH groups at IPP Greifswald[3], FZK Karlsruhe[5], and IPF Stuttgart[1].

[1]*Institut für Plasmaforschung, Universität Stuttgart, Pfaffenwaldring 31, D-70569 Stuttgart, Germany*
[2]*Institute of Applied Physics, RAS, 603950 Nizhny Novgorod, Russia*
[3]*Max-Planck-Institut für Plasmaphysik (IPP), EURATOM-Association, D-17491 Greifswald, Germany*
[4]*Istituto di Fisica del Plasma, EURATOM-ENEA-CNR Ass., via R. Cozzi 53, 20125 Milano, Italy*
[5]*Forschungszentrum Karlsruhe, Association EURATOM-FZK, IHM, D-76021 Karlsruhe, Germany*

e-mail: walter.kasparek@ipf.uni-stuttgart.de

Neoclassical tearing modes (NTMs) in tokamaks can be stabilized by electron cyclotron current drive (ECCD) at the corresponding resonant flux surface. Best efficiency is obtained if ECCD is applied in the O-point of the island, which requires modulation of the launched EC power synchronously with the rotating islands.

An alternative to modulation, which makes full use of the installed gyrotron power could be synchronous switching of the millimetre waves from a continuously operating source: A fast directional switch (FADIS) toggles the beam between two launchers in different poloidal or toroidal planes, which are 180° apart from each other with respect to the phase of the NTM. Generally, the device can be used to share the installed EC power between different types of launchers or different applications (e.g. in ITER, midplane / upper launcher), whichever is given priority during a plasma discharge. The switching is performed electronically without moving parts by a small frequency-shift keying of the gyrotron (some tens of MHz), and a narrow-band diplexer, which directs an input beam to one of the two output channels. The device can be operated as a beam combiner also, which offers attractive transmission perspectives in multi-megawatt ECRH systems.

The principle and the design of a four-port quasi-optical resonator diplexer is presented. Low-power measurements of switching contrast, mode purity and efficiency are shown and are compared with theory. First results from high-power tests using the ECRH system for W7-X are presented. Requirements and techniques for frequency control of the gyrotrons are discussed, and the results of preliminary frequency modulation experiments of two different types of gyrotrons are shown. Finally, the integration of this type of diplexer into corrugated waveguide transmission lines, as well as alternative waveguide diplexer concepts are discussed.

This work is carried out in the frame of the virtual institute "Advanced ECRH for ITER" (collaboration between IPP Garching and Greifswald, FZK Karlsruhe, IHE Karlsruhe, IPF Stuttgart, IAP Nizhny Novgorod, and IFP Milano), which is supported by the Helmholtz-Gemeinschaft deutscher Forschungszentren.

A General Method for Relativistic Plasma Dielectric Tensor Evaluation

S.S. Pavlov[1], F. Castejon[2]

1) Institute of Plasma Physics. NSC KhIPT.Akademicheskaya 1, 61108,Kharkov (Ukraine)

2) Asociación EURATOM-CIEMAT PARA Fusión. 28040, Madrid (Spain)

The study of EC waves in ITER requires to take into account relativistic effects, which are essential in the quasiperpendicular wave propagation regime. The exact relativistic tensor for Maxwellian plasma was first given by Trubnikov, but it was proved not to be practical for applications in the general case of arbitrary plasma and wave parameters, since its Hermitian parts contained complicated Cauchy singularities that had not been studied in detail before. To overcome difficulties, the weakly relativistic plasma dispersion functions (PDFs), which contain some approximation of those singularities, were introduced by Shkarofsky. The weakly relativistic approximation is widely used now in applications although the validity of this approximation is limited to the cases in which the relativistic effects are not strong enough.

A recipe to estimate the relativistic tensor in the general case was given in [1]. This method allows one to transform the tensor into a closed form suitable for applications. In this way, the dielectric tensor is presented as a Larmor radius expansion in terms of the exact PDFs, which have been especially introduced to deal with singularities. The exact PDFs are a natural generalization of the weakly relativistic PDFs for the case of an arbitrary plasma temperature.

Really, this method is more general and can be used also for evaluation of any Stieltjes and Hilbert integrals given at the real axis, provided that the density inside the integrals is continuous and tends to zero in the infinite limit. This method can be useful, in particular, for evaluating non-Maxwellian plasma distributions.

Since ITER plasmas will reach temperatures larger than 10 keV the former method can be useful for studying EC waves in such plasmas on the base both wave and ray tracing calculations. Therefore, one of the scopes of the present report is giving this method for the case $|N_{//}| < 1$ ($N_{//} = k_{//}c/\omega$ is longitudinal refractive index), which is relevant to EC waves.

The analytic continuation of the exact PDFs from the real region to the complex one using the same method is also presented. For the case $|N_{//}| < 1$ it is shown that the Riemann surface for every exact PDF is two branched analytic function in the entire complex plane with pricked out real branch point. This method can be used also to obtain the properties of EC plasma wave instabilities in ITER in the frame of the initial value problem in relativistic regimes.

1. F. Castejon and S.S. Pavlov, Phys. Plasmas, **13**, 072105 (2006).

Relativistic Effects in Electron Cyclotron Resonance Heating and Current Drive†

Abhay K. Ram
Plasma Science and Fusion Center
Massachusetts Institute of Technology
Cambridge, MA 02139-4307. USA

Joan Decker
Association Euratom-CEA/DSM/DRFC
Cadarache, 13108 Saint Paul lez Durance, France

In many conventional tokamaks waves in the electron cyclotron range of frequencies (ECRF), the extraordinary X mode or the ordinary O mode, have been successfully used for generating plasma current and for modifying the current profile. The same is envisioned for ITER. It is well-known that relativistic effects need to be included in a proper description of the damping of the EC waves. We have developed a code R2D2 which numerically solves the fully relativistic dispersion relation for EC waves [1]. This code is also used to construct the relativistic quasilinear diffusion operator which describes the interaction of EC waves with electrons. The relativistic three-dimensional code LUKE [2] solves the Fokker-Planck equation, with this quasilinear operator included, for the electron distribution function. R2D2 and LUKE have already been used to study EC waves in conventional tokamaks and in ITER, and, for the propagation of electron Bernstein waves, in spherical tori. R2D2 has recently been extensively modified using new physics based computational algorithm so that the code now runs more efficiently and the computational time has been significantly reduced. We will present results obtained with R2D2 on the relativistic characteristics of EC waves in the vicinity of the electron cyclotron resonance (and its harmonics). The effect on the electron distribution function and the driven plasma current will be studied using LUKE. An analysis of the results and their relevance to ITER will be discussed.

†Work supported by DoE Grants DE-FG02-91ER-54109 and DE-FG02-99ER-54521.

[1] A. K. Ram, J. Decker, and Y. Peysson, *J. Plasma Physics* **71**, 675 (2005).

[2] J. Decker and Y. Peysson, *Euratom-CEA Report EUR-CEA-FC-1736* (2004).

Electron Cyclotron Heating Modelling in Large Tokamaks and ITER with 3D Full Wave Code

Vdovin V.L.
RRC Kurchatov Institute, Nuclear Fusion Institute
Moscow, Russia

Abstract

We present modeling results of basic Electron Cyclotron Heating scenarios in several tokamaks and ITER performed with updated 3D full wave STELEC (stellarator_ECH, tokamaks included as particular case) code [1]. Code includes all basic wave physics as interference, diffraction, wave tunneling, mode conversion at Upper Hybrid (UH) resonance to electron Bernstein waves and appropriate boundary conditions. Code operates in real 3D magnetic geometry and uses massive parallel terabyte computers and firstly permitted solution of above problem. The basic non waited from ray tracing technique results are not only influence of diffraction effects but discovering UH resonance importance both at X-mode antenna excitation and at O-mode antenna excitation for fundamental harmonic (last one is contrary to ray tracing predictions). Thus so called "O and X" modes are coupled ones in exact solution. These effects, partly experimentally supported by DIII-D tokamak observed heating efficiencies, lead to another power deposition profiles and their space location, in compare with ray tracing technique. Coupling to X-mode reveals strong role of Electron Bernstein Waves previously neglected in ITER ECH modeling. These results can influence ECH/CD NTM suppression predictions for ITER, in parallel significantly decreasing requirements (respectively price) on ECH hardware (converters, polarization, etc) .Code permitted to investigate recent urgent issue: O-X-B ECH scenario for over dense tokamak/stellarator plasma "is miff or reality"? Finally, code discovered importance of diagnostic probe wave multi passes account for interferometry diagnostic interpretations in the past in many experiments and new view to plasma turbulence diagnostic using "increased wave scattering " at UH layer.

References

1. Vdovin V.L. *Role of Upper Hybrid resonance and diffraction effects at Electron Cyclotron Heating in tokamaks* , Proceedings of 14th Joint Workshop on Electron Cyclotron Emission and Electron Cyclotron Resonance Heating, invited lecture, p.323-333, 9 - 12 May 2006, Santorini island, Greece (Publisher: HeliotoposConferances Ltd., Athens, Greece; ISBN: 960-89228-2-8, December 2006)
2. Vdovin V.L. *Electron Cyclotron Heating modeling in tokamaks with 3D full wave code,* 33rd EPS Conference on Plasma Physics, Roma, Italy June 19–23, 2006 ECA vol.**301**, oral talk O2.015 (2006)

Nuclear Analyses for the ITER ECRH Launcher

A. Serikov[1a], U. Fischer[1a], R. Heidinger[1b], P. Spaeh[1b], S. Stickel[1a], H. Tsige-Tamirat[1a]
serikov@irs.fzk.de
[1] *Forschungszentrum Karlsruhe, Association FZK-EURATOM,*
(a) Inst. for Reactor Safety, (b) Inst. for Materials Research I, D-76021 Karlsruhe, Germany

The computational results of the nuclear analyses for the ECRH launcher integrated into the ITER upper port are presented in the paper. The purpose of the analyses was to provide the proof for the launcher design that the nuclear requirements specified in ITER project can be met. The aim was achieved on the basis of 3D neutronics radiation transport calculations using the Monte Carlo code MCNP and activation analysis of the launcher components using the FISPACT inventory code. In the course of the analyses an adequate shielding configuration against neutron and gamma radiation was developed keeping the necessary empty space for mm-waves propagation in accordance with the ECRH physics guidelines.

Special effort has been devoted to the generation of 3D MCNP neutronics models by the conversion from CAD geometry data. The dedicated interface program McCad has been used for the automatic conversion. The neutronics peculiarity of the launcher includes the problem of deep penetration radiation in the shield blocks combined with the neutron streaming effect along the launcher waveguides. Accordingly, MCNP variance reduction techniques such as weight windows, particles splitting, Russian roulette and point detectors have been applied in the neutron transport simulations. A wide range of nuclear responses has been calculated in the critical components of the launcher and its vicinity for the task of radiation shielding development. Neutron damage (dpa), nuclear heating, helium production rate, neutron and gamma fluxes have been calculated under the conditions of ITER operation. For the human safety issue of personnel access after the reactor shutdown the dose rate calculations have been performed by applying the Rigorous 2-Step (R2S) method. In conclusion, the analyses confirm that radiation shielding criteria and shutdown dose rate are met for the ITER nuclear design limits.

This work is being carried out under the EFDA technology research programme activities:
EFDA technology task TW6-TPHE-ECHULA and B

Thermal and Electromagnetic Study of the UPP for the ECRH in ITER

D. Strauß*[1a], R. Heidinger[1a], G. Gantenbein[1b], G. Hailfinger[1c], K. Kleefeldt[1a], A. Serikov[1c], P. Späh[1a]

[1] Forschungszentrum Karlsruhe, Association FZK-Euratom, Karlsruhe
(a) Institute for Materials Research I, (b) Institute for Pulsed Power and Microwave Technology, (c)Institute for Reactor Safety
P.O. Box 3640, D-76021 Karlsruhe, Germany

The ECRH system in the ITER upper launcher provides a stable structure during ITER operation. Besides of the neutron and mm-wave loads during regular operation, plasma disruptions lead to fast changes in the magnetic field, eddy currents are induced interacting with the static toroidal field. As a consequence, high mechanical forces and torques act on the launcher structure. In numerical electro-dynamic simulations these currents and the resulting mechanical loads have been calculated. These loads are applied to a 3D finite element model of the upper launcher structure, its deformation and occurring stresses are studied.

* Corresponding author:

Dirk Strauß, Forschungszentrum Karlsruhe, P.O. Box 3640, 76021 Karlsruhe, Germany, email: dirk.strauss@fzk.de, phone: +49 7247-823964, fax: +49 7247-824567

Structural System of the ECH Upper Port Plug for ITER

R. Heidinger[1a], P. Spaeh[1a], R. Chavan[2], U. Fischer[1b], M. Henderson[2],
K. Kleefeldt[1a], A. Meier[1a], A. Serikov[1b], D. Strauß[1a]
roland.heidinger@imf.fzk.de
[1] *Forschungszentrum Karlsruhe, Association FZK-EURATOM,*
(a) Inst. for Materials Research I, (b) Inst. for Reactor Safety, D-76021 Karlsruhe, Germany
[2] *CRPP, Association EURATOM- Conf. Suisse, EPFL, CH-1015 Lausanne, Switzerland*

Four positions of the ITER upper port configuration are foreseen to accommodate the EC wave launching system for inducing localised current drive (CD) at the magnetic island locations. Its main objective is to stabilise the Neoclassical Tearing Modes (NTM) which is achieved by steering the EC beams over a range of plasma areas (typically ρ_p of 0.64 - 0.9) with a CD efficiency sufficient to stabilise 3/2 and 2/1 NTMs for the reference scenario 2, but also for the hybrid scenario (#3) and for the low-q operation (scenario #5). It is further desirable that the Upper Launchers can be used to access the q=1 region of plasmas with positive shear (function potentially shared with the equatorial launcher) to provide control of the sawtooth period and amplitude. In present reference design, the "Extended Performance front steering Launcher" or EPL, this functionality is achieved with a steering range covering ρ_p of 0.4 - 0.9. The development of the structural system of the EPL is presented in this paper.

The mm-wave components are integrated into the upper port plug structure which consists of two separate units, namely the blanket shield module (BSM) forming the plasma-facing component and the launcher main structure, which are connected with a bolted flange allowing axial access to the plug internals for maintenance and disassembly. The development of the structural system covers the adaptation of the shielding and cooling configuration of the BSM and of the internal neutron shield in the main structure to comply with the mm-wave beam line design. The evolving manufacturing concepts striving for communality with diagnostics plugs has led to the current slim wall design for the main structure. It is characterised by a passively cooled single wall section forming the central part intermediate to the actively cooled double wall sections at the front and back end. With a removable cover it allows vertical access to the internals. The baking concept is analysed for this design proving that the baking temperature at the port plug does not fall below 200°C.

This work is being carried out under the EFDA technology research programme activities:
EFDA technology task TW6-TPHE-ECHULA and B

Interface Issues Associated with the ITER ECH System

M.A. Henderson[1], G. Saibene[2], T. Bonicelli[2], R. Chavan[1], D. Farina[3], D. Fasel[1], R. Heidinger[4], E. Poli[5], G. Ramponi[3], O. Sauter[1], H. Zohm[5],

[1] CRPP, EURATOM – Confédération Suisse, EPFL, CH-1015 Lausanne Switzerland
[2] EFDA Close Support Unit, Boltzmannstrasse 2, D-85748 Garching, Germany
[3] Istituto di Fisica del Plasma, EURATOM- ENEA- CNR, 20125 Milano, Italy
[4] Forschungszentrum Karlsruhe, EURATOM-FZK, D-76021 Karlsruhe, Germany
[5] IPP-Garching, Max Planck-Institute für Plasmaphysik, D-85748 Garching, Germany

First Author e-mail: mark.henderson@epfl.ch

The ITER ECH system, is an in kind procurement consisting of four different types of gyrotrons (from EU, IN, JA and RF), transmission lines (from US) and two launcher (from EU and US). Each of these subsystems has to interface not only between themselves but also with the ITER auxiliary and control systems as well as with the plasma (in the case of the launcher). The definition and management of interfaces is therefore essential for the system to guarantee the required performance, availability and reliability and proper description of each interface boundary is essential for assembly and operation of the entire system as a single unit. In addition, the present ITER ECH system was essentially specified prior to 2000, since then progress has been made in the development of high power, long pulse systems and associated components. The ultimate physics performance and operational reliability is limited by this older technology, which has not taken advantage of the knowledge and experience gained in operating the multi-megawatt ECH systems on present tokamaks and stellarators.

The objective of this paper is to review the present ITER ECH system, which includes the power supplies, gyrotrons, transmission lines and launchers. Then propose modifications that are performance driven and engineered for reliability and maintainability, while reducing the complexity and costs. Potential operating scenarios will be discussed that require an intelligent and automatic decision making process, for example directing the EC power to either of the two EC launchers based on the immediate physics requirements. The interfaces between the subsystems will be described and when possible improvements to each interface will be proposed.

4th IAEA Technical Meeting on ECRH Physics and Technology for ITER
Vienna, 6-8 June 2007

The Advantage of Early Application of Electron Cyclotron Waves for the Suppression of Tearing Modes: --Assessment for ASDEX Upgrade and Results from TEXTOR--

A. Lazaros[1,2], M. Maraschek[3], E. Westerhof[2], H. Zohm[3],
M.F.M. de Bock[2], I.G.J. Classen[2], A.J.H. Donné[2], R.J.E. Jaspers[2], H.R. Koslowski[4],
A. Krämer-Flecken[4], J.W. Oosterbeek[4], S. Soldatov[4], G.W. Spakman[2], O. Zimmermann[4]

[1]*National Technical University of Athens, Greece,*
Association Euratom-Hellenic Republic

[2]*FOM Institute for Plasma Physics Rijnhuizen, Nieuwegein, The Netherlands,*
Association Euratom-FOM

[3]*Max Planck Institute for Plasma Physics, Garching, Germany,*
Euratom Association

[4]*FZJ Institute for Plasma Physics, Jülich, Germany,*
Association Euratom-FZJ

ABSTRACT

The advantage of the early application of electron cyclotron waves (ECWs) for the suppression of tearing modes, which are destabilized by high beta in ASDEX Upgrade and perturbation fields (generated by the dynamic ergodic divertor) in TEXTOR, is attributed in both cases to a second stable solution of the Rutherford equation, which appears (at smaller island width than the usual saturated island width) in the presence of localized heating and current drive, despite the fact that the dominant terms of the Rutherford equation in these two limits are completely different. It is shown analytically that the advantage of early ECCD (for the suppression of NTMs) in ASDEX Upgrade is favored by the broad deposition profiles compared to the critical island width (at which perpendicular transport across the island becomes comparable to the parallel transport). The model is consistent with the preliminary experimental results. In TEXTOR it was previously established that the suppression of tearing modes by ECWs is dominated by heating, but the advantage of the early application is a consequence of the non-inductive current drive, which is found to be more important for small islands. The model is confirmed by the experiment.

Locked Neoclassical Tearing Mode Control on DIII-D by ECCD and Magnetic Perturbations*

F. Volpe,[1] R.J. La Haye,[2] R.Prater,[2] and E.J.Strait[2]

[1] *Max-Planck-Gesellschaft, Germany*
[2] *General Atomics, P.O. Box 85608, San Diego, California 92186-5608 USA*

email: volpe@fusion.gat.com, Phone: 858-455-2495, Fax: 858-455-4156

Neoclassical tearing modes (NTMs) in ITER are expected to initially rotate very slowly and thus be prone to stop rotating and "lock" to the resistive wall and error field. They can lock with a toroidal phase such that they cannot necessarily be accessed and suppressed by electron cyclotron current drive (ECCD), therefore a more general stabilization approach becomes necessary.

New techniques where ECCD is assisted by magnetic perturbations exerted by internal coils (I-coils) were tested at DIII-D. Balanced neutral beam injection allowed reproducing ITER-like conditions of high β (for mode onset) and low rotation.

In the first type of experiment, magnetic perturbations were used to steer the mode and lock it with a new phase such that it could be stabilized by ECCD. Slowly rotating fields and a radial jog of the plasma were used to toroidally and radially align the island to the ECCD. Mitigation of the locked NTM was obtained with this technique with 1.3 MW of ECCD power; modeling suggests that 3 MW would completely suppress the island.

In the second class of experiments, rotating fields unlocked the mode and sustained its rotation. This is useful in many ways: (1) it prevents further locking; (2) it rotationally mitigates the mode; (3) it brings the locked mode case into the well-studied, easy-to-stabilize rotating NTM case; and (4) it opens up the possibility to synchronize and phase-lock the mode rotation to the ECCD modulation, which is simpler than adapting the ECCD to the natural mode frequency and phase. Sustained rotation at up to ~60 Hz was demonstrated and a sudden mode mitigation was observed at ~10 Hz, which might be due to the island shrinking below a critical width, as a result of the rotationally improved shielding.

Finally, new detectors of rotating precursors of locked modes were developed. First results and plans for their application to pre-emptive locked mode control and avoidance will be presented.

---

*Work supported by the U.S. Department of Energy under DE-FC02-04ER54698.

Modeling and Control for Fusion Plasma Stabilization by Means of a Mechanical ECRH Launcher at TEXTOR

B.A. Hennen [1], J.W. Oosterbeek [2], P.W.J.M. Nuij [1], M. Steinbuch [1], E. Westerhof [3], B.C.E. Vaessen [3] and the TEXTOR-team

[1] *Eindhoven University of Technology, Control Systems Technology Group, PO Box 513, 5600 MB Eindhoven, the Netherlands, www.tue.nl*

[2] *Forschungszentrum Jülich GmbH, Institut für Plasmaphysik*, Association EURATOM-FZJ, 52425 Jülich, Germany, www.fz-juelich.de*

[3] *FOM-Institute for Plasma Physics Rijnhuizen*, Association EURATOM-FOM, PO Box 1207, 3430 BE Nieuwegein, the Netherlands, www.rijnh.nl*

** Partners in the Trilateral Euregio Cluster*

Tokamak plasmas typically constitute high-order nonlinear distributed systems, which suffer from a large number of instabilities or off-normal events requiring actively controlled stabilization. This paper discusses several relevant control aspects, regarding the specific task of feedback-controlled stabilization of Neoclassical Tearing Modes (NTMs), for the state of the art ECRH installation operated at TEXTOR by the Electron Cyclotron Wave Physics Group of the FOM Institute for Plasma Physics Rijnhuizen, the Netherlands.

The work reported primarily focuses on the mechanics and control of a mechanical ECRH launcher system with a 2 rotational degrees of freedom (DOF) steerable mirror, which is applied as the last component of a transmission line, coupling gyrotron power into the tokamak plasma. Based on dynamical analysis and Frequency Response Function (FRF) characterization, the dynamics of this complex and high-precision instrument, operated under challenging conditions, are modeled in terms of transfer function estimation. Since the experiments clarify quite non-linear behavior of the system (mainly induced by friction) an analysis of the friction forces is also given, based on existing models and experimental observations. The impact of external sources of disturbances has been considered as well. The system identification procedure conducted allows design and implementation of a cascaded control strategy for improved actuation of the mechanical launcher and also provides information on the physical limitations of the system.

Furthermore, a proposal is made for development of more advanced NTM control scenarios dedicated to the TEXTOR ECRH installation. Such development will require extensive simulation incorporating mechanical as well as physical models in parallel. These models will be subject to theoretical and experimental verifications. Before application in a real-time system environment these strategies might first undergo an extensive test-bed. Controllers designed throughout this procedure should, for example, be capable of magnetic island stabilization on different flux surfaces and localization and tracking of fluctuating islands. Many elements of this method possess a high level of complexity but fortunately specific control expertise, modeling of process dynamics and system identification techniques, used in different engineering disciplines, can equally well be applied for fusion plasma control.

Status of the New Multi-Frequency ECRH System for ASDEX Upgrade

D. Wagner, F. Leuterer, A. Manini, F. Monaco, M. Münich, H. Schütz,
J. Stober, H. Zohm, T. Franke
Max-Plank-Institut für Plasmaphysik, EURATOM-IPP, Boltzmannstr.2, D-85748 Garching, Germany
M. Thumm[1], R. Heidinger[2], I. Danilov[2], G. Gantenbein[1], J. Flamm[1]
Forschungszentrum Karlsruhe, EURATOM-FZK
[1]Institut für Hochleistungsimpuls- und Mikrowellentechnik,
[2]Institut für Materialforschung,
PO Box 3640, D-76021 Karlsruhe, Germany
W. Kasparek
Institut für Plasmaforschung, Universität Stuttgart, Pfaffenwaldring 31, D-70569 Stuttgart, Germany
A.G. Litvak[3], L.G. Popov[4], V.O. Nichiporenko[4], V.E. Myasnikov[4], G.G. Denisov[3],
E.M. Tai[4], E.A. Solyanova[4], S.A. Malygin[4]
[3] Institute of Applied Physics, RAS, 46 Ulyanov St., Nizhny Novgorod, 603950, Russia
[4] GYCOM Ltd, 46 Ulyanov St., Nizhny Novgorod, 603155, Russia

A two-frequency GYCOM gyrotron, Odissey-2, has been installed and put into operation in the new multi-frequency ECRH system at the ASDEX Upgrade tokamak experiment. It works at 105GHz and 140GHz with output power 640kW and 880kW respectively at a pulse length of 10s. Meanwhile, the first two-frequency gyrotron Odissey-1, which was damaged last year, is back at GYCOM for repair. It will be equipped with a broadband Brewster output window and therefore become a multi-frequency gyrotron. A further extension of the system with 2 more gyrotrons is underway. Depending on the success of Odissey-1, these gyrotrons will also be step-tunable and operate at two additional intermediate frequencies between 105 and 140GHz. Construction and cold test of a first broadband double-disc torus window are completed. The transmission to the torus is in normal air, through corrugated aluminum waveguides with I.D.=87mm over a total length of about 70m. Calorimetric measurements gave a total transmission loss of only 12% at 105GHz and 10% at 140GHz. The variable frequency will significantly extend the operating range of the ECRH system with respect to the applied toroidal magnetic field. A feedback system for the mirror control is under development, to keep the ECRH deposition on the resonant q-surface while the plasma is evolving. This is required for applications such as the suppression of neoclassical tearing modes.

INDEX OF AUTHORS

LAST NAME	INITIALS	TITLE OF PAPER
Alberti	S.	Experiments with Real-time controlled ECW experiments on the TCV Tokamak
Bongers	W.A.	Recent developments of the Upper port ECH&CD launcher systems for ITER based on the remote steering concept
Bucalossi	J.	First experiments of plasma start-up assisted by ECRH on Tore Supra
Denisov	G.G.	Development in Russia of High Power Gyrotrons for Fusion
Denisov	G.G.	Window Development at IAP
Erckmann	V.	Advanced Gyrotron Collector Sweeping with smooth power distribution
Felch	K.L.	Operating experience on six, 110 GHz, 1 MW gyrotrons for ECH applications
Giruzzi	G.	Integrated Modelling of ITER Scenarios with ECCD
Goodman	T.P.	Ten Years of Experience in Integrated Control of the Multi-Megawatt ECW system on the TCV Tokamak
Heidinger	R.W.	Structural system of the ECH Upper Port Plug for ITER
Henderson	M.A.	Design status of the ITER upper port launcher
Henderson	M.A.	Interface issues associated with the ITER ECH system
Hennen	B.A.	Modeling and control for fusion plasma stabilization by means of a mechanical ECRH launcher at TEXTOR

Kasparek	W.	FADIS, a fast switch and combiner for high-power millimeter wave beams.
La Haye	R.J.	Requirements for Alignment of ECCD for NTM Stabilization in ITER
Lazaros	A.	The advantage of early application of electron cyclotron waves for the suppression of tearing modes: Assessment for ASDEX Upgrade and results from TEXTOR
Manini	A.	Plasma Stability Enhancement using EC-RH/CD in ASDEX Upgrade
Marushchenko	N.B.	Electron Cyclotron Current Drive Predictions for ITER: comparison of different models
Moro	A.	Beam characteristics including general astigmatism effects in the Remote Steering ITER ECRH Upper Launcher
Pavlov	Sergiy	A general method for relativistic plasma dielectric tensor evaluation
Piosczyk	B.A.	Status of the 2 MW, 170 GHz Coaxial Cavity Gyrotron for ITER
Platania	P.	Numerical calculations of beam patterns for the ITER ECRH Upper Launcher
Plaum	B.M.	Optimum Corrugations for Low-Loss Square and Cylindrical Waveguides
Poli	E.	Performance Evaluation of the Remote- Steering Option for the ITER EC Upper Launcher
Ram	A.K.	Relativistic Effects in Electron Cyclotron Resonance Heating and Current Drive

Ramponi	G.	Physics analysis of the ITER ECW system for an optimized performance
Rao	S.L.	Gyrotron Source System for ITER Plasma start up
Rasmussen	D.A.	Design of the ITER Electron Heating and Current Drive Waveguide Transmission Line
Sakamoto	K.	Demonstration of 1MW high efficiency oscillation on 170 GHZ CW Gyrotron
Serikov	A.	Nuclear analyses for the ITER ECRH launcher
Strauss	D.	Thermal and electromagnetic study of the UPP for the ECRH in ITER
Takahashi	K.	Development of ITER Equatorial EC Launcher for Reliability Improvement
Temkin	R.J.	Estimation of the loss in the ECH Transmission lines for ITER
Vdovin	V.L.	Electron cyclotron heating modelling in large tokamaks and ITER with 3D full wave code
Volpe	F.	Locked Neoclassical Tearing Mode Control on DIII-D by ECCD and Magnetic Perturbations
Wagner	D.H.	Status of the New Multi-Frequency ECRH System for ASDEX Upgrade

LIST OF PARTICIPANTS

Last Name	Initials	Country	Address	Email
Bouquey	F.	France	CEA Centre de Cadarache DRFC/SCCP Bat 508 13108 Saint Paul lez Durance France	francis.bouquey@cea.fr
Bucalossi	J.	France	CEA Centre de Cadarache DRFC/SCCP Bat 508 13108 Saint Paul lez Durance France	jerome.bucalossi@cea.fr
Darbos	C.	France	CEA Centre de Cadarache DRFC/SCCP Bat 508 13108 Saint Paul lez Durance France	caroline.darbos@cea.fr
Giruzzi	G.	France	DRFC/CEA/Cadarache Centre d'études de Cadarache 13108 Saint-Paul-lez-Durance France	gerardo.giruzzi@cea.fr
Erckmann	V.	Germany	Max Planck Institut für Plasmaphysik Teilinstitut Greifswald Wendelsteinstrasse 1 17491 Greifswald, GERMANY	volker.erckmann@ipp.mpg. de
Heidinger	R.W.	Germany	Forschungszentrum Karlsruhe, Association FZK-EURATOM, Inst. for Materials Research I P.O. Box 3610 76021 Karlsruhe, Germany	roland.heidinger@imf.fzk.de
Kasparek	W.	Germany	Institut für Plasmaforschung der Universität Stuttgart Pfaffenwaldring 31 D-70569 Stuttgart, Germany	walter.kasparek@ipf.uni- stuttgart.de
Manini	A.	Germany	Max-Planck-Institut für Plasmaphysik EURATOM Association Boltzmannstrasse 2 85748 Garching, Germany	adriano.manini@ipp.mpg.de

Marushchenko	N.B.	Germany	Max-Planck-Institut fuer Plasmaphysik (IPP) Teilinstitut Greifswald, EURATOM Association Wendelsteinstr. 1 17491 Greifswald, Germany	nikolai.marushchenko@ipp.mpg.de
Piosczyk	B.A.	Germany	Forschungszentrum Karlsruhe Postfach 3640 D-76021 Karlsruhe Germany	bernhard.piosczyk@ihm.fzk.de
Plaum	B.M.	Germany	Institut für Plasmaforschung Universität Stuttgart Pfaffenwaldring 31 D-70569 Stuttgart, Germany	plaum@ipf.uni-stuttgart.de
Poli	E.	Germany	Max-Planck-Institut für Plasmaphysik Boltzmannstrasse 2 85748 Garching bei München, Germany	Emanuele.Poli@ipp.mpg.de
Saibene	G.	Germany	EFDA Close Support Unit – Garching 2 Boltzmannstrasse 85748 Garching Germany	gabriella.saibene@efda.org
Serikov	A.	Germany	Forschungszentrum Karlsruhe Association FZK-EURATOM P.O. BOX 3640 76021 Karlsruhe, Germany	serikov@irs.fzk.de
Strauss	D.	Germany	Forschungszentrum Karlsruhe Association FZK-Euratom, Karlsruhe Institute for Materials Research I , Germany	dirk.strauss@imf.fzk.de
Thumm	M.K.A.	Germany	Forschungszentrum Karlsruhe Institute fur Hochleistungsimpuls und Mikrowellentechnik (IHM) Hermann von Helmhotz Platz 1 76344 Eggenstein-Leopoldhafen, Germany	Manfred.Thumm@ihm.fzk.de
Wagner	D.H.	Germany	Max-Planck-Institut für Plasmaphysik Boltzmanstrasse 2 D-85748 Garching Germany	dietmar.wagner@ipp.mpg.de

Lazaros	A.	Greece	School of Electrical and Computer Engineering, 9 Iroon Polytechniou Str., GR-157 73 Zographou, Athens, Greece	avrilios@central.ntua.gr
Malaquias	Artur Jorge	IAEA		A-J.Louzeiro-Malaquias@iaea.org
Rao	S.L.	India	Institute for Plasma Research Near Indira Bridge Bhat, Gandhinagar, 382428, India	slrao@ipr.res.in
Cirant	S.	Italy	Istituto di Fisica del Plasma Consiglio Nazionale delle Ricerche via Cozzi 53 - 20125 Milano - Italia	cirant@ifp.cnr.it
Moro	A.	Italy	Istituto di Fisica del Plasma CNR, Ass. EURATOM-ENEA-CNR Milano, Italy	moro@ifp.cnr.it
Platania	P.	Italy	Istituto di Fisica del Plasma CNR, Ass. EURATOM-ENEA-CNR Milano, Italy	platania@ifp.cnr.it
Ramponi	G.	Italy	Istituto di Fisica del Plasma, CNR, Ass. EURATOM-ENEA-CNR via Cozzi 53 20125 Milano, Italy	ramponi@ifp.cnr.it
Sakamoto	K.	Japan	Japan Atomic Energy Agency (JAE) 801-1 Mukoyama, Naka, Ibaraki 311-0193 Japan	sakamoto.keishi@jaea.go.jp
Takahashi	K.	Japan	Japan Atomic Energy Agency 801-1 Mukoyama, Naka, Ibaraki 311-0193 Japan	takahashi.koji@jaea.go.jp
Namkung	Won	Korea, Republic of	POSTECH, Department of Physics San-31, Hyoja-dong Pohang 790-784, Korea	namkung@postech.ac.kr

Bongers	W.A.	Netherlands	FOM Institute for Plasma Physics "Rijnhuizen" Association Euratom-FOM, Edisonbaan 14 NL-3439 MN Nieuwegein The Netherlands	W.A.Bongers@rijnhuizen.nl
Hennen	B.A.	Netherlands	Eindhoven University of Technology Control Systems Technology Group P.O. Box 513 5600 MB Eindhoven Netherlands	b.a.hennen@student.tue.nl
Ronden	D.M.S.	Netherlands	FOM Institute for Plasma Physics Rijnhuizen Edisonbaan 14 3439 MN Nieuwegein, Netherlands	D.M.S.Ronden@rijnhuizen. nl
Verhoeven	A.G.A.	Netherlands	FOM-Institute for Plasma Physics P.O. Box 1207 3430 BE Nieuwegein The Netherlands	verhoeven@Rijnhuizen.nl
Denisov	G.G.	Russian Federation	Institute of Applied Physics of RAS 46 Ulyanov Str. Nizhny Novgorod, 603950 Russian Federation	den@appl.sci-nnov.ru
Litvak	A.G.	Russian Federation	Institute of applied physics of RAS 46 Ulyanov Str. Nizhny Novgorod 603950 Russian Federation	litvak@appl.sci-nnov.ru
Vdovin	V.L.	Russian Federation	RRC Kurchatov Institute Nuclear Fusion Institute Kurchatov sp.1 Moscow 123182, Russia Federation	vdov@pike.pike.ru
Albajar	F.	Spain	European Fusion Development Agreement Close Support Unit – Garching Boltzmannstr.2 D-85748 Garching bei Muenchen Germany	Ferran.Albajar@tech.efda.o rg

Alberti	S.	Switzerland	Centre de Recherches en Physique des Plasmas Association EURATOM-Confédération Suisse Ecole Polytechnique Fédérale de Lausanne, SB Station 13 CH-1015 Lausanne, Switzerland	stefano.alberti@epfl.ch
Goodman	T.P.	Switzerland	Centre de Recherches en Physique des Plasmas Association EURATOM-Confédération Suisse Ecole Polytechnique Fédérale de Lausanne, SB Station 13 CH-1015 Lausanne Switzerland	timothy.goodman@epfl.ch
Henderson	M.A.	Switzerland	Centre de Recherches en Physique des Plasmas Association EURATOM-Confédération Suisse Ecole Polytechnique Fédérale de Lausanne, SB Station 13 CH-1015 Lausanne, Switzerland	mark.henderson@epfl.ch
Sauter	O.J.-L.	Switzerland	Centre de Recherches en Physique des Plasmas Association EURATOM-Confédération Suisse Ecole Polytechnique Fédérale de Lausanne, SB Station 13 CH-1015 Lausanne, Switzerland	olivier.sauter@epfl.ch
Pavlov	Sergiy	Ukraine	Institute of Plasma Physics, National Sciences Center "Kharkov", Institute of Physics and Technology, Academicheskaya Str., 1, Kharkov 61108, Ukraine	pavlovss@ipp.kharkov.ua
Callis	R.W.	United States of America	General Atomics, P.O. Box 85608, San Diego, CA 92608, United States of America	callis@fusion.gat.com

Felch	K.L.	United States of America	Communications and Power Industries 607 Hansen Way, MS-B-450 Palo Alto, CA 94304, USA	kevin.felch@cpii.com
La Haye	R.J.	United States of America	General Atomics P.O. Box 85608 San Diego, CA 92186-5608 USA	lahaye@fusion.gat.com
Luce	T.C.	United States of America	General Atomics P.O. Box 85608 San Diego, CA 92186-5608 United States of America	luce@fusion.gat.com
Ram	A.K.	United States of America	Plasma Science and Fusion Center Massachusetts Institute of Technology Room NW 16-260 77 Massachusetts Avenue Cambridge, MA 02139-4307 United States of America	abhay@mit.edu
Rasmussen	D.A.	United States of America	US ITER Project Office Oak Ridge National Laboratory 055 Commerce Park PO Box 2008, Oak Ridge, TN 37831-6483 United States of America	Rasmussenda@ornl.gov
Temkin	R.J.	United States of America	Plasma Science and Fusion Center MIT Building NW16-186 167 Albany Str. Cambridge, MA 02139, USA	temkin@mit.edu