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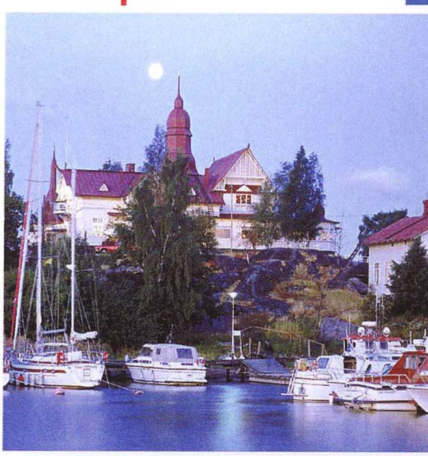
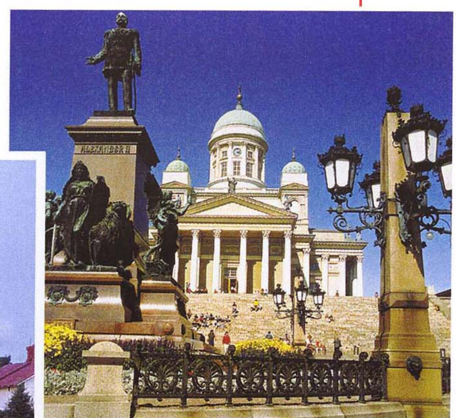
22nd
SOFT
2002

22nd Symposium on Fusion Technology

Book of Abstracts





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The 22nd Symposium on Fusion Technology

Book of Abstracts

Helsinki, Finland, 9th–13th September 2002

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Harri Tuomisto
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Text preparing Åsa Åvall, Tarja Haapalainen and Arja Grahn

Otamedia Oy, Espoo 2002

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PREFACE

The objective of the Symposium on Fusion Technology (SOFT) is to exchange information on design, construction and operation of fusion experiments and the technology of present fusion machines, the next step and power plant devices. It includes oral and poster presentations as well as an industrial and R & D exhibition.

The Symposium will be held in Helsinki at the Marina Congress Center, from 9th to 13th September 2002. The Symposium is organized by the Association Euratom-Tekes and hosted by the VTT Technical Research Centre of Finland, Fortum Nuclear Services Ltd. and PrizzTech Oy.

This 22th SOFT Symposium includes invited and contributed papers on all the aspects of fusion technology i.e:

A	Current and Future Devices
B	Plasma Facing Components
C	Plasma Heating and Current Drive
D	Plasma Engineering and Control
E	Diagnostics, Data Acquisition and Remote Participation
F	Magnets and Power Supplies
G	Fuel Cycle
H	Remote Handling
J	Vessel, Blanket and Shield
K	Safety and Environment, Power Plant and Socio-Economic Studies
L	Inertial Fusion Energy
M	Transfer of Technology.

The international organizing committee has invited 15 speakers to give overviews and updates on different topical subjects. A total of 484 submitted abstracts were reviewed to select 22 oral and 404 poster presentations. All accepted abstracts are included in this book. The distribution of presentations is shown in the following table.

	A	B	C	D	E	F	G	H	J	K	L	M	Total
Posters	13	60	48	28	38	53	30	15	67	44	5	3	404
Orals	1	3	1	2	1	3	0	1	3	6	1	0	22
Invited	6	1	1	1	0	1	1	0	1	1	0	1	14+1

All invited, oral and poster papers will be peer reviewed by two referees for consideration to be published in a special issue of the journal Fusion Engineering and Design. Consequently, the final number of published papers may be significantly less than the number of contributions at the symposium.

The 22nd SOFT Symposium is complemented by two exhibitions, i.e., Industrial Exhibition and Association Exhibition. The industrial companies active in fusion technology will present their processes and products and exchange information with experts. The Association exhibition gives an overview of fusion research and development activities in different European Countries participating in European Fusion Programme.

"Vision is not enough, it must be combined with venture. It is not enough to stare up the steps, we must step up the stairs."

Vaclav Havel

The editors wish to express their sincere gratitude to the authors for high quality abstracts and to the International Organising Committee for reviewing and selecting these abstracts. We also like to thank the Local Organising Committee and special thanks are due to Ms Auli Leveinen and Mr Tuomo Hokkanen for their valuable contributions for helping to edit the book of abstracts.

The International and Local Organizing Committees would like to thank the following Institutions and Companies for supporting the Symposium:

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MONDAY, 9th September

Poster session P1
Chairmen: Dr. Duchateau, J.-L. and Prof. Salomaa, R.

Poster presentations

THE EU FUSION ENERGY RESEARCH

J.-P. Rager

European Commission, Directorate General Research

The status and content of the EU Fusion Energy Research activities in the Euratom 6th Framework Programme will be presented. The overall objectives of FP6 fusion activities and their reactor orientation will be recalled and the priorities during FP6 (Associations' programmes, exploitation of the JET facilities in the framework of EFDA, and Next Step / ITER activities) will be detailed.

The status of negotiations on the Joint Implementation of ITER, the possible EU participation in ITER construction, if and when so decided, and the schedule will be presented. This participation will depend inter alia on the outcome of the negotiations with the EU international partners and in turn on the location of the ITER site. It will include the supply of equipment and services as well as contributions to management and staffing.

The transition from the ITER "design" to "construction" activities will require a number of organisational and programmatic changes for EU activities. The management structure for ITER construction, the respective roles of industry and Associations will be discussed. Technological areas of highest EU interest will be described.

A meaningful accompanying R&D programme in physics and technology will have to be maintained in parallel with the realisation of ITER.

IN-1

THE IMPORTANCE OF FUSION DEVELOPMENT TOWARDS A FURURE ENERGY SOURCE

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World energy consumption will increase substantially throughout the 21st century. To dampen the increasing need for new primary energy, energy efficiency must be improved by using smarter technologies. However, the reduction of world-average energy intensity is expected to remain between 0.8 % and 1.4 % per year. Although this is a noteworthy decrease, the global energy requirements are nevertheless expected to multiply because of the growth of the global economy. Cheap availability of fossil sources oil and gas will gradually come to an end. Already today, the lack of geographic spread of oil leads to challenging price fluctuations. Because of environmental concerns, especially the anthropogenic greenhouse effect, the use of fossil sources must be diminished unless CO2-sequestration and storage are successfully developed. Only then can the substantial coal reserves be tapped. The only mature greenhouse-gas-free electricity generation source is nuclear fission. Although commercially competitive in favourable licensing and construction circumstances, and abundantly available if breeder reactors are used, the further use of nuclear fission is hampered by the public’s perception of the risks associated with the technology. Only two other routes appear to be able to contribute to the primary energy provision: renewable energy and controlled thermonuclear fusion. Some renewable sources have considerable potential. Much work is still needed, however, to make the renewable-based technologies cost effective and to integrate them into the overall energy system. Large-scale electricity storage, or a hydrogen economy (whereby fuel cells may be used to their full potential) may help in the penetration of renewable sources. The amount of the renewable contribution, however, is currently small and for the future remains uncertain. In any case, massive use of intermittent sources will be expedited by, and possibly even require, a considerable fraction of continuously operating and robust base-load electric power generation fuelled by more steady sources.

In the light of these uncertainties, the sensible approach is not an “either-or”, but an “and-and” philosophy. All three long term options, renewables, fission and fusion, should be further explored and developed so that future generations can choose the composition of an appropriate energy-source basket. It would be irresponsible towards future generations not to pursue a potentially successful energy source such as nuclear fusion.

Indeed, future fusion power plants have good prospects to qualify as economic and environmentally benign base-load electricity generation plants. The progress of fusion development has been remarkable; all available techno-scientific information shows that steady and significant progress is being made towards a successful reactor. The slow (but steady) pace of progress, however, is linked with the need for large and expensive experimental devices. In the present context of liberalising energy markets, whereby most actors focus on short time survival and profit making, and the indifference by the public at large towards science and technology development, it is not obvious to convince the decision makers to invest in a long term energy research strategy. Nevertheless, political decisiveness is required to keep the time schedule to establish commercial fusion by the middle of the century.

IN-2

EVOLUTION OF THE ITER PROJECT DURING THE CTA

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In July 2001 the Engineering Design Activities of ITER came to fruition with the issue of the ITER Final Design Report. Although this was more than sufficiently detailed for the ITER Parties to decide on where to construct ITER, and how to organise and share the procurement and costs between them, further developments have been necessary in order to prepare the project for construction to start on time should the Parties agree terms on the schedule originally planned, i.e with initialling of the Joint Implementing Agreement (JIA) at the end of 2002 in preparation for its signature/ratification in 2003.

Technical developments have arisen from the preparations for long-lead time procurement specifications by the International and Participant Teams of the Coordinated Technical Activities (CTA) underpinning the Negotiations, and from improvement of the design integrity, with a view to maintaining the highest quality of machine construction. As plans advance for the makeup of the ITER Project Team during construction, which should consist of a central team and field teams, organisational systems and procedures are being developed and installed which will be essential later. Furthermore, the preparations for licensing have proceeded as fast as has been possible, with the authorities of those Parties which are known to be considering making site proposals, given that a preferred site has not been identified as of March 2002.

The end result is that from the technical standpoint, preparations are now well advanced for ITER construction. There is more that can be done, but further steps largely require the commitment of considerable funds (for example to launch manufacturing R&D with firms already chosen for procurement), choice of site, and assignment of the personnel who will lead the project in the construction phase.

This paper gives the status of technical preparation for construction, in particular focusing on work done to complete the technical specifications for long-lead time items, or of systems significantly affected by them. It highlights where design changes have been necessary, for instance in the design of vacuum vessel supports, and where further elaboration of the design has been made to ensure its viability, as in the port cell and thermal shield designs. The paper also describes the advances made in preparations for and as a consequence of licensing discussions with the various potential hosts. Finally it describes the approach that will be taken by the project to procurement, and the resulting project organisation.

IN-3

OVERVIEW OF JET RESULTS, NEAR TERM PLANS

J. Pamela and contributors to the EFDA-JET workprogramme*

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In preparation of ITER operation the JET programme has been focussed to the consolidation of the ITER reference scenario, the ELMy H-mode, and to mature the alternative Advanced Tokamak Scenarios. In type-I ELMy H-modes simultaneously high confinement (fH98~1) and high densities (n/nGW>0.85) were achieved in stationary conditions by several techniques: (a) increasing plasma triangularity (close to ITER values), (b) application of argon seeding and (c) periodic pellet fuelling. Techniques for controlling high-pressure operation (control of Neoclassical Tearing Modes) and amelioration of transient divertor heat loads due to ELMs are presented. In Advanced Tokamak Scenarios the improvement of LHCD coupling to the plasma did lead to reversed shear plasmas with the formation of strong ITBs at lower auxiliary heating powers. Steady state conditions ($20x\tau_E$) have been achieved by successful real-time feedback schemes. Crucial topics in ITB plasmas like the impurity transport and high edge density scenarios are investigated. The characterisation of the Mk-II Gas Box Divertor with and without septum leads to a further understanding of the divertor physics. Studies of in-vessel material migration, in particular under new conditions of operation at lower ITER-like wall temperature did lead to new insights, underlining the importance of carbon erosion in the main chamber and chemical erosion in the inner divertor. Determination of the tritium inventory in selected JET tiles and flakes retrieved from JET after the DTE1 campaign contributed to close the tritium balance. Measurements confirm that a significant fraction of the tritium remaining in the machine is immobilised in flakes accumulated in the sub-divertor region. Detritiation technologies have been developed, some of which could find application on ITER. In further support to ITER a number of enhancements are foreseen. In order to extend the operational range closer to ITER (plasma parameters and plasma geometry) the heating and fuelling systems will be enhanced. By the beginning of 2003 the NBI system will be upgraded from 17.5 MW to 25 MW and in 2004 a new ITER-like ICRH antenna will be installed leading to a total heating power of ~40 MW. A new pellet system with a quasi-vertical track will be operable in late 2002. The present divertor will be upgraded in order to allow for high triangularities, approaching ITER matched shapes, at high plasma currents (3.5–4 MA). A number of diagnostics enhancements, in order to improve time and space-resolution of temperature, density, current and radiation measurements, will strengthen the physics programme in particular with respect to pedestal, ELM and ITB physics.

* See appendix in J. Pamela et al, "Overview of Recent JET Results and Future Perspectives", Fusion Energy 2000 (Proc. 18th Int. Conf. Sorrento, 2000) IAEA, Vienna (2001)

IN-4

PHYSICAL PROGRESS AND TECHNOLOGICAL DEVELOPMENT OF THE JT-60U

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The main objectives of JT-60U are to demonstrate high plasma performance integrated that contributes to establishment of the physical and technological bases of ITER and a steady state tokamak fusion reactor. Recently, performance exploration in advanced tokamak regimes has been conducted intensively, by using 500 keV negative-ion based neutral beam injection (N-NBI) and 110GHz electron cyclotron (EC) systems for plasma heating and current drive, and a repetitive centrifugal pellet injector for efficient core particle fueling. In addition, a semi-closed divertor with pumping (so called the W-shaped divertor) has been equipped for effective heat and particle control.

Following achievement of the current drive efficiency up to 1.55x 10^{19}A/m^2/W using a combination of the N-NBI and the EC heating in 2000, the enhanced current drive efficiency enabled full current drive for a high performance plasma in 2001. A high fusion triple product, $n_i(0)\tau_E T_i(0)$, of 3x10^{20} keVs/m^3, was obtained under the full non-inductive current drive by the increased N-NBI power of 5.7 MW at 402 keV. High power injection of 6.2 MW (381 keV, 1.7 s) was achieved by enhanced negative ion current with hydrogen. In addition, a long pulse injection of 10 s (355 keV, 2.6 MW) of hydrogen beam was attained through improvement of beam divergence by the correction of beamlet deflections and by optimization of operational parameters of the ion source. Spatial uniformity of the negative ion production is a critical issue in the large-scaled ion source. Several ways to obtain uniform distribution of negative ion beams have been performed. In the EC system, four gyrotrons capable of generating 4MW for 5 s have been commissioned for local EC heating and current drive. So far the maximum EC power into the torus reached 2.8 MW for 3.6 s, while the output power from a gyrotron sustained 1 MW for 5 s. The gyrotron has a chemical vapor deposition diamond window and a single stage depressed collector. Antennas can steer the four EC beams in the poloidal direction during a discharge. The EC heating in combination with lower hybrid wave injection produced a very high electron temperature of ~26 keV for a reversed shear plasma. It was demonstrated that the injected EC power of 2.9 MW generated about 1.5 MA of the EC driven current, where the temperature at the current drive location was 16 keV, and the corresponding current drive efficiency was 0.8 x10^{19}A/m^2/W. The repetitive centrifugal pellet injector has been applied to extend the high confinement mode regime to a high plasma density. The pellet injector can eject trains of up to 2.1 mm cubic pellets at frequencies of 1-10 Hz and velocities of 0.1–1.0 km/s. Pellet injection from the high-field-side (HFS) top successfully increased the plasma density up to 0.7nGW while keeping high confinement, where nGW is the Greenwald density limit. Pellet injection from the midplane in the HFS has also performed well, in which more efficient fueling is expected through the effect of ExB drift. A conceptual design of the modification of JT-60 for further economical and environmental attractiveness in a tokamak fusion reactor was completed. A fully superconducting tokamak has been designed to achieve high performance steady-state operation and to demonstrate the plasma applicability of low activation material for a break-even class plasma.

In the talk the development of fusion technology and associated enhancement of plasma performance of JT-60U will be presented.

IN-5

DIVERTOR OPERATION IN STELLARATORS: RESULTS FROM W7-AS AND IMPLICATIONS FOR FUTURE DEVICES

P. Grigull, K. McCormick, H. Renner, J. Baldzuhn, R. Burhenn, R. Brakel, H. Ehmler, Y. Feng, F. Gadelmeier, L. Giannone, D. Hartmann, D. Hildebrandt, M. Hirsch, R. Jaenicke, J. Kisslinger, T. Klinger, J. Knauer, R. König, D. Naujoks, H. Niedermeyer, E. Pasch, N. Ramasubramanian, F. Sardei, F. Wagner, A. Weller, U. Wenzel, and the W7-AS Team

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In connection with larger stellarator projects (e.g. LHD and W7-X), and in view of future stellarator reactor applications, enhanced efforts have been started to find divertor solutions which are compatible with the specific stellarator edge configurations. The paper gives an overview of concepts under discussion, summarizes briefly the physics basis and major technological aspects of the first step divertor for W7-X, which is presently under construction, and reports in detail on first proof-of-principle results from a similar divertor recently installed in W7-AS.

The W7-X divertor will utilize the flux diversion by inherent, large magnetic islands at the edge in order to establish plasma exhaust scenarios similar to those achieved in tokamak poloidal x-point divertors. The divertor consists of ten identical modules – two per field period – placed at the top and bottom of the bean-shaped plasma cross sections. Each module is composed of 3D shaped, actively cooled targets (CFC) which intersect the islands, and of baffles at the radial outside (open island divertor). The arrangement preserves large configurational flexibility and is designed to withstand up to 10 MWm^{-2} in cw-operation. The intention is to explore the reactor potential of this concept. Apart from the smaller size, the W7-AS divertor is geometrically rather similar, but technically much simpler (inertially cooled CFC targets) and less ambitious. It enables, nevertheless, access to a new NBI heated, very high density (up to $\overline{n}_e \approx 4\ 10^{20}\ m^{-3}$) operating regime with improved confinement properties (IC regime). The energy confinement time steeply increases with density and exceeds the ISS95 scaling by up to a factor of two. In contrast, the particle and impurity confinement times dramatically decrease with increasing density. The regime is characterized by flat density profiles with steep gradients at the edge. It shows similarity to ELM-free H-mode scenarios previously observed in W7-AS, but avoids impurity accumulation. These new features enable full density control and quasi steady-state operation over many confinement times also under conditions of partial detachment from the divertor targets. The impurity radiation is always peaked at the edge. Edge radiated power fractions are moderate in attached regimes and reach up to 90% in detachment scenarios. The IC regime is robust against changes of the magnetic field configuration. Stable partial detachment is observed with sufficiently large boundary islands and distances Δ_x between x-points and targets > 2.4 cm.

These results – successful operation of an island divertor in combination with the IC-mode – render excellent prospects for W7-X and advise this exhaust concept as a serious candidate also for other devices with inherent or externally induced magnetic islands at the edge.

IN-6

SUPERCONDUCTING TOKAMAK PROGRAM IN CHINA

HT-7U team presented by P. D. Weng

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The steady state operation is one of the basic requirements for future fusion reactor. In order to explore this issue, Institute of Plasma Physics started the superconducting tokamak project in 1991. The project includes two steps: the first step is to build HT-7 tokamak; second step is to construct an upgrade device HT-7U.

The first Chinese superconducting tokamak HT-7 was completed in 1994 and it has been operated since 1995. The experiment on it mainly concentrated on the topic related with long pulse operation, ICRH and LHCD. Some important modification in the HT-7 has been done last year to reduce the magnet field ripples, enhance the power and particle removal ability, improve current driven efficiency. Great progresses have been achieved after these modifications. Repeatable high performance long pulse discharges (>10s) have been achieved. Maximum electron temperature was close 3 keV in the plasma. Significant synergy between LHW and IBW has been observed. Good confinement can be obtained by proper optimization of plasma parameters and parallel index of injected LHW. Various issues on IBW heating, confinement improvement by IBW and LHCD and their edge behavior, LHCD, ICRF wall conditioning and etc are fruitfully investigated. The successful operation and experiment on HT-7 accumulated rich experience in both science and technology provide a steady basis for Chinese Superconducting tokamak project. The experiment on HT-7 will be carried over 4-5 years and the next target of HT-7 is to reach longer high performance plasma up to 30-60 Seconds.

HT-7U superconducting tokamak is currently under construction in ASIPP. The mission of HT-7U project is to develop an advanced full superconducting tokamak to establish the scientific and technological bases for an economic and continuously operation tokamak fusion reactor. It contains the following challenging physical objectives: Demonstration of Steady-state operation with high plasma performance; Investigation of advanced tokamak physics and demonstration of stationary H-mode operation by strong shaping, current profile control and auxiliary heating; Exploring of the advanced particle and heat fluxes handling on a time scale much longer than the wall equilibration time. To meet the requirement of this mission, the HT-7U design features are full superconducting magnets; enough CW non-inductive current driven and heating systems; large operation space for the advanced operation flexibility; advanced real-time plasma position and shaping controls; divertor for power and particle handling. The engineering design of the device is optimized. Most of R&D program is completed last year. CIC conductor jacketing line, two winding machines, a set of VPI equipment and a test facility for the TF and PF coils are ready in ASIPP now. The manufacture of TF CIC conductor and TF coils has begun already. The prototype of sixteenth vacuum vessel and thermal shield is fabricating in industry now. The CS model coil has been tested already.

In this paper, the HT-7 and HT-7U project, their present status and progress is described.

AO-1

NEXT STEP SPHERICAL TORUS DESIGN STUDIES*

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This paper summarizes studies which have been performed to identify and characterize a design point for a Next Step Spherical Torus (NSST) experiment. This would be a "Proof of Performance" device which would follow and build upon the successes of the National Spherical Torus Experiment (NSTX) "Proof of Principle" device which has operated at PPPL since 1999. With the disassembly and removal of the Tokamak Fusion Test Reactor (TFTR) nearly completed, the TFTR test cell and facility will soon be available for a device such as NSST. By utilizing the TFTR test cell NSST could be constructed for a relatively low cost on a short time scale. In addition, while furthering spherical torus (ST) research, this device could achieve modest fusion power gain for short pulse lengths, a significant step toward future large burning plasma devices now under discussion in the fusion community.

For the regime of interest, copper magnets with liquid nitrogen cooldown prior to pulsing provide the best possible performance. Since ST non-inductive start-up and current drive is not yet well characterized (this is a major goal of NSTX), and since an experimental device will benefit from some degree of inductive capability in any case, the initial concept includes a central solenoid (CS).

A Systems Code with physics and engineering algorithms was used to characterize the plasma and engineering behavior of NSST. This characterization includes estimation of the temperature rise and stress in the toroidal field (TF), CS, and poloidal field (PF) magnets. The overall facility electrical power and energy consumption per pulse were estimated, including the neutral beam (NB) and radio frequency (RF)systems.

The selected design point is Q = 2 at HH = 1.4, Pfusion = 60MW, 5 second pulse, with R0 = 1.525m, A = 1.6, Ip = 10MA, Bt = 2.6T, CS flux = 16 weber. Most of the research would be conducted in DD, with a limited DT campaign planned for the last year or two of the program.

The full report will summarize detailed engineering studies, including finite element analysis to develop the structural design of the device, and will address the key issues which determine the feasibility and drive the cost.

*This work is supported by USDoE Contract No. DE-AC02-76CH03073

A-1

CONSTRUCTION OF THE HL-2A TOKAMAK

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HL-2A is the first Tokamak with divertor in China which is under construction at Southwestern Institute of Physics (SWIP), Chengdu, China. The objectives of HL-2A are to produce more adaptable divertor configurations to study energy exhaust and impurity control, and to study enhanced plasma confinement by profile control and moderate plasma shaping. HL-2A has well optimized operational flexibility and excellent accessibility for the diagnostic systems to facilitate various plasma experiments. The main components of HL-2A device were transfored from IPP of Germany, installation of the machine started in October 2000 and will be finished in the middle of 2002, then comes the commissioning and operation.

The main structure installation and vacuum vessel preassembly processes have been accomplished last year; the magnetic field coils have been assembled and tested, vacuum performance experiment without vessel baking have been carried out, significant progress of construction of HL-2A have been achieved up to now. This paper introduces the subsystems of the HL-2A, The development of major components and the installation of the machine are also described, many measurement and testing results are presented.

A-2

JET ENHANCEMENTS UNDER EFDA

C. Damiani and contributors to the EFDA-JET workprogramme

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As part of the JET activities conducted under EFDA, enhancements are prepared and implemented to increase the performance capabilities of the JET facilities. The enhancements projects can be currently divided in two main categories :

- Near term projects, implemented in the 2000–2002 period
  The main objectives of these enhancements are:
  1. To increase the heating power, in particular by upgrading the Neutral Beam Injector (NBI) on Octant 8 and improving the Ion Cyclotron Resonance Heating (ICRH) and the Lower Hybrid Current Drive (LHCD) coupling capabilities.
  2. To improve the JET diagnostics capabilities, in particular the plasma edge and divertor diagnostics in order to provide information for the detailed design of the ITER divertor and core diagnostics improving the capability to develop scenarios with internal transport barriers.
  3. To improve the fuelling capabilities, particularly regarding the pellet size and penetration.
- Longer term projects, aiming to further upgrade of the JET capabilities after the 2004 shutdown
  The main objectives of these enhancements are:
  1. to consolidate the preparation of ITER operating scenarios by increasing the present operating domain of ELMy H-modes (the reference mode of operation of ITER) and further expanding the range of operating scenarios ;
  2. to support design choices in key areas of subsystems of ITER which could be finalized even after the start of construction of ITER.

  A new 8 MW ITER-like ICRH antenna and new or improved Diagnostics shall be constructed. The existing MkII-GB divertor will be modified to operate at higher triangularity and higher power. The main milestones are the following: start of the design activities July 2000; start of the procurement in industry March 2002; start of JET EP shutdown January 2004; restart of JET operation February 2005.

  Design and R&D had also been carried out in 2000–2001 for an Electron Cyclotron Resonance Heating (ECRH) system and a new divertor (Mark II HP). Although these projects could not be pursued due to budgetary constraints, the main results of the design phase will be presented.

The JET Enhancements benefit from significant contributions from non-European laboratories under international collaborations.

The paper will present an overview of the technical and scientific objectives, the description of the systems under design, the organization of the activities and the planning.

A-3

MACHINE MODIFICATION FOR ACTIVE MHD CONTROL IN RFX

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Recent theoretical studies and experimental evidences in present RFP devices and in Tokamaks call for an active control of the MHD modes to induce plasma mode rotation and to prevent the mode phase locking with the consequent non-axisymmetric configuration of the plasma column. Also the stabilisation of resistive wall modes is a key issue for future machines.

On the RFX device we have already demonstrated the possibility of inducing slow rotation of phase locked modes thus preventing wall locking. Furthermore, quasi single helicity (QSH) states characterised by one dominant helical mode have been observed. These results support the possibility of extending active MHD mode control to a variety of experimental scenarios. In the last two years an extensive analysis of the device modifications was performed and different possible design options have been considered. The design has been finalised and the new components are presently under construction.

A new first wall with optimised tapered shape of the tiles has been realised. A new system of electric and magnetic transducers will be integrated in the new first wall to characterise the plasma boundary, in particular at high frequency, with a good spatial resolution.

A close fitting 3 mm thick copper shell replaces the old 65 mm aluminium shell. The thickness is a compromise between the requirements of passive control of equilibrium and magnetic fluctuations on short timescales and of penetration of externally induced magnetic fields to control the MHD modes. A very accurate optimisation of the poloidal and equatorial gaps has been carried out to minimise the error fields that induce a braking torque on the rotating modes and a distortion of the last closed plasma surface. The shell is clamped to the vessel through stainless steel bands that sustain the electrodynamic forces arising in the shell. The vessel-shell system also required an accurate choice of electrical insulation systems for a maximum temperature of 200°C. In the very narrow space between the vessel and the shell a uniformly distributed system of thermocouples and magnetic transducers, to measure all three magnetic field components, has been designed to reconstruct the complete 3-D map of the energy deposition and of the magnetic field. The magnetic transducers will also provide the feedback signals for the active control of the MHD modes.

A new toroidal support structure made of stainless steel replaces the old shell in supporting all the electrodynamic forces acting on the torus. A new system of 192 saddle coils having a rating of 24 kAturns is provided to actively control the MHD modes. The saddle coil system completely surrounds the toroidal surface with a discretisation of four coils in the poloidal direction and forty-eight coils in the toroidal direction. This system allows the generation of $m = 0$ and $m = 1$ poloidal modes with a very clean toroidal spectrum up to $n = 24$.

The paper will present the main design features of the new toroidal assembly. The electro-magnetic analyses performed to optimise the control performances and the thermal and electromechanical calculations, as well as some specific technological developments will be presented and discussed. Finally, the present status of the construction phase will be reported.

A-4

STATUS OF TOKAMAK T-15M PROJECT

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The Engineering Design stage of the T-15M tokamak has been completed in Russia. The facility is proposed to be built within 2003 – 2007.

Main goals of the T-15M project are:

- creation of the national experimental base to be involved in solving the physical and engineering problems for ITER;
- development and testing of diagnostic systems for ITER;
- attainment of regimes necessary for ITER operation within shorter time;
- training of Russian specialists for the works on ITER;
- solution of various physical and technological problems, necessary for reliable justification of fusion reactor parameters.

The objective of the T-15M tokamak is to help to realise the ITER physics program and to solve a member of engineering problems of the ITER project.

The T-15M is an up-to-date ITER-like installation with divertor and with the possibility to control the plasma configuration. Geometrical parameters of the T-15M are scaled in proportion ¼ to ITER. The main parameters of the T-15M tokamak are presented in Table in comparison with the ITER parameters (two sets of T-15M parameters (I) and (II) are proposed to be realised in one facility).

	ITER	T-15M	
		(I)	(II)
Plasma current I_P, MA	15	1,7	1,7
Aspect ratio A	3,1	3,1	3,1
Plasma major radius R_o, m	6,2	1,55	1,55
Plasma minor radius a, m	2	0,5	0,5
Plasma shape elongation k_{95}/k_x	1,7/1,85	1,7/1,85	1,7/1,85
Plasma shape triangularity δ_{95}/δ_x	0,35/0,5	0,35/0,5	0,22/0,35
Toroidal magnetic field at plasma axis $B_{t\|Ro}$, T	5,3	2,5	2,5
Plasma current plateau duration $\Delta t_{plateau}$, s	400	2–3	2–3
Auxiliary heating power P_{AUX}, MW	40	≤ 3	15–20
Plasma configuration	SN	SN	SN

Maximal use of Nuclear Fusion Institute of RRC "Kurchatov Institute" infrastructure is supposed.

Tokamak T-15M with normal conducting electromagnetic system will be located on the site of the existing large superconducting tokamak T-15. Power supply system of T-15M facility will combine the power supply systems of T-10 and T-15 installations.

A-5

DESIGN OF THE NATIONAL COMPACT STELLARATOR EXPERIMENT (NCSX)*

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The National Compact Stellarator Experiment (NCSX) is being designed as a proof of principal test of a quasi-axisymmetric compact stellarator. This concept combines the high beta and good confinement features of an advanced tokamak with the lower current, disruption-free characteristics of a stellarator. NCSX has a 3 field period plasma configuration with an average major radius of 1.4 m, an average minor radius of 0.33 m and a toroidal magnetic field on axis of up to 2 T. The magnetic field has a nominal flattop pulse length of 0.5 s at 1.7 T. Heating is provided by up to four, 1.5 MW neutral beam injectors and provision is made to add 6 MW of ICRH. The experiment will be built at the Princeton Plasma Physics Laboratory.

The NCSX stellarator core is a complex assembly of three coil systems that surround the highly shaped plasma. The coil systems provide magnetic field for plasma shaping and position control and inductive current drive. The primary magnetic field is provided by a set of 18 modular coils. Due to symmetry, only three different types of modular coils are required. A set of 18 background toroidal field (TF) coils can be used to vary the field on axis by +/- 0.5 T. There are also six pairs of poloidal field (PF) coils. The modular, TF, and PF coils operate at 80K and are supported by an integral structural shell that also serves as the winding form for the modular coils. The vacuum vessel is built in 3 identical sections that are bolted together at assembly. Provisions are made for a complete internal liner of formed carbon fiber composite panels that are bakeable to 350ºC. The entire system is surrounded by a cryostat to permit operation of the coils at cryogenic temperature.

Design and development challenges for the stellarator core include fabrication of the highly shaped vacuum vessel, fabrication of the integral modular coil structure and winding form within a geometric tolerance requirement of +/- 1 mm, and optimization of the coil winding path, support structure and PFC systems to provide sufficient access for heating, diagnostics, and maintenance. The conceptual design review is scheduled for May of 2002, prototype coil and vessel fabrication will occur in 2003–2004, and the first plasma is anticipated in 2007.

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A-6

OPERATION OF JET AS A USER FACILITY

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Since the beginning of 2000, the Joint European Torus (JET) is used under the European Fusion Development Agreement (EFDA) as a user facility. In this framework the UKAEA as "operator" maintains and operates the facility, and installs enhancements/refurbishments, on behalf of EFDA under a contract to the European Commission. The experimental programme is proposed and implemented by the 'users', scientists and engineers on secondment from the European laboratories. These laboratories supply enhancements to be installed and operated by UKAEA, whilst the Operator supplies equipment for refurbishing the existing facilities. The overall programme is co-ordinated and managed by the EFDA Close Support Unit.

Four experimental campaigns have been completed under this new organisation, and a fifth is in progress. In the course of these campaigns, the historical rate of successful pulses has been largely maintained by the Operator, and has resulted in many publications by the users. One major shutdown has been completed during which substantial in-vessel and ex-vessel tasks have been carried out. Various small enhancements have been installed and commissioned; the first major enhancement (the upgrade of the neutral beam system) will be completed over the second half of this year.

The UKAEA also has an overriding legal responsibility for the safety and the environmental impact of the project, including the management of waste, and a long term responsibility for the decommissioning of the Facility. These responsibilities need to be reconciled with the operation of the facility.

This paper will review the Operator requirements necessary for control of the interface between the Operator and the 'users' and procedures introduced to meet these requirements. It will summarise various measures of the output from JET operation during this period, and review the experience of the Operator in the preparation and implementation of the physics programs and in implementation of EFDA enhancements and Operator refurbishments of the facility. The implications for ITER will be discussed.

This work was performed under the European Fusion Development Agreement.

A-7

THE ASSEMBLY OF W7-X – AN OVERVIEW

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WENDELSTEIN 7-X (W7-X) represents the continuation of fusion experiments of the stellarator type at Max-Planck-Institute for Plasma Physics and is being set up in Greifswald, Mecklenburg-Vorpommern, Germany. Design and assembly of W7-X are determined by the superconducting magnet system, a complicated shaped plasma vessel, 309 ports, an outer vessel with many openings, support structures and small clearances. The stellarator is made up from five symmetric modules which span a torus with an inner diameter of about 6 and an outer diameter of about 16 m. Overall height of the torus is about 5 m and the weight amounts to 725 t. Each module comprises 5 non-planar and 2 planar superconducting coils, a plasma vessel, an outer vessel, a coil support structure, about 60 ports, thermal protection and the machine support. The stellarator will be connected to a pipe system of about 20 km to provide cooling water, helium coolant and connect vacuum pumps.

The sequence of assembly is subdivided in a pre-mounting to form half-modules followed by the assembly of two half-modules to modules of the magnet system and the final assembly of the modules and closure of the torus. Massive structures and high-precision tools and mounting devices are used to position the components with an accuracy of about 10^{-4}. These devices are produced by European industry.

In a first mounting stand five different non-planar coils and two different planar coils are strung across the plasma vessel by tilting, rotation and linear translations. Positioning of the coils with a diameter of typically 3 m and a mass of 6 t is done by a computer-assisted coil handling unit. Thermal insulation onto the plasma vessel is completed successively during assembly. Next the coil assembly is screwed to a section of the coil support structure and completed with support elements between the coils. The resulting half module of the magnet system weighs about 50 tons.

In the second mounting device two half modules are connected mechanically to form a coil module. The half modules of the plasma vessel are welded together. Next the 14 coils of the module are electrically connected by means of a superconducting bus. Disconnectable joints are being developed. Finally the helium cooling lines for the coil windings, the casings and the support structure are mounted.

Assembly of the five modules comprises integration of the outer vessel, the ports and parts of the thermal protection. Precision alignment of the modules and final assembly is done on five individual mounting stands which can be moved by a rail system and hydraulic actuators. All mounting steps are accompanied by precision measurements using a laser tracker system and theodolites.

The main technical characteristics of the mounting devices and the mounting procedures are shown and explained in the presentation.

A-8

THE PHYSICS DESIGN FOR FUSION DRIVEN SUB-CRITICAL SYSTEM

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The Fusion Driven Sub-critical System (FDS) is a sub-critical nuclear energy system drive by fusion neutron source, which provides a feasible, safe, economic and highly efficient potential of disposing High Level Waste (HLW) and produce fission nuclear fuel as a early application of fusion technology. This paper reviews the past physics reactor design of fusion-fission hybrid reactor in China, and a low aspect ratio tokamak energy system has been proposed, which aims at high β, good confinement, and steady-state operation in a compact configuration at modest field.. The system includes a low aspect ratio tokamak as fusion neutron driver, a radioactivity clean nuclear power system as blanket and novel concept of liquid metal conductor as centre conductor post. Parameters of such kind reactor are following.

Major radius 1.4m, Minor radius 1m, plasma current 9.2MA, Toroidal field 2.5T, Plasma edge q = 5, Average density $1.6\times10^{20}m^{3}$, Average temperature 10keV, Plasma volume $50m^3$, Bootstrap current fraction 0.72, Fusion power 100MW, Drive power 28MW, Neutron wall loading $1.0MW/m^{-2}$.

The plasma configuration is an important part in the low-A tokamak. The Eq code has been used to getting equilibrium. By this calculation, we can found the simple PF coils satisfied the requirement of the large elongation plasma configuration, the vertical field with less curve field line in the low-A tokamak, those results for the natural elongation can be attributed mostly to differences in the current density profile. In order to determine the feasibility of the low-A tokamak operation, a transient simulation has been made which includes the equilibrium, transport and plasma position shape control in the low-A tokamak. A 1-1/2 equilibrium evolution code has been used to make this simulation. The code is two-dimensional time dependent free boundary simulation code that advances the MHD equations describing the transport time-scale evolution of an axisymmetric tokamak plasma.

A-9

ACTIVITIES ON IFMIF LITHIUM TARGET AT ENEA

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A 3-year Key Element Technology Phase (KEP) has begun in Year 2000 with the objective of reducing the key technology risk factors needed to reach the expected power handling capabilities in the IFMIF liquid Li target system with sufficient availability and reliability. The ENEA target activity deals with the following items: Li jet flow stability, Water testing, Li corrosion and compatibility, backplate remote handling simulation and Li target safety analysis.

The stability of the lithium jet is essential for the correct operation of IFMIF target. To analyse the thermal-hydraulic conditions of the liquid lithium jet for the present design requirements a numerical simulation is ongoing. The analysis is performed by means of the RIGEL code, already set up during IFMIF-Conceptual Design Analysis. For a reference beam power of 10 MW/m^2 and a beam footprint of 20×5 cm, the temperature, velocity and pressure distribution of the jet have been calculated. In particular, the evaporation rate at the free surface was found to be 16 gr/year and a free surface boiling margin of 35°C have been estimated. A negligible effect of different backwall curvature radius on the thermal hydraulical parameters has also been assessed.

Due to the intense radiation damage of the Li target backwall, the present design foresees the use of a replaceable backplate, but one of the issues of this solution is a small discontinuity in the Li jet faced surface. For this reason an experimental task is ongoing to investigate the effect of such discontinuity on the thermal hydraulical stability of a water jet with the same thermal-hydraulic parameters of the liquid lithium as estimated by hydraulic similitude. This similitude analysis has been completed and a mock up representative of the real target geometry has been designed and it is under manufacturing. The mock up is provided of a double reduced nozzle and a curved backwall and will be inserted in the ENEA Brasimone CEF1-2- water loop for testing. Possibility of Gortler vortices for instability on concave wall has been also investigated. The monitoring of cavitation occurrence close to the nozzle and the flow straighter is also part of the task and the CASBA apparatus (Cavitation and Subcooled Boiling Apparatus) has been optimized for the scope.

A conceptual remote handling procedure design has been completed for the alternative replaceable target backplate proposed by ENEA, and based on the so called bayonet concept. This concept should avoid the removal of back plate in hot-cells. The main advantages of such a solution are concerned with the design simplicity and with the possibility to change the back plate without removing the entire lithium target and also the Vertical Test Assembly. Therefore objective of this activity is to manufacture a replaceable backplate mock-up and to test the bayonet back plate remote handling. The operation of such concept will be finally experimentally verified by using the remote handling facilities at ENEA Brasimone set up for the ITER divertor remote handling.

Finally a safety analysis and deterministic evaluations during thermal hydraulic transients are going performed to verify the respect of the safety criteria for the target Li loop and all cooling circuits (primary Li loop, secondary loop with organic oil and tertiary water loop).

The relevant results of the ongoing activities will be illustrated in the paper.

A-10

LATEST DESIGN OF LIQUID LITHIUM TARGET DURING KEY ELEMENT TECHNOLOGY PHASE AND FUTURE PROSPECTS IN IFMIF

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International Fusion Materials Irradiation Facility (IFMIF), being jointly developed by EU, JA, RF and US, is a deuteron-lithium (Li) reaction neutron source for fusion materials testing. In the end of 2002, 3 year Key Element technology Phase (KEP) to reduce the key technology risk factors will be completed. This paper describes the latest design of the liquid Li target system, especially, on a reference target configuration, a Li purification system and a remote handling system, reflecting the KEP results. Future prospects will be also summarized.

To handle an averaged heat flux of 1 GW/m^2 under a continuous 10 MW D^+ beam deposition, a high-speed Li flow of 20 m/s, a double reducer nozzle and a concaved flow are applied to the target design. Thermal-hydraulic characteristics of the Li target design have been validated in a water jet experiment. Moreover, Li loop experiment is being in progress. To obtain a control scenario of the Li loop in an accident of the D^+ beam trip, a transient analysis of the Li loop has been done. With a temperature control of a heat exchanger, a solidification of the liquid Li is not observed although without the temperature control the Li solidification is observed after 380 s from the beam trip. Based on a thermal stress analysis, a routing of the Li loop pipe has been modified to reduce the thermal stress below a permissible level. To control tritium and impurities such as oxygen and nitrogen in Li, a cold trap and two kinds of hot trap are adopted in Li purification loop. Tritium concentration in Li is estimated to be 0.66 appm by an yttrium hot trap. Design value of the yttrium amount is estimated to be 140 kg assuming 20 years operation. To reduce corrosion of stainless steel by formation of ternary compound (Li_9CrN_5) less than a few tens microns/20 years, nitrogen content is controlled below 10 appm by the hot trap. Based on the KEP results, careful operation of the Li purification system is planed to avoid a degradation of the hot trap due to surface oxidation. To maintain reliable continuous operation, various diagnostics on surface waves, Li thickness, surface temperature and cavitation are attached to the target assembly. The target assembly needs to be exchanged every 9 months due to swelling of the back wall under neutron irradiation of 40 dpa. Therefore, to exchange the target assembly and back wall, a remote handling system with a multi axis arm and welding/cutting tool are designed. Three lip seals are introduced for the welding/cutting of connection flanges. This system can hang up to 1500 kg. As an option, design of a replaceable back wall with a mechanical seal is being in progress. In this concept, only the replaceable back wall is exchanged by the remote handling arm. In a next phase of IFMIF beyond 2004, a Li test loop and a remote handling system will be constructed for engineering validation. The task contents in the next phase have been defined. In summary, the development of the liquid Li target for IFMIF has been successfully performed reflecting the 3 year KEP results on analysis and experiment of the liquid Li, an impurity control and a remote handling.

A-11

LITHIUM FREE SURFACE FLOW EXPERIMENT FOR IFMIF

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Liquid lithium free surface flow experiment for IFMIF is planned to validate a high speed Li flow under vacuum condition by modifying the Lithium Loop facility at Osaka University. This project is carried under collaboration between National Institute of Fusion Science and Japan Atomic Energy Research Institute as a part of Key Element Technology Phase activities in IFMIF.

The purpose of the present study is to validate the hydrodynamic stability of Lithium flow of up to 15m/s in a open channel, as a function of cover gas pressure. In order to get this velocity, a nozzle with 70mm wide and 10mm deep was fabricated. The electromagnetic pump at this facility has the capability of driving 700 l/min of Li at 3 atm pressure drop at 300^{o}C. In order to accommodate this open test section the Li inventory of the loop was increased to 440 litter. Diagnostics are being prepared on stability of the flow surface, evaporation and splash of Li, cavitation characteristics of the pump system, and corrosion of the nozzle surface.

The test channel is being designed on the basis of the water jet experiment conducted at JAERI. The test channel with view ports is placed on the horizontal direction, because of easiness of equipment set-up and diagnostics. It is considered that orientation in the horizontal direction is much easier in operation, if very large variation in velocity from the stagnant condition to 15m/s is considered. It is estimated that effect of the gravity on the flow stability is very small. R&D’s were conducted about the fabrication accuracy of the nozzle block, and gas liquid separation in the buffer tank where the fluid flows in with very high speed. The nozzle was fabricated by the electric discharge machining and by the electron beam welding. The effective gas liquid separation is very important in order to suppress the carry under of bubbles in the small test loops like in our case, because the height difference of the apparatus is very low. The modification work of the loop will be completed by the end of June, followed by commissioning and experiment. Experiment will start on July. In the conference, the newest experimental results could be presented in detail.

A-12

DESIGN OF THE QUASI-POLOIDAL STELLARATOR EXPERIMENT (QPS)*

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The Quasi-Poloidal Stellarator (QPS), currently in the early design phase, is a low-aspect-ratio (R/a = 2.7), concept exploration experiment with a non-axisymmetric, near-poloidally-symmetric magnetic configuration. The QPS design parameters are $\langle R \rangle = 0.9$ m, $\langle a \rangle = 0.33$ m, B = 1 T, and a 1-s pulse length with 1-3 MW of ECH and ICRF heating power.

The stellarator core consists of the modular coil set that provides the primary magnetic field configuration, auxiliary coils including vertical field and toroidal field coils and an ohmic current solenoid, machine structure, and an external vacuum vessel. The modular coils represent the most difficult part of the core design and fabrication. The coil set has two field periods with 8 modular coils per period. Due to symmetry, only four different coil types are required. The baseline concept uses flexible copper cable conductor wound on a form and vacuum impregnated with epoxy. The coil form also provides the support features that allow the coils to be connected into an integral structure.

The vacuum vessel is an external “bell jar", with the modular coil set and vertical field coils in the same vacuum region as the plasma. The coils must be canned for compatibility with the vacuum, but inside the vessel there is excellent access between the modular coils for plasma diagnostics and heating. This approach has been successfully used on other experiments, such as the MAST spherical torus and the H-1 heliac.

The QPS device will be located at the Oak Ridge National Laboratory, where existing plasma-heating systems (1.2 MW ECH, 3 MW ICRF), power supplies (>40 MW), de-mineralized water system, and other equipment are available for this experiment.

The QPS device is estimated to require 4 years from start of design to first plasma in 2007. This paper describes the status of the engineering design and analysis.

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A-13

BEHAVIOUR OF GRAPHITE SAMPLE EXPOSED TO THE DIII-D DIVERTOR PLASMA IN EXTREME CONDITIONS OF CYCLIC LOADING

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Behaviour of graphite sample exposed in DIMES-DUST system to the DIII-D divertor plasma in extreme conditions of cyclic loading is analysed. The graphite sample has a slot for collect of sputtered material on a silicon disk collector. ATJ graphite sample was exposed during three seconds in each from four plasma discharges. The plasma current was ~1.4MA, neutral injection power ~10.5MW, plasma density $\sim 3–5\cdot 10^{13}$cm^{-3}, electron temperature ~4keV, graphite concentration ~2–3%, plasma density in divertor region $\sim 3–5\cdot 10^{13}$ cm^{-3}, temperature ~2eV. Maximum power on the sample surface was ~500W/cm^2, the parallel heat flux ~15kW/cm^2. The surface topography in various areas of sample and silicon collector after plasma influence has a rather various form and character. On the graphite sample surface it is possible conditionally to select nine characteristic bands and areas. As a result of intensive erosion under plasma thermal flux influence the leading edge surface represents a continuous solid destruction zone by depth from 0.8 mm and up to 2 mm in the central part opposite to slot, that exceeds the ledge size. The silicon collector surface can be conditionally divided into three bands. The average deposition thickness is ~50–60 microns, but in some areas is up to 140–150 microns. The total volume of ablated material from erosion band is ~17–18 mm^3. It is established by TDS method, that average H(D) concentration in the sample is ~0.06% for hydrogen and ~0.0004% for deuterium. Such low H(D) concentration in the deposited layer can be explained a heating of the graphite sample to temperature above 1300K during plasma discharges.

B-1

JOINING OF CFC COMPOSITES TO COPPER

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The ITER divertor design comprises the joint between CFC composites and Cu alloy heat sink. The joints must withstand some strict conditions, typical of the fusion reactor environments: operate under vacuum and cyclic thermal, mechanical and neutron loads. The CFC-Cu joint is part of the Lower Vertical Target where, during the transient events, the heat flux can reach 20 MW/m^2. Furthermore, the joints must have an adequate lifetime and reliability in order to limit the overall cost of the component. Several joining methods have been studied for CFC and copper for ITER technology: these methods foresee the use of silver-free brazing alloys, HIP assisted brazing or the copper casting on a surface modified composite (Active Metal Casting, AMC®). Most of these methods involves the use of Ti, in the brazing alloy composition or directly on the CFC's surface to form TiC, to improve the wettability of CFC.

In this work, two methods to join silicon doped CFC (CFC NS31) to pure copper are described. One method concerns the use of a commercial brazing alloy (70Ti-15Cu-15Ni). The brazing process was optimised and the shear strength of the joined samples resulted to be comparable to the interlaminar shear strength of the CFC composite. The second technique is based on the casting of copper on CFC. The CFC surface was modified by direct reaction with a transition metal. The obtained modified CFC resulted very well wettable by molten copper. The morphological analysis of the CFC-Cu samples was performed.

B-2

EFFECT OF THE DECREASE OF THERMAL CONDUCTIVITY IN MATERIALS FOR THE ITER HIGH HEAT FLUX COMPONENTS ON THEIR STRESS-STRAIN CONDITION

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High heat load on a number of in-vessel components is a typical feature of fusion reactors. Heat sink components require the application of materials possessing high thermal conductivity (W, Be, Cu) that have not been widely used in fission reactors. This study considers thermal conductivity change as primary factors leading to degradation of PFC and heat sink materials during manufacturing and operation.

In this study on the basis of earlier measurements of thermal conductivity of PFC and heat sink materials (the materials were investigated both after impact of manufacturing cycle and after the neutron irradiation in the SM-2 and BOR-60 reactors) the changes in thermal conductivity of these materials were calculated with reference to operational conditions and ITER neutron spectrum.

The consideration of different materials used in PFC, i.e. Be, W, W-Re, reveals that radiation damage and accumulation of transmutants (He, Li in Be, and Re and Os in W) are their principal factors. Thermal conductivity when irradiated to 1 dpa reduces by about 10%.

Pure copper and copper alloys (Cu-Cr-Zr, GlidCopAl25) demonstrate decrease of thermal conductivity by 20% at 5 dpa due to accumulation of transmutants Ni and Zn. Besides, the HIP procedure may cause the double reduction of thermal conductivity of Cu-Cr-Zr. An additional problem is created by the junction zone of Cu//SS – due to diffusion of Cr and Fe in Cu (during hot isostatic pressing (HIP)) thermal conductivity in the zone reduces by 20 to 30 %. For the 316LN steel that has low thermal conductivity its change resulted from irradiation and other factors (HIP) is minimal.

Analysis of calculated results on reduction of thermal conductivity of materials resulted from both irradiation and manufacturing allows to rank them according to the degree of thermal strength factor: $M=\sigma\lambda(1-\nu)/\alpha E$. On the basis of the performed assessments of thermal conductivity reduction calculations of stress-strain condition in different zones of W/Cu/SS joints were performed using the finite element method at a given level of thermal conductivity reduction that corresponds to some neutron irradiation dose. The calculations show that the maximum increase of stresses is observed in the zone adjacent to the Cu/SS joint line from the copper side. This is controlled by very high local drop of thermal conductivity in this zone due to doping by the elements diffusing from the steel.

The conclusion is made that reduction in thermal conductivity results in substantial growth of stresses in ITER components and may play significant role in the reduction of life time of these components.

B-3

MOLECULAR DYNAMICS SIMULATION OF DEUTERIUM AND TRITIUM DIFFUSION IN TUNGSTEN

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Erosion by sputtering is one of the critical issues for the use of tungsten divertors in future tokamaks. Corrosive impurities, for instance oxygen, coming to the tungsten surface with the scrape-off-layer deuterium-tritium (DT) plasma may deteriorate the erosion performance. The removal of oxygen due to the surface reactions $2H + O \rightarrow H_2O$ with H = D and T may mitigate the corrosion effect. Therefore the surface density of DT atoms under DT bombardment is important.

To investigate this process the molecular dynamics simulation code 'CADAC' ('Classical Atomic Dynamics Analysis Code'), now under development, is applied. The processes in the subsurface layer of size of 20 A heated at 1500 K are simulated with about 10^3 tungsten atoms and a few deuterons or tritons during time intervals of 10^{-11}–10^{-10} s. A new numerical instrument: the time cluster solver is introduced into CADAC allowing a significant acceleration of the calculations, especially for assemblies of atoms of large mass ratio, keeping exact conservation of momentum and fairly good conservation of energy.

In order to obtain the DT concentration behind the divertor surface the calculated diffusion coefficient, the slowing down time and other kinetic parameters are finally used in an equation for the DT energy distribution function. The surface reactions are modeled in the frame of the embedded-atom pair-potential approach for the interatomic forces. Slowing down of the incident DT atoms with energies below the W sputtering threshold is due to electron-hole pair emission and elastic energy exchange with the moving lattice. First erosion results, including direct W sputtering due to oxygen impact, are discussed based on the reaction kinetic equations.

B-4

FURTHER DEVELOPMENT OF CAST SS/CuCrZr JOINING METHOD FOR ITER

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Presented work continues investigation of physical and mechanical properties, thermal fatigue performance of SS/CuCrZr joint produced by casting and studies new possible applications of mentioned method to produce plasma facing components (PFC) in ITER.

Paper gives evaluated thermal cyclic test results of the first wall (FW) mock-up produced by casting (5000 cycles at 1–1.5 MW/m^2 on Be armour, 6000 cycles at from 4 to 8 MW/m^2 on un-armoured part). These results look promising to propose casting to other PFC areas (divertor dome and port-limiter) with moderated heat flux. In respect to check such possibilities another two mock-ups armoured with Be and W tiles were manufactured and tested. During the thermal fatigue test the mock-up with W tile (44 mm × 44 mm × 4 mm thick castellated in four parts 21 mm × 21 mm) successfully withstood more than 4000 cycles at heat flux from 3 up to 7.0–7.5 MW/m^2. After testing a partial delaminating of a rolled W tile was found but no defects in a cast CuCrZr. The second mock-up with different size and thickness Be tiles was subjected to 2000 thermal cycles at 7.2–7.5 MW/m^2. In general, all Be tiles withstood applied heat flux. No defects were also occurred in a cast heat-sink. To check a quality of a bulk cast CuCrZr the tested mock-ups were cut and metallography of their cross-sections was done. The results are presented and discussed.

Collecting new data on properties of cast CuCrZr after neutron irradiation paper presents tensile curves of cast CuCrZr and SS/CuCrZr joint samples irradiated in nuclear reactor up to 0.15 dpa at about 200^{o}C as well as the results of measurement of a thermal conductivity of mentioned materials before and after a neutron irradiation. Optimising a heat treatment regime authors present the results of thermal conductivity measured in the temperature interval 20–450^{o}C of cast CuCrZr having different heat treatment history. Paper also gives the results of comparative measurements of K_{1C} values for conventional and cast CuCrZr at RT and 300^{o}C.

Evaluation of recent experimental results, further testing plans as well as advantages of mentioned method to manufacture components having a complex form are presented and discussed.

B-5

MICROMECHANICAL STUDY OF PLASMA-SPRAYED BORON CARBIDE COATINGS FOR THE USE IN FUSION EXPERIMENTS

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Low Z-materials like boron carbide are promising candidates for the use as protective coatings for plasma facing components of fusion experiments, in particular for stellarators with complicated wall geometries. Especially plasma-sprayed B_4C with a well defined porous, crack containing, lamellar structure provides good fusion plasma erosion resistance. Furthermore, such structures exhibit adequate fracture and thermal shock resistance due to their low mechanical properties like strength and elastic modulus. During the spraying process of 3 dimensionally shaped parts, different spray angles will have an influence on the structure and the mechanical properties like hardness and Young's modulus of such B_4C coatings.

In the present study multilayer systems (MLS) were fabricated by vacuum plasma spraying. On grit-blasted stainless steel substrates 40 µm thick nickel-chromium bond coats and subsequently 200 µm thick B_4C coatings were sprayed. The spray angle was varied from normal incidence (perpendicular to the substrate) to 60° with respect to the substrate normal.

The effects of the spraying angle on microstructure, hardness and elastic modulus of the B_4C coatings were investigated at room temperature. The MLS were examined for their load-displacement behavior by means of depth sensitive indention parallel and perpendicular to the surface. In addition, four-point bending tests were carried out to determine the deformation behavior on the top of the B_4C coatings. The surface strain was evaluated by optical microscopy.

Hardness and Young's modulus of all coatings were lower than the respective value of the bulk material, due to porosity and lamellar structure. Further, all the B_4C layers exhibited higher hardness and Young's modulus values perpendicular to the surface during depth sensitive indentation. The underlying mechanisms of this difference are discussed from the viewpoint of anisotropy and variations in microstructure.

B-6

DEVELOPMENT OF W/Cu-FUNCTIONALLY GRADED MATERIALS

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The Tokamak is the best investigated alternative to fulfil the needs of future fusion reactors. The realization of an experimental fusion-device like ITER-FEAT (International Thermonuclear Experimental Reactor – Fusion Energy Advanced Tokamak) is connected to high standards in theoretical understanding and technology. One of the most critical issues is the development of plasma facing components which will be exposed to thermal heat loads in the range of 5 to 20MWm^{-2}. In this field this paper is situated with the aim to develop an alternative design option for the divertor.

The new concept deals with a functionally graded structure (FGM) between the plasma-facing material (W or W-alloy) and the heat sink (Cu-alloy). Both materials are free from mutual chemical interactions and exhibit different material properties. Due to the differences in the coefficient of thermal expansion ($\alpha_{Cu} \approx 4\alpha_W$) and Young's modulus ($E_{Cu} \approx 0.2E_W$) the joining of these two materials result in high thermal stresses in the interface, when exposed to high heat loads. In addition these stresses may lead to cracking and delamination, which finally results in reduced lifetime of the components. Differences in melting temperature ($\Delta T \approx 2400$) and density ($\Delta\rho \approx 11 \cdot 10^3\, kgm^{-3}$) pose high demands on fabrication techniques.

The present study is focused on two metallurgical processes: laser sintering by the "blown powder process" and plasma spraying. For both fabrication techniques the behavior of different powders was investigated for a mixture W-25vol%Cu. Results of thermo-mechanical, thermo-physical and non-destructive tests are discussed in respect to different microstructures. Limitations and advantages of each method are outlined from the viewpoint of anisotropy in material properties, interfacial bonding and melting behavior of W and Cu during the deposition process.

B-7

TUNGSTEN COATING ON JET Mk IIGB-SRP DIVERTOR TILES

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Erosion and deposition at the JET vessel walls and divertor have been investigated over the last years by means of various coating structures. This work is still ongoing and a new set of CFC tiles with tungsten stripes was installed in the divertor during the 2001 shutdown at JET.

Six divertor tiles were coated with tungsten marker stripes using the DIARC® plasma coating method. The thickness of the coating was measured with profilometry and Rutherford Back-scattering Spectroscopy (RBS). The thickness profile of the deposited layer was determined by measuring the thickness of a tungsten layer grown simultaneously on a silicon wafer by profilometry. On the 2 cm wide marker stripe, the thickness varied ± 0.1 µm and the average thickness was 3.4 µm. The thicknesses measured directly on the coated tiles by RBS were in excellent agreement with the profilometry results. The contents of major impurities were determined using Time-of-Flight Elastic Recoil Detection Analysis (TOF-ERDA), Particle Induced X-ray Emission (PIXE), RBS and Secondary Ion Mass Spectrometry (SIMS). The impurities found were O, Fe, Ni and Cr in concentrations typically less than 1 at.%. Thickness and impurity measurements were performed prior to mounting in the vessel.

The endurance of the tungsten coatings under severe conditions was tested in the Neutral Beam Test-bed at JET. For testing purposes, a block of CFC material 200 mm × 150 mm × 50 mm (which is approximately half the area of a divertor tile) was coated with a tungsten layer of approximately 3 µm thickness. The tungsten coated block was exposed to a total of 113 pulses with a maximum peak power density of 15.7 MWm^{-2}. At 15.7 MWm^{-2} maximum surface temperature went up to 1280°C. Visual inspection of the coating revealed no obvious changes as a result of thermal cycling. No sputtering of the film by deuterium ions could be detected either. RBS spectra along the sample, taken before and after the NB tests were compared and found to be identical.

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B-8

MECHANISM OF BLISTER FORMATION ON TUNGSTEN SURFACE

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Next-step of fusion reactors will use various kinds of materials for PFM. For example, in the case of the International Thermo-nuclear Experimental Reactors (ITER), Beryllium will be used for the first walls and carbon and tungsten for the PFM of the divertors [1]. By using these materials simultaneously, cross-contamination takes place at all surface, which would affect plasma surface interaction, not only erosion, deposition, gas retention but also surface deformation such as blister formation. Pure hydrogen isotope ion irradiation into tungsten has been performed with ion beam sources or plasma simulators, and the blisters formation was observed [2–3]. But no data regarding the effects of impurities on blister formation is available. Therefore, intensive studies focusing on this issue are necessary in order to evaluate tungsten performance correctly as PFM.

Lately, the experiment of carbon-hydrogen mixed beam irradiation to tungsten material has been performed with the High Flux Ion Test Device (HiFIT) [4]. In the case that carbon concentration of 0.95%, large number and various size of blisters (up to 1.0 mm) were formed, while in the case that low carbon concentration of 0.11%, no significant blisters were formed on the sample. From this result, it was revealed for the first time that small carbon impurity concentration in the beam played an important role in large blister formation. But the mechanism of blister formation by carbon impurity has not been clear yet, and the experiment must be done in more detail.

In this study, we have been examining the effects of beam fluence, energy, surface roughness and crystallization of tungsten on blister formation. It was found that the blisters already appeared with fluence of 1×10^{24} m^{-2} and with increasing fluence to 7×10^{24} m^{-2}, the blisters became larger and accumulated. The experiments with the fluence less than the order of 10^{24} m^{-2} and more than 10^{25} m^{-2} will be made to clarify the critical value of the fluence for the blisters formation, the change of blister shape with increasing fluence over the range of 10^{25} m^{-2}.

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B-9

BUBBLES AND SWELLING IN TUNGSTEN ARMOUR

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Tungsten (W) is currently considered to be a prime candidate material for plasma-facing armour and divertor targets in a D-T fusion tokamak power plant. In ITER, this high-Z metal has been selected as the armour material in the divertor baffle region where sputtering and erosion is a critical issue. In ITER's D-T phase, a move to all-tungsten armoured targets is planned to reduce the large tritium co-deposited inventory that would accumulate if, instead, graphitic surfaces were to be used. Recent experiments studying the effect of using W in the plasma divertor of the ASDEX Upgrade tokamak showed no degradation of plasma performance compared with conventional low-Z graphitic structures. The main advantage of using a high-Z material such as tungsten is its resistance to erosion by sputtering from light, low energy (< 500 eV) plasma ions. This resistance stems from its great cohesive strength. This same strength is expected to convey other advantages, notably resistance to swelling from helium (and hydrogen) bubbles formed by (n, α) and (n, p) transmutation reactions resulting from 14 MeV neutron irradiation from D-T reactions.

This paper presents calculations of the swelling produced in fusion power plant tungsten components. The working temperature, T, of such components is assumed to be about 1000K, with a service lifetime about 10^8 s (≈3 years). In the bulk tungsten crystal, the swelling of voids and helium or hydrogen gas bubbles can occur because these defects acquire vacancies from the surrounding crystal lattice. Vacancies are produced by knock-on processes. In the case of W at T = 1000K, thermal supply of vacancies in the lattice is negligible, leaving only the surplus knock-on vacancies to drive swelling. This bias-driven process is considered here. However, in the grain boundaries, there is a ready source (and sink) of mobile vacancies and the operating temperature can no longer be considered to be low. As a result, gas bubbles on the boundary will rapidly adjust their volumes, by emitting or absorbing vacancies, in order to relax to mechanical equilibrium. The swelling conditions on a boundary are thus very different from those inside the grains. The general conclusion of the calculations is that the swelling in both cases will be small, giving a volume change, $\Delta V/V$, of order 1% in the plant service lifetime of the material.

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B-10

THE EFFECT OF HELIUM GENERATION AND IRRADIATION TEMPERATURE ON TRITIUM RELEASE FROM NEUTRON IRRADIATED BERYLLIUM

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The effect of neutron irradiation condition on tritium release from beryllium is described in this paper. Beryllium samples were irradiated in SM reactor with neutron fluence (E > 0.1 MeV) (0.37-2.0) × 10^{22} cm^{-2} at 70–100°C and 650–700°C. Mass-spectrometry technique was used in out of pile tritium release experiments during stepped-temperature anneal within 250-1300°C temperature range. The total amount of helium accumulated in irradiation beryllium samples varied from 600 appm to 3050 appm.

The first signs of tritium release detected at temperature of 400–450°C. It was shown that irradiation temperature and helium generation level significantly affect on tritium release. The 60–100% of tritium contented in samples after low temperature irradiation (70–100°C) released from beryllium at annealing temperature up to 800°C, while that for samples after high temperature irradiation (650–700°C) was only 14%. The most part of tritium release (65–70%) released within 800–920°C temperature range. The increase of helium generation from 600 appm to 3050 appm results in lowering the temperature of maximal rate of tritium release from beryllium and the upper temperature of tritium release by 100–130°C and 200–250°C, correspondingly. On the basis of data obtained, the diffusion coefficients of tritium in beryllium were calculated.

B-11

THE PROCESSING OF VACUUM PLASMA-SPRAYED TUNGSTEN-COPPER COMPOSITE COATINGS FOR HIGH HEAT FLUX COMPONENTS

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Tungsten is the material of choice for the first wall and divertor components in ITER and is or will be applied in different other fusion experiments. The transition from the first wall material tungsten to heat sink materials made of copper alloys might be achieved by a gradient composition of tungsten/ copper FGM to avoid spallation.

In this study, the influence of different vacuum plasma spraying conditions was investigated to deposit a composite tungsten-copper coating. Three miscellaneous powder conditions were used: copper coated tungsten powder (d_{50} = 55 µm W + 20 µm Cu), a pre-mixed powder of tungsten (75 wt%, d_{50} = 5.5 µm) and copper (25 wt%, d_{50} = 35 µm) and the same initial powders of the pre-mixed powder, but not mixed until injected separately in the plasma plume. Additionally, the influence of transferred arc cleaning (TAC) is studied in all three cases.

The effects of the various spraying conditions on microstructure, porosity, chemical composition, hardness and elastic modulus of the W-Cu composites were analyzed by optical microscopy of cross-sections, image processing for porosity estimation, chemical analysis and micro indentation. Furthermore thermophysical properties will be presented.

The coatings made of the coated powder either with or without TAC show a copper rich phase with inclusions of tungsten being probably not melted during the process. In contrast, the production routes using powders of tungsten and copper show a homogenous structure with different amounts of pores and copper concentrations. Especially the use of TAC reduces the porosity and leads to a better distribution of copper. The mixing of the powders inside the plasma torch was used to deposit graded structures to get a very smooth transition between the top layer of tungsten and the heat sink material.

B-12

CORRELATION OF DEUTERIUM RETENTION WITH ANNEALING EFFECTS OF DAMAGES IN BORON THIN FILM

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Because of the wall-conditioning leading to remarkable improved the plasma performance, it have been performed on a lot of fusion test device. In situ boronization is one of the most useful and effective wall-conditioning methods for the purpose of reduction of impurity levels and strong oxygen gettering. Although the boronization has been reported the remarkable ability for reduction of impurity levels and strong oxygen gettering, the properties of hydrogen retention have not been elucidated in detail. From the safety point of view for D-T fusion reactors, we focused on tritium inventory in boron thin film. In present study, the correlation of deuterium retention with annealing effects of damages was investigated by means of X-ray photoelectron spectroscopy (XPS) and thermal desorption spectroscopy (TDS).

Boron thin films were deposited on the isotropic graphite (IG-430U, Toyotanso Co. Ltd.) with DC glow discharge in mixture gas of $B_{10}H_{14}$ and He. The boron thin films were characterized by XPS measurement of the depth profile using Ar^+ sputtering of constituent elements After heating treatment, D_2^+ implantation temperature was varied in the range from 173 K to 863 K. It's condition was 1.0 keV D_2^+ and 1.0×10^{22} D^+m^{-2} in both experiment. After D_2^+ implantation, XPS measurement was carried out after sample temperature reached to ambient temperature. In TDS measurement, the sample was heated with ramp rate was 0.5 K s^{-1}.

The XPS measurement of the prepared boron thin film exhibited that it consisted of not only B but also C, O, and N. And also it indicated that the prepared boron thin film was almost governed B and C in boron thin film. From the results of XPS measurement for the depth profile, the atom ratio of B to C was determined to be 0.9 on the surface and 2.6 in the boron thin film. From the results of TDS experiments, D_2, BD_3, and CD_4 release spectra were observed. In TDS spectrum after D_2^+ implantation, D_2 and BD_3 were released in the temperature region from almost implantation temperature to 1100 K. However, CD_4 release was observed above 600 K. Therefore, it revealed that implanted D into the prepared boron thin film was trapped by B and C forming B-D and C-D bond. In the case of D_2^+ implantation at 873 K, the deuterium retention was less than one tenth compared with that at 173 K. That was the reason why the annealing effects of damages were induced during D_2^+ implantation. In the present study, correlation of deuterium retention with annealing effects of damages is discussed to take account of XPS and TDS results in detail.

B-13

TUNGSTEN PLASMA FACING COMPONENTS IN ASDEX UPGRADE

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In most current fusion experiments carbon based plasma facing components are used mainly due to the good thermomechanical properties of carbon materials and the comparatively small radiative losses inflicted by carbon as plasma impurity. Large scale use of carbon in a fusion device operating with tritium seems not possible due to the formation of prohibitively high tritium inventories by codeposition. In ASDEX Upgrade tungsten has been used to study the feasibility of a metallic first wall for suppression of carbon sources and to prove the suitability of tungsten as high-Z alternative to beryllium used as main chamber wall material in the current ITER design.

In recent experimental campaigns, the graphite tiles at the central column heat shield were replaced in several steps by tungsten coated tiles, starting with two toroidal tile belts in 1999 to full coverage in 2002. The campaign averaged tungsten erosion flux was determined ex-situ by measuring the remaining thickness of the W-layers after the experimental campaigns by means of Rutherford backscattering and sputter Auger electron spectroscopy and comparing this to the corresponding thickness of the original layer. The observed lateral variation and the total amount of eroded tungsten, which significantly exceeds results from previously installed W test coatings, show that erosion by ion impact is the predominant tungsten erosion channel. From carbon deposits found on top of the W-layer and from the low plasma edge temperature one can infer that the erosion is mainly due to impact of carbon ions. The large scale variation of the tungsten erosion is compared to the pattern of scrape-off layer flux surfaces intersecting the surface of the inner column tiles.

Langmuir probes installed at the central column show that particularly at the central part of the inner column the plasma ion flux during the ramp down phase of discharges significantly exceeds the values during the flat top phase. This indicates that a considerable fraction of the observed tungsten erosion might not contribute to the measured tungsten concentration in the confined plasma, which was found to be generally of the order 10^{-6}, well below the maximal limit of 2×10^{-5} for ITER. Concentrations above this limit were only observed under special discharge conditions leading to W accumulation. In these cases it could be shown that W accumulation can be avoided and even reverted by moderate central heating using ICRH or ECRH.

Migration and redeposition of eroded tungsten were investigated mainly by post-campaign analysis of tungsten deposition on long term collector probes and on wall and divertor tile samples. The spatial distribution of redeposited tungsten at plasma facing components indicates that a significant fraction of the eroded tungsten migrates along direct transport channels in the outer region of the plasma scrape-off layer without penetrating the confined plasma. This observation is in good agreement with the small W-penetration probability measured spectroscopically in discharges with tungsten laser ablation.

B-14

STRUCTURAL REFINEMENT OF CHROMIUM BY SEVERE PLASTIC DEFORMATION

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Since chromium is superior to most materials with regard to the low neutron-induced radioactivity, it is considered as material in fusion technology. Limitations of the application in industrial design are the low ductility at room temperature and a ductile to brittle transition temperature which lies significantly above the room temperature. In the last years intense research has started to produce nanostructured materials by severe plastic deformation. Compared with the undeformed materials these materials with grain sizes clearly smaller than 1 µm are distinguished by an increase in strengths without loosing ductility.

In the present study pure chromium (99.97%) has been deformed by high pressure torsion. The applied pressure in all deformation processes has been 7.8 GPa. The deformation has been performed between 320°C and 390°C which lies clearly above the ductile to brittle transition temperature of the material in its recrystallized condition and at room temperature. Specimens with different degrees of deformation have been produced. A reduction of grain size from about 80 µm to microstructures with a typical grain sizes between 50 nm and about 500 nm have been found by examinations in the scanning electron microscope by means of back scattered electron images and orientation imaging microscope analysis. The thermal stabilities of the microstructures have been investigated by annealing the samples in vacuum for 10 hours at 500°C, 600°C, and 700°C. A coarsening of the grain structure has been observed (especially at 600°C and 700°C) depending on the initial structure but even after the heat treatment at 700°C a fine grain size of 3–5 µm prevails in all types of microstructure.

Furthermore micro hardness measurements have been carried out. The hardness after an applied degree of deformation of 25.8 (von Mises) at room temperature of the deformed sample has been about 4 times higher than the hardness of the undeformed material. After the heat treatment at 700°C the hardness has been still about two times higher.

It has been possible to produce nanostructured chromium by high pressure torsion. A large increase of hardness has been obtained which prevails up to 700°C.

B-15

EFFECT OF THE NEUTRON IRRADIATION TO 2 DPA ON TENSILE PROPERTIES AND LOW-CYCLE FATIGUE OF Cu//SS JOINTS MANUFACTURED BY DIFFERENT HIP TECHNOLOGIES

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In the structures of the divertor and first wall of the ITER a bi-layer structure of the Cu/SS type is supposed to be in use for the heat sink system. Hot isostatic pressing (HIP) is assumed to be the main method of joining. This study presents the investigation results for joints of GlidCopAl25/316 and Cu-Cr-Zr/316 type, manufactured using the HIP procedure in Japan, EU and RF. Tensile and LCF testing were performed on these joints. Specimens were irradiated to 0.2, 0.4 and 2 dpa at 150, 200 and 300°C in the SM-2 reactor (SRIAR, Dimitrovgrad, Russia).

Irradiation at 150°C and 200°C was shown to lead to hardening and embrittlement of joints of the GlidCop/SS type. For all three types of joints (RF, EU, JA) the level of σ_u after irradiation is nearly the same, i.e. ~340–370 MPa. The ductility characteristics of different irradiated joints vary over a wider range. Thus, EU joint fractures at zero deformation; RF joints have an elongation of about 0.2%; JA joints, when irradiated, have a rather high level of total elongation of ~3%. The Cu-Cr-Zr/316 joint has, when irradiated, a somewhat lower strength level σ_u (~230 MPa), but demonstrates a high (~16%) ductility. The LCF life-time of GlidCop/SS type joints also reduces. LCF life-time of Cu-Cr-Zr/SS after irradiation reveals slight changes.

Irradiation at 300°C does not result in substantial changes of strength properties of the joints, but the ductility of GlidCop/SS joints is very low. Cu-Cr-Zr/SS joints have satisfactory high strength and ductility properties. When testing at 300°C the LCF life-time of all the joints is lower than at 150°C in both unirradiated and irradiated states.

SEM investigations of irradiated specimens showed that in general the materials reveal the ductile fracture but the GlidCop/SS type joints at $T_{test} = T_{irr} = 300°C$ demonstrate the tendency to the brittle intergranular fracture.

The comparison of three methods (JA, EU, RF) of HIP joints of GlidCop/316 shows that the Japanese one has somewhat higher level of tensile and LCF properties over the others. Performed analysis of distributed stresses and deformations in the Cu//SS joints using the finite element method shows that the observed radiation embrittlement in GlidCop/SS primarily caused by concentration of stresses and plastic deformation on the brazing zone in the GlidCop part. The performed analysis allows concluding that the joints of Cu-Cr-Zr/316 type due to their high ductile properties are more promising for heat sink components that undergo high thermal load.

B-16

3-D SIMULATION OF MACROSCOPIC EROSION OF CFC UNDER ITER-FEAT OFF-NORMAL HEAT LOADS

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Because of their high heat conductivity carbon fiber composites (CFC) are considered as armour material for the vertical target in the divertor strike point region where high heat loads are expected during off-normal heat loads. The newly developed CFCs have a complex 3-D framework from ex-PAN and ex-pitch carbon fibers. The framework is filled with a carbon matrix. Both fibers are anisotropic materials. The linear thermal expansion coefficient, the heat conductivity and the Young's modulus along and across the fiber direction are 10–50 times different. This anisotropy and the difference of the linear thermal expansion coefficient of the matrix and the fibers cause larger internal thermostressses in CFC as compared to graphite. The thermostress concentrates at the interfaces between fibers and matrix, especially close to the sites of the perpendicular intersection of the fibers. There is experimental evidence of formation of large macroscopic pits at such intersections under high heat loads [1] and if those pits combine there occurs large macroscopic erosion by brittle destruction [2]. Indeed first experimental results for CFC samples heated by quasistationary surface heat loads to temperatures of about 3050 K have shown rather large macroscopic erosion by brittle destruction of fibers, parallel to the CFC surface [3].

For the simulation of the thermomechanical properties of the CFC materials an existing 3-D lattice model previously used for the simulation of fine grain graphite has been modified to reproduce the CFC structure with fibers and matrix. Simulation of CFC erosion under high surface heat loads, typical for transients and off-normal regimes has been performed. The 3-D numerical simulations show that brittle destruction of CFC occurs under such heat loads and might result in large macroscopic erosion of the CFC armour. As a consequence the lifetime of the CFC armour is drastically reduced and considerable amounts of dust are produced.

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B-17

ANALYSIS OF HIGH HEAT FLUX TESTING OF MOCK-UPS

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EU ITER Participating Team is performing a large R&D effort in support of the development of High Heat Flux Components (HHFC) for ITER. In this framework, the activities described in this paper consist in the analysis (thermal, lifetime evaluation and transient VDE including material melting) of the high heat flux thermomechanical tests performed in the JUDITH facility of several mock-ups corresponding to different proposed HHFC designs: Be flat tile (with 3 various Be thickness), W-macrobrush, W-monoblock and CFC flat tile. For the two last mock-ups, the VDE behaviour has not been performed.

The four types of mock-ups were swirl tube concepts (tape thickness 1 mm and twist ratio 4). The hydraulics conditions, used during the tests and considered in the analyses, were roughly the same for each mock-up: water velocity of about 13 m/s corresponding to flow rates of about 60 l/mn, pressure of 3 MPa and inlet temperature at ambient. The heat transfer coefficient in the FW water cooling tubes takes into account the sub-cooled boiling regime. Thermal steady state and transient analyses have been performed taking into account the various input loads: 5 MW/m² for the Be flat tile mock-ups, up to 10.2 MW/m² for the CFC flat tile, and up to 18 MW/m² for the W mock-ups. For the VDE simulations, the incident energy was in each case 60 MJ/m² but with different shot duration: 0.1 s and 0.3 s, different heated surfaces, and different assumptions concerning the absorbed power (only 55% of the input power is absorbed by the mock-up in case of W and 90 % in case of Be). The incident fluxes taken into account in the calculations are therefore varying between 140 MW/m² and 620 MW/m². The radiation + evaporation cooling which occurs at the surface of the tiles at high temperature is taken into account in the calculations, as well as the melting of the surface materials, computed with specific thermal model for $W1\%La_2O_3$ and Be. Moreover, to take into account the fact that convective motions tend to homogenize the melt layer temperature, the surface material thermal conductivities have been artificially increased to high values above the melting point. The fatigue lifetime evaluation is based on the level of strain variation obtain in the copper-alloy heat sink for the highest flux transient, computed with elastoplastic analyses using the generalised plane strain mode (a plastic behaviour with kinematic hardening model is considered for the OFHC, the other materials are assumed elastic).

This comparison experiments/analyses of tested mock-ups is useful for the interpretation of the phenomena occurring in the tests and will help in the prediction of the thermal fatigue behavior of the plasma facing components during ITER operation. Moreover, the quite good agreement in the results shows that it is possible to simulate correctly the high heat flux testing.

* SOCOTEC Industries

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ACTIVELY COOLED HIGH-INTENSITY HEAT SHIELD (FORM LOCKED) DESIGN ANALYSIS

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To operate in the high intensity long pulsed/steady-state conditions envisaged for present and envisaged nuclear fusion experiments, actively cooled heat shields are under development. There are metallurgical bonding (brazing, hot isostatic pressing, etc.) and non-brazed types. In the former, heat shield tiles, normally carbon fibre composites (CFC), are bonded directly to the cooling tubes (active cooling). The metallurgical bonding concepts require high quality standards, as the range of acceptable parameters is narrow. Also significant shear stress is induced at the bonded interface during thermal deformations, which can be a source of subsequent failure.

An alternative, is the non-brazed form locked monoblock concept. A monoblock tile contains a semi-closed circular bore. The tiles are assembled by sliding them over cooling tubes. For working under high intensity heat loads, the effective wall thickness of a CFC tile is made thin. To retain an active form locking and to create an initial contact pressure, each monoblock tile is clamped on to the tube at the rear part (opposite to the plasma-facing side) with a special spring-bolt joint. The contact pressure within the form locked monoblock is then further enhanced due to the thermal expansion of the cooling tube and of inward thermal deformation of the tile - thus creating an active cooling system. Shear stress is negligible at the tile/tube interface.

A prototype of the form locked heat shield structure underwent its first high heat flux tests for divertor applications in the Jülich MARION ion source and material test stand. To take up a steady-state surface heat flux above 10 MW/m², the effective wall thickness of the CFC tile (SEP N11) was reduced to 7 mm. A cooling tube (CuCrZr) was coated with pure copper (OFHC) to promote effective contact with the tile. Water was used as coolant. The results, obtained from cyclic and screening tests, are encouraging. Parallel to the tests, a 2D-finite element thermomechanical analysis was performed. Since the form locked monoblock tile in the prototype almost fully encloses the cooling tube, the analysis shows a high hoop stress in CFC tile at the point of contact with the tube in the front part.

This paper deals with a feasibility study of a modified design of the above prototype monoblock tile using a 2D-finite element model. The elements at the rear part of the tile are so re-shaped, that this gives a free space for the tube to expand at the rear region. The cooling water at 100°C and with 4 MPa pressure is identical to ITER's hydraulic parameters for divertor designs. Geometrical boundary conditions and the overall shape of the non-brazed heat shield resembles that of ITER's bonded divertor design. A thermomechanical analysis of the modified form locked monoblock design shows a significant reduction of the hoop stress in the CFC tile. Thermal and stress analyses results are discussed in details.

An additional feasibility study is also performed using pure tungsten as a non-brazed form locked monoblock tile. The effective wall thickness of the tile is reduced here to 3 mm. Design analysis results are presented.

B-19

TESTS AND ANALYSES FOR THE MECHANICAL AND THERMAL QUALIFICATION OF THE NEW RFX FIRST WALL TILES

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The new graphite tiles of the RFX first wall will be thinner than the previous ones, due to the reshaping of the plasma facing surface to achieve a more uniform power deposition. The thickness of many tiles will be further reduced to allow the installation of a large number of electrostatic, magnetic and thermal sensors between the tiles and the vacuum vessel.

During the experimental sessions it has been observed that severe mechanical and thermal loads act on the RFX first wall, due to intense halo currents and enhanced power deposition in the regions where the plasma modes lock to the wall. The average power density over the whole surface of a single tile may reach a peak value of 40 MW/m^2 during pulses with 1 MA plasma current, lasting 100–150 ms. During the same pulses evidences of halo currents flowing from plasma to the vessel through the tiles were observed and the estimated maximum halo current intensity is 2–3 kA per tile.

For these reasons the new first wall tiles are critical components and have to be carefully qualified both from the mechanical and thermal point of view, carrying out several tests and analyses.

The mechanical strength of different tiles (the original and the reshaped ones, with and without the recesses for sensors on the back surface) have been measured, applying various loading conditions that cause different failure modes. A thorough analysis of the experimental results has been executed, comparing the influence of different geometry and loading conditions on the failure probability of the tiles. Further information about the stress distribution in graphite at rupture has been obtained by means of analytical remarks and some detailed finite element analyses.

The thermal transient response of a new tile subject to the most severe plasma wall interaction has been studied and the cumulative effect of repeated wall locked modes has been estimated. The thermal gradients inside the tile volume and the maximum temperature foreseen on the plasma facing surface and on the back surface are the main parameters considered for the thermal qualification of the new tiles.

A thorough discussion of the experimental results integrated with numerical analyses allowed us to qualify the new RFX first wall tiles and to obtain estimates of the safety factors and operating margins during future enhanced experimental sessions.

B-20

MANUFACTURING OF SMALL SCALE W MONOBLOCK MOCKUPS BY HOT RADIAL PRESSING

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The aim of this activity is to develop an alternative technique for the manufacturing of plasma-facing components with monoblock geometry (i.e. the vertical target) of the ITER machine.

The basic idea was to perform a radial diffusion bonding between the cooling tube and the tungsten tile and keeping the process parameters in the range in which the thermo-mechanical properties of the copper alloys are not degraded yet.

The activity started in the frame of ITER EFDA R&D tasks. The feasibility of the joining of Cu//Cu and Cu//W obtained by diffusion bonding was studied and finally some small-scale W monoblock mock-ups were successfully manufactured and tested with respect to thermal fatigue testing (20 MW/m^2 for 1000 cycles). A particular stainless steel container, which is not deformed during the hot isostatic pressing process, was used.

Following these good results obtained using this technique, a new container was then designed in order to perform the Hot Radial Pressing (HRPing) in a standard furnace in which only a section of the canister is heated and only the internal tube is pressurized up to the bonding pressure.

The main advantages of this technique are that both the high temperature/pressure furnace and the machining of the assembling after the joining to separate the container from the mock-up are not anymore required.

The design of the canister and the equipment will be reported in the paper as well as the results obtained by the manufacturing of the W monoblock mock-ups by HRPing.

B-21

EFFECT OF MANUFACTURING RESIDUAL STRESSES ON LIFETIME OF ITER PLASMA FACING COMPONENTS

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In thermonuclear reactors the different parts of divertor are under severe fluxes of energy and particles. The divertor parts which immediately contact with hot plasma are in the hardest conditions. They are Dome, Baffles and Vertical Target (VT). These devices besides cyclic loads with steady amplitude of heat flux can be exposed by extremely high loadings during transient operation. The maximal loadings at transient condition on the VT achieve $20MW/m^2$. Selection of the most reliable and cheap manufacturing technologies is the final stage after structure optimization. Residual stresses after manufacture can strongly change the lifetime of these parts. Lower manufacturing temperature is thought to be favorable, but this is not always. In some cases high residual stresses can compensate the thermal stresses during operation.

This paper was looked at two jointing processes for lower part of Vertical Target with regard to residual stress, and then examines the effect on the low cycle fatigue of the bronze heat sink. For the same joint the following manufacturing processes were analyzed:

- HIP: Low temperature HIP (Hot Isostatic Pressing) with a temperature of 500°C.
- Fast brazing: High temperature brazing with a temperature of 880°C.

3D finite element (FE) model has been developed for the thermo elastic plastic analysis with help ANSYS code. Static, lifetime and initial fatigue crack propagation analysis were performed for two geometric options of VT. These options are tungsten flat tile (FT) and CFC monoblock (MB). The same analysis were performed and compared for both manufacturing processes (with residual stresses) and without residual stresses.

Analysis of residual stress after manufacturing process was shown that all stress in bronze heat sink and copper structural element are in allowable level. More stressed state is after fast brazing manufacturing process. Lifetime analysis was shown that work number of cycles more than design number of cycles (3000). Cumulative fatigue usage fraction in all analyzed nodes was significantly less than 1 (for all options). It is determined, the more residual stresses the less cumulative fatigue usage fractions and more lifetime.

The mechanical properties of bronze heat sink are degrading at temperatures exceeding 550^0C and at long endurance. Therefore, the properties after these manufacturing processes are different. Influence of this effect was taken into account. This part of analysis was shown then better properties the more lifetimes.

Comparison analysis of cumulative fatigue usage fraction for the options were shown that lifetime MB is greater than FT. Lifetime of all geometric option after HIP is greater and properties of bronze are better but it is more costly method. Lifetime after fast brazing is less but more than design number of cycles and the method is cheaper.

B-22

EXPERIMENTAL OPTIMISATION OF A HYPERVAPOTRON® CONCEPT FOR ITER PLASMA FACING COMPONENTS

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The order of heat flux magnitudes (10–20 MW/m^2) to be removed in specific plasma facing components for ITER is in the same range than observed in electron tubes. Historically, fin enhancement cooling techniques with boiling/condensation effect named vapotron® (1950) then hypervapotron® (1973) concepts were developed by Thomson CSF Company in order to cool such devices.

Even if the monoblock concept with a swirl promoter cooling system is now considered as a reference solution for ITER vertical target divertor design, the hypervapotron® concept with a flat tile armour is still an interesting alternative for many reasons:

- Bonding of flat tile is easier than monoblocks, especially in the lower part of the vertical target where compact solutions are necessary;
- Flat tile attachments are expected to be cheaper than monoblocks and would allow to decrease the number of units in the divertor cassette;
- The critical heat flux limit (CHF) is higher for hypervapotrons than for swirl tubes for a given pumping power and for the same width of compared elements.

Nevertheless, from an engineering point of view, a design optimisation is required based on the maximum acceptable element width in order to decrease the number of divertor cassette units.

In case of a hypervapotron® concept, CHF limits are expected to be independent of the heat sink width, contrary to a swirl tube design.

Such an assertion is currently experimentally checked: 9 metallic mock-ups with 3 different width (27; 40 and 50 mm, 500 mm length) are under high heat flux testing in the European facility EB200 (Elecron Beam 200 kW).

In this paper, analyses of tests results of the on-going experimental campaign will be presented.

B-23

QUALITY CONTROL OF PLASMA FACING COMPONENTS FOR TORE SUPRA

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Components in fusion devices are submitted to an increasing performance that has to be ensured by keeping a high control quality level. As far as high heat flux (HHF) plasma facing components are concerned, innovative non-destructive techniques have to be taken into account in the frame of an acceptance inspection procedure.

In order to qualify the efficiency of power exhaust capabilities of HHF components under normal operation, it is necessary to control their thermal and mechanical integrity. This means, that fabrication defects in joints and materials have to be detected, localized and characterized. During the last four years the Tore Supra quality control team improved acceptance inspection procedures for each HHF component before installation in the Tokamak. Innovative investigations of Non-Destructive Techniques (NDT) were performed in the frame of such procedures and appeared to be compulsory tools for quality control of critical parts.

The TORE SUPRA toroidal limiter elements, designed to sustain high heat fluxes in the range of 10 MW/m^2, are made of Carbon Fibre Composite (CFC) tiles bonded to a copper alloy heat sink through a compliant layer of OFHC copper. A strict quality control of all different welded junctions has improved the reliability of copper alloy joints. Several difficulties were encountered and overcame during the fabrication:

- The most important difficulty was the cracking of some tiles leading to a partial debonding between the armour and the heat sink. Systematic Infrared examination testing was operated on each tile of each component to detect and eventually reject debonded tiles. X-ray tomography has shown heterogeneities of copper infiltration after the AMC® process into the CFC tiles.
- With regards to heterogeneous metallic welding, tests such as X-ray examination of stainless steel-copper welded junctions allowed to improve the quality of welding process
- To guarantee good thermomechanical properties, boron carbide coatings, which cover parts of finger elements, were characterised during production samples with many techniques, such as wrenching test, thermal conductivity and thickness measurement.

The paper deals with the experience gained with this fabrication and the quality control of a 600 pieces series with the encountered difficulties and the developed solutions. A detail overview of non-destructive evaluation techniques performed on HHF Tore Supra components is presented.

B-24

PROGRESS ON FATIGUE CHARACTERIZATION OF ITER PRIMARY FIRST WALL MOCK-UPS

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Since 2001, in the frame of the European Technology Workprogramme for ITER, ENEA has launched the thermal fatigue characterization on Primary First Wall mock-ups, armored with Beryllium (Be) tiles, manufactured during the ITER EDA phase. The mock-ups, manufactured by different producers with dimensions of 250(P)x66(T)x90(R) mm, are armored with Be grade S65C tiles with different dimensions and thickness (10 or 20 mm). The heat sink is made from Dispersion Strengthened Copper alloy (DS-Cu Al25) joined onto an AISI316 L Stainless Steel (SS) back plate (BP), both being provided with water cooling channels. The main objective of this activity is to check the thermal fatigue performance of different Be/DS-Cu/SS joints, manufactured by Hipping at different conditions, and tested up to 30,000 cycles. The comparison of the different fabrication procedure will be made on the basis of ultrasonic tests to be carried out at the end of the fatigue tests on the Be/Cu alloy joints.

Preliminary tests on the first four mock-ups have started at ENEA Brasimone CEF 1-2 thermal hydraulic facility at the end of 2001. These mock-ups, hosted inside a special vacuum vessel, are heated by a special electric radiative CFC resistor operating at an heat flux higher than 0.8 MW/m^2. The mock-up reference thermal hydraulic boundary conditions are also regulated by a proper water cooling. The thermal fatigue tests are imposed by controlling the incident heat flux up to the maximum load with a cycle period of 300 s. The fatigue stresses on the Be tile joints are magnified, during the dwell phase, by switching the inlet cooling water temperature from 120°C to 20°C. During the tests, the thermal hydraulic parameters (coolant, heater and material temperatures, water flow rate and pressure) are measured and stored.

The paper presents the thermo-mechanical theoretical analysis, carried out by FEM ANSYS code, and the comparison with the experimental results, obtained during both the steady state and the thermal fatigue tests.

B-25

DEVELOPMENT OF TUNGSTEN ARMORED VERTICAL TARGET FOR ITER DIVERTOR

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Carbon based materials and tungsten are candidate plasma-facing material for the divertor of International Thermonuclear Experimental Reactor (ITER). However the safety concern, resulting from dust and tritiated carbon films formation due to erosion of vertical target (VT), has stimulated to consider design option of VT armored by tungsten only. This paper presents the design of all tungsten VT confirmed by analysis and middle scale mock-up testing.

Vertical target of the proposed design consists of the set of thin high heat flux (HHF) elements fastened to the monolithic stainless steel support structure. Each HHF element of 27…32 mm width is a heat sink beam armoured with tungsten. The most heat loaded (≤ 20 MW/m^2) lower part of the HHF elements incorporates hypervapotron heat sink armoured by ~9×9×10 mm^3 W tiles. Moderate loaded (≤ 3 MW/m^2) upper part of the HHF elements incorporates heat sink of the same width with smooth rectangular cooling channel armoured by ~27×27×10 mm^3 W tiles. Tungsten tiles in both upper and lower part are jointed to CuCrZr heat sink through 2 mm compliant cast copper interlayer to reduce interface stresses due to large mismatch in thermal expansion coefficient of CuCrZr and W. The heat sink beam is bimetallic CuCrZr/SS structure 27…32mm width with direct and reverse cooling channels. Such a design allows organizing coolant manifolds in moderate loaded upper part of VT and using only SS/SS welding at the final assembling. The heat sink beams are joined with support structure by laser or e-beam welding that restricts longitudinal expansion of the HHF elements. Nevertheless mechanical analyses revealed only 1.3 times increasing of the cumulative fatigue usage fraction (0.304 vs. 0.235) in comparison with free longitudinal expansion fastening concept.

The design of the VT was optimised on the base of thermal hydraulic and 3D mechanical analyses. Thermal hydraulic analysis confirms design compliance with reference ITER divertor cooling parameters. Low cycle fatigue lifetime evaluation for heat sink has shown that proposed design meets ITER requirements with considerable (~ 3 times) margin.

To confirm viability of the VT concept, middle scale mock-up was manufactured and tested. The mock-up, reflecting the main features of the design, has two tungsten armoured HHF beam of ~600 mm length laser welded to monolithic SS support structure. HHF tests were performed with reduced areas due to facility limitation: lower straight part was tested at the areas of 27×42 mm^2 and 18×42 mm^2 at heat fluxes of 18.5 MW/m^2 and 20 MW/m^2 during 1000 and 150 cycles respectively, no damages in the joints were detected; upper curved part was tested at the area of 54×42 mm^2 at heat flux of 5 MW/m^2 during 1000 cycles without damages.

B-26

CRACK GROWTH IN FIRST WALL MADE OF REDUCED ACTIVATION FERRITIC STEEL BY TRANSIENT CREEP DUE TO LONG PULSE OPERATION

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Long pulse operation is assumed in ITER. In operation duration, high heat flux is loaded to a surface of a first wall, and there exists large temperature gradient in the wall since other side is cooled. Thermal compressive stress appears in the plasma side and tensile stress appears in the coolant side because of bending constraint. The first wall may have an unexpected surface crack by virtue of plasma disruption and so on. The crack may be propagated by the cyclic thermal stress. More than 300 sec of flattop during plasma burning is expected in ITER and future reactor. To clarify the crack growth mechanism and the effects of transient creep, a three-dimensional finite element model is proposed including transient thermal analysis, elastic-plastic stress analysis and transient creep analysis. And a Blackburn type creep strain-time equation is proposed to represent the transient creep behavior so as to satisfy recent data reported in JAERI-Tech.

In our recent simulation experiments using tensile specimens made of reduced activation ferritic steel F82H, in which electron beam is employed as a heat source and the operation duration is taken as 5 sec, 60 sec and 240 sec, the transient creep gives rise to more crack extension in spite of compressive stress. But the effect of operation duration is diminished gradually when the duration is more than 240 sec. The experimental results are compared with our previous experimental results using type 316 stainless steel specimens.

Numerical analysis indicates that: In long pulse operation, the transient creep relaxes the thermal compressive stress, so the compressive inelastic strain is further enlarged in the yielded region. Therefore, the operation duration becomes longer, the residual tensile stress in the cooling period becomes higher, consequently, stress intensity factor $\boldsymbol{K}$ is also higher. The crack is propagated not by transient creep itself but by thermal fatigue with higher value of stress intensity factor $\boldsymbol{K}$. Using this model, the crack growth can be predicted in quantitatively and crack lengths in the assumed operation durations and repetition cycles are in good agreement with the experimental results. The difference of the operation duration is definitely explained.

B-27

THERMAL STRESS ANALYSIS OF FIRE DIVERTOR MODULE

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The FIRE device is designed for high power density and advanced physics operating modes. Due to the short distance of the divertor from the X–point, the connection lengths are short and the SOL thickness is small. A relatively high peak heat flux of 25 MW/m^2 is expected on the divertor.

The FIRE divertor engineering design is based on design approaches developed for ITER. The geometry of the FIRE divertor consists of water cooled copper fingers and a tungsten brush armor as plasma facing material. A swirl tape insert is used in the cooling channels to increase the critical heat flux (CHF). The assembly consists of modular units for remote handling. A 316 stainless steel back plate is used for support and manifolding. The backing plate is joined to the copper fingers by pins which can slide axially in slots. The coolant channel diameter is 8 mm at a pitch of 14 mm. A peak heat flux of 25 MW/m^2 is expected on the surface of the outer divertor.

The total power flow to the outer divertor is 35 MW. With water at an inlet temperature of 30°C, 2 MPa and a flow velocity of 10 m/s and two channels in series, a margin of about 1.5 is obtained on the CHF.

Thermal hydraulic correlations derived for ITER were used to perform the thermal analysis. A three dimensional thermal stress finite element (FE) analysis of this geometry was performed. Design changes were made to reduce the stresses and temperatures to acceptable levels.

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B-28

TECHNOLOGICAL HEATING FACILITY "PEKLO" FOR MANUFACTURING OF ITER PLASMA FACING COMPONENTS

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Multipurpose vacuum heating facility built at Efremov Institute in the frames of ITER program is presented.

The basis of this facility is one phase transformer, which co-axially envelopes cylindrical (D_{in} = 750mm) vacuum chamber. The secondary winding of transformer propagates into vacuum chamber and may be connected to electrical load. Special heater for indirect radiant heating of components or direct ohmic heating of components itself is used. Nominal power of transformer is 1 MW, which may be increased for a short time up to 2 MW. Secondary winding voltage is varied in the range of 4–24 V and maximum heating current may achieve 250 kA.

The length of vacuum chamber is 3 m, background pressure- 10^{-3} Pa, and so rapid vacuum heating of large components may be performed.

Vacuum casting of copper, fast brazing of armour to heat-sink elements and other possible technological applications for manufacturing of ITER plasma facing components are discussed. The procedure of fast brazing of large-scale mock-up of ITER start-up limiter is also presented in details.

B-29

BEHAVIOUR OF THE PFC TILES ON THE ITER DIVERTOR UNDER CRITICAL LOADING CONDITION

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In the divertor the majority of the heat load on the vertical target (VT) is delivered by the particle flux that follows the field lines in the scrape-off layer (SOL). This flux intercepts the VT at a glancing angle (~1°) causing erosion of the actively cooled carbon fiber composite (CFC) monoblock tiles that form the plasma-facing surface at the strike point. This erosion is a function of surface temperature and particle flux. If, for some reason, there is a lack of thermal contact between the CFC monoblock and the CuCrZr cooling tube, the surface temperature of the CFC monoblock will increase with a subsequent increase in the surface erosion. The higher erosion with respect to the neighbouring monoblock tiles will result in the surface being hidden and thus a reduction in the heat load, surface temperature and surface erosion thereby, providing a stable system.

The manufacturing repeatability of the CFC to CuCrZr joint can never be 100% guaranteed and joint defects will exist. In tiles with defective joints there is a localized increase in heat flux at the coolant boundary. To study the impact of these defects on the performance of the ITER divertor VTs, three cases are considerated: complete failure of one CFC tile joint but with the tile remaining in position and protecting the cooling tube, partial failure of one CFC tile joint leading to localized enhanced heat flux at the coolant interface, and complete removal of one tile from the cooling tube which may produce a cascade effect when the neighbouring tile receives twice the heat load.

The complete loss of a monoblock tile from the heat sink, though considered a remote possibility, is also studied in terms of the effect on the exposed heat sink and support structure under both normal and transient conditions (including disruptions).

The results from this study show that a partial or total detachment of the monoblock tile from the heat sink does not lead to a critical burnout condition. Moreover, it is shown that the monoblock design is very robust and can tolerate fairly large defects. This paper presents the analyses supporting this conclusion and supporting R&D test results.

B-30

ASSEMBLY AND SET-UP OF THE CIEL PROJECT PLASMA FACING COMPONENTS ON TORE SUPRA

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Tore Supra is the only one superconducting Tokamak equipped of actively cooled plasma facing components. The *CIEL* Project (**C**omposants **I**nternes **E**t **L**imiteurs) consists in a complete major upgrade of all the existing internal components. This new device should enable removing 15 MW of convected flux on the Toroidal Pump Limiter (TPL) and should sustain high radiating plasma power losses of at least 10 MW on the first wall during pulse length up to 1000s. The associated primary pressurised water loop (3 MPa at 150°C) has been upgrade too, that is able to exhaust the injected power in a steady state configuration.

At the beginning of 2000, after twelve years of operation (1988–1999), all the existing internal components of the plasma vessel have been definitely removed as the stainless-steel panels of the outer wall, the SS/graphite inner first wall, the six movable modular limiters, and also the magnetic measurement system. This preliminary phase was achieved by a refurbishment of the inner vessel using a specific dry cleaning method with ceramic balls, in order to remove bore and carbon coatings leaved on the vessel wall. The set-up of the CIEL new components on Tore Supra started in April 2000. This second phase was devoted first on site, to the achievement of assembly through specific quality control procedures carried out at each step of the fabrication (dimensional survey of the Toroidal Pump Limiter, specific helium tightness tests dedicated to water cooled components, etc.).

The setting-up operations in the machine were carried out requiring an Optical Metrology System (OMS). The OMS was always referenced to the existing torus magnetic axis. The aim of purchasing a final positioning accuracy of less than 0.5 mm for the whole plasma facing components has been reached. By this way the main CIEL component that consists in a flat actively cooled circular Toroidal Pump Limiter (5m diameter) located at the bottom part of the vessel and supported by six motorized independent jacks, has been positioned very accurately towards the magnetic axis. It was also the same for the set of 12 inner poloidal guard limiters so called “bumpers” dedicated to runaways protection. A set of 120 water cooled stainless steel panels that completely shields the inner vessel against thermal radiation have been installed and connected to the water manifolds by TIG welding. Previously to the water filling up of the machine, the plasma vessel has been placed under vacuum, followed by a specific procedure that consists in a global tightness test of all the water cooled plasma facing components pressurised with helium gas (6 MPa at room temperature).

The paper describes assembly operations and procedures that have been carried out during the fabrication and setting-up of plasma facing components as part of the CIEL Project during the 2000 and 2001 shutdown.

B-31

MANUFACTURING OF PROTOTYPE COMPONENTS FOR THE ITER DIVERTOR BAFFLE

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Among the different Plasma Facing Components which are developed in the frame of the ITER concept, our manufacturing concerned the baffle or limiter type modules which have to respectively sustain heat fluxes from 1–3 MW/m^2 to 10–20 MW/m^2.

A program for the supply of prototypes was realised as a demonstration of the manufacturing features for different designs of Baffle and Limiter Modules. The scope of the manufacture was aimed on the FW heat sink materials such as DS-Copper, Carbon Fibre Reinforced Carbon (CFC) monoblocks, and armour materials, such as CFC and Tungsten vacuum plasma spray.

The supply concerns four Baffle prototypes with a quasi-poloidal shape and having different geometries with two cooling channels each. Each panel is attached onto a Steel block, as a shield block.

Type A panels have a DS-Copper heat sink with two thin 0.5 mm thick Stainless Steel tubes (outer diameter 12 mm, thickness 0.5 mm). The FW is either plasma-sprayed Tungsten (type A/W), or CFC tiles (type A/CFC). Each prototype panel is 830 mm long, 44 mm wide, with two lateral 200mm long curved parts (R = 57.5 mm). During the R&D phase, preliminary Tungsten armoured mock-ups representing one half of the final prototype with its curved part, were manufactured and tested under heat fluxes at the FE200 Facility in Le Creusot. The curved W plasma sprayed-mock-ups sustained 2000 cycles at 2 and 3 MW/m² on a flat side and 1880 cycles at 2 and 4 MW/m² on the castellated side.

The manufacturing routes of the baffles needed two HIPing processes for the Stainless Steel support/first grooved Glidcop plate/S.S tubes/second grooved Glidcop plates bonding. After HIPing, the DS-Copper plate-plate and tube-plate junctions were controlled by ultrasonic inspection. The control was made by the inside of the tube with a rotating probe developed for small diameter tubes. The control of the straight inter-tile bonding was developed with a TOFD sensor. Adapted calibration blocks were developed and used for each configuration of inspection (S.S/DS.Copper or DS-Copper/DS-Copper plate/plate, straight inter-tile bonding, inside of tubes). No defect was observed after realisation of the heat sink part of these prototypes before coating by Tungsten Plasma Spray or CFC tiles.

The panel type B consists in two assembled single 832 mm long, 23 mm wide CFC U-shaped prototypes with a rear slot cutting and a CuCrZr tube, leaving a 1mm gap between them. The monoblocks are supported by means of S.S pads, which have been attached to an active metal casted OFHC layer by an HIP procedure operating at 900°C. The baffle consisted in a lot of CFC monoblocks machined and HIPed around the CuCrZr tube at temperatures close to 500°. The feet are welded onto the S.S shield, as 2 rails along the component. The manufacturing of this panel B was shown to present a lot of difficulties concerning machining as well as HIPing, and which were progressively solved.

The panel type C consists also in two assembled single 832 mm long, 23 mm wide CFC U-shaped prototypes with CuCrZr tubes, put on a S.S. dove-tail mechanical attachment, and leaving a 1mm gap between them.

Both monoblock designs comprise an OFHC copper layer in the bore hole applied by active metal casting (AMC®). In order to join such a way prepared monoblocks onto the CuCrZr cooling tubes, a solid HIP process has been developed operating at a temperature low enough to retain the mechanical and microstructural properties of age hardened CuCrZr alloy after the joining process. The panel C was UT inspected by the inside of the tubes with a rotating probe.

B-32

MANUFACTURE AND JOINING DEVELOPMENT PROGRAM OF THE ITER PRIMARY FIRST WALL:

MANUFACTURING OF BERYLLIUM ARMOURED PROTOTYPES

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In the frame of the Blanket development programme for ITER, the design envisaged a blanket-shield constructed from modules. Primary Wall Modules consisted of a water-cooled austenitic stainless steel Shield Block and a separated First Wall. The Primary First Wall uses a DS-copper alloy as the heat sink material bonded to the shield block and a protection material such as Beryllium, to cope with the plasma/wall interaction. The module was designed to sustain a peak heat flux of 0.5 MW/m^2, a maximum neutron wall load of 1.25 MW/m^2, and an average neutron fluence of 0.3 MWy/m^2 (maximum 3 dpa steel).

A manufacturing program was implemented to demonstrate the feasibility for manufacturing the Primary First Wall panels, including a part of R&D work concerning the joining of Beryllium plates onto a Glidcop heat sink by HIPing or brazing.

The aim of the paper is to present the resulting Beryllium 100 mm × 100 mm mock-ups (manufacturing and tests) and the manufacturing of two 900 mm × 254 mm × 70 mm Beryllium covered prototype panels.

The mock-ups as well as the prototype panels are built on a 48mm thick Stainless Steel support. This S.S support is grooved and HIPed together with a 10mm thick Glidcop plate and S.S 12 mm diameter tubes. They includes welded S.S collectors on each side. The Beryllium armour is either HIPed in TCS (Tecphy), or brazed in Atmostat. The R&D on Beryllium joining concerned the different ways (HIPing and brazing) to bond Beryllium plates onto the Glidcop heat sink.

After HIPing or brazing, the plate-plate and tube-plate junctions were controlled by ultrasonic inspection. Concerning tube/plate bonding, the control was made by the inside of the tube with a rotating probe developped for curved tube configurations. UT inspection demonstrated that the performed joining S.S / tubes / DS-Copper and DS-Copper / Beryllium plate fully satisfied the acceptance criteria. After UT, three 100 mm × 100 mm mock-ups were tested in Jüelich under heat fluxes up to 3MW/m^2.

For the prototype panels, the heat flux tests will be made in ENEA – Brasimone (Italy).

B-33

HIGH HEAT FLUX TESTS OF MOCKUPS WITH W ROD ARMOR*

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Water-cooled heat sinks armored with tungsten rod armor can endure heat fluxes near 25 MW/m^2 without cracking, melting or debonding of the armor. PFCs armored with tungsten rod armor are considered for use on portions of the ITER-FEAT divertor and as the reference design for the FIRE divertor. Sandia National Laboratories has been involved with the development and testing of mockups armored with tungsten rods for most of the last decade and pioneered the initial development of tungsten rod armor as part of the US effort for ITER in the 1990's. In early high heat flux tests on small mockups armored with W rods done in Sandia's 40kW electron beam test stand (EBTS), the mockups exhibited excellent thermal performance; however, the values of the heat flux were reported [1–3] were evidently overestimated due to difficulty in precisely estimating the rather small heated areas in these tests. We have now retested some of these smaller mockups in our large electron beam facility, EB1200, where the available power (1.2 MW) is more than enough to heat the entire surface area of the small mockups. We have also tested three large mockups with tungsten rod armor in EB1200. [4] And recently we remachined two of these mockups to shorten the exposed rod length of 10mm to 7mm and then tested these remachined mockups in the Particle Beam Engineering Facility (PBEF) at JAERI. In both the EB1200 tests and the PBEF tests, the mockups sustained heat fluxes in the range of 22–24MW/m^2. The mockups are 32 mm wide and 400 mm long with dual (water) cooling channels. Plasma Processes, Inc. fabricated the armor by plasma spraying (PS) copper onto the back of close-packed arrays of 3-mm-diameter tungsten rods with an exposed height of 10 mm. In one mock-up, the PS armor bed was HIPped to a copper heat sink using a Ni interlayer. In another mock-up, the joint was made with a full-penetration electron beam weld. We are also pursuing 3-D thermal modeling of the rod-armored PFCs and have concluded that the simple 1-D model initially used in evaluating some results from the EBTS testing of small mockups was not adequate. This paper summarizes test results, briefly discusses issues in diagnostics and interpretation of thermography and calorimetry, and presents results on thermal modeling of tungsten rod armor.

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MANUFACTURING OF ITER LIMITER PROTOTYPICAL MOCK-UPS

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The port-limiter is a high-heat flux component of the fusion reactor ITER. Its main function is to limit the plasma during the start-up and shut-down phases. The limiter has a modular design and is located in equatorial ports. Each port-limiter module consists of a comb type assembly of bimetallic panels, attached together in their rear part, with a gap between them in the front part. This special structure has been designed to reduce the electromagnetic loads during plasma disruptions. The limiter first wall (FW) provided with a Cu-alloy heat sink is coated with beryllium tiles.

An R&D program was developed and executed with the purpose of developing the manufacturing technologies and establishing a suitable manufacturing procedure. The R&D has been mainly carried out manufacturing and testing small and medium scale mock-ups.

The small scale limiter mock-ups consist of two bimetallic panels with built-in cooling channels, welded in the rear side, and completed with manifolds and branch pipes. The FW surface is coated with beryllium. The overall dimensions of the mock-ups without manifolds are 250x250x88 mm. The mock-ups were manufactured by following methods: solid HIP, EB-welding, TIG-welding and machining. Beryllium coating was brazed to the FW. One mock-up was subjected to a cyclic heat test. During testing the flat and radial parts of the FW with beryllium coating were exposed to a high heat flux by an electron beam testing facility.

The medium scale mock-ups consists of bimetal panels without beryllium coating. Overall dimensions of the mock-ups are 600x500x44 mm. The mock-ups were intended for the large scale testing of limiter panels manufactured by HIP and HIP-assisted brazing. After manufacture, the panels were examined by both non-destructive and destructive methods.

Before performing the final one-step HIP or HIP-assisted brazing processes of the SS and the FW Cu-alloy parts, several bondings of three-dimensional parts were tried out. Also SS serpentine pipelines with elbows of the small radius (0.5D) were developed and fabricated.

Flat and curved double-side castellated beryllium macrotiles were developed and produced; the brazing of the beryllium macrotiles to the FW by furnace method with the use of amorphous braze was developed and carried out; the properties of diffusion bonded and brazed joints were investigated. Cyclical heat testing of the mock-ups with beryllium coating under maximal heat flux up to 7MW/m^2 was carried out.

Additional mock-ups manufactured starting with (3 times) thicker plates, joined together with HIP and HIP assisted brazing technology and machined after joining were also tested in order to assess the possibility of reducing the machining and the overall limter costs.

The paper describes the R&D performed and the results so far achieved.

B-35

ULTRASONIC STUDIES ON ITER DIVERTOR AND FIRST WALL PROTOTYPE MODULES

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Blanket and divertor components of ITER are complicated in design and reliable reactor operation sets special requirements on applied manufacturing methods and on quality and integrity assessment methods. R&D programmes have been implemented to develop suitable nondestructive testing method as a part of ITER Primary First Wall and Divertor component manufacturing programmes. The suitability of several nondestructive testing methods have been evaluated i.e. ultrasonic inspection, eddy current, X-ray, infrared thermography and leak testing during different manufacturing stages as a quality assurance method and after thermal fatigue tests as an integrity assessment method.

In this study ultrasonic methods have been applied to evaluate structural integrity of several ITER Primary First Wall and Divertor Vertical Target mock-ups before and after thermal fatigue and high heat flux tests. The techniques used were leaky Rayleigh wave and longitudinal wave reflection techniques. The applicability of ultrasonic methods was evaluated by using small scale mock-ups with artificial manufacturing defects on different interfaces. In addition to outer surfaces of the mock-ups several dissimilar metal planar and tubular interfaces have been examined i.e. Be and Cu surfaces and Be/Cu, Cu/Cu, Cu/Stainless steel, W/Cu and CfC/Cu interfaces. The ultrasonic examinations were carried out before and after cyclic thermal loading for ITER Primary First Wall mock-ups up to 4 MWm^{-2} and for ITER Divertor Vertical Target mock-up up to 15 MWm^{-2}.

The leaky Rayleigh wave technique was used to detect surface defects and interface defects in different joints. The longitudinal wave reflection was used only for interface defects. New probes were designed for inspection tasks. The NDT methods are compared briefly in this paper as well as the accuracy of detecting defects with ultrasonic testing methods.

B-36

EFFECTS OF SUPRA-THERMAL PARTICLE IMPACTS ON TORE SUPRA PLASMA FACING COMPONENTS

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Plasma facing components (PFC's) for Tore Supra have been designed basically for heat exhaust of "normal" (convected and radiated) plasma power. However, in some cases, fast particles have been observed, which locally increase the power flux density, leading to damages of these PFC's.

Runaway electrons in the 10–20 MeV energy range are routinely observed in Tore Supra during disruptions. Usually the plasma drifts towards the high field side due to the loss of its internal pressure and the runaway electrons impinge on the belt limiter where a few graphite tiles, brazed on water-cooled stainless steel tubes- the first generation of PFC's, have been broken. The replacement of some modules of this limiter with improved CFC (carbon fibre composite) tiles in 1997 has provided a more robust solution to cope with runaway electrons, due to the enhanced thermal shock resistance of this material. However, during some disruptions, the plasma equilibrium is maintained and the runaway electrons drift toward the low field side, where they impinge on outboard PFC's. A copper leading edge tube of an actively cooled modular limiter has been damaged in 1993, leading to a water leak.

Fast particles are also accelerated in Tore Supra during lower hybrid or/and ion cyclotron heating in the 100–300 keV energy range. Some of them are trapped between the toroidal field coils due to the magnetic field ripple, and drift downward/upward for the electrons/ions respectively. In 1998, during a high power pulse with ion cyclotron heating, a stainless steel pumping duct, which is located about 30 cm away from the last closed flux surface in an upper torus port, has been perforated by fast ions.

During the experimental campaign of 1999, a water leak occurred on the stainless steel heat sink of an improved CFC brazed belt limiter module, due to supra-thermal electrons created during lower hybrid current drive. The power flux density locally reached more than 10 MW/m^2, well above the 2 MW/m^2 design value of the CFC - on steel technology.

The "CIEL" upgrade of the PFC's, installed during 2000–2001, consists of an actively cooled high heat flux toroidal pump limiter located at the bottom area of the vacuum vessel and six robust CFC guard rings (GR) on the high field side. An actively cooled but bolted GR armour concept allows rapid replacement of individual tiles if necessary. Both of these elements have been conceived to cope not only with the expected "normal" power deposition, but also with the previously mentioned supra-thermal particle fluxes. In addition, a movable low field side limiter, using the same bolted GR armour concept, has been designed and installed in order to protect the RF antennas leading edges against runaway electrons. Back walls and borders of upper vertical torus ports are protected against fast trapped electrons by high heat flux CuCrZr components coated with a B_4C layer (in the new CIEL configuration, the drift direction of trapped particles has been reversed since the toroidal field direction has been reversed.

We will present the damages of in-vessel components related to fast particles and the improved design concepts retained for CIEL components. We will report preliminary results in the new configuration, indicating the robustness of the technological choices to support "normal" and off-normal power fluxes from the plasma.

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FILM AND DUST DEPOSITION IN GROOVES ON ITER-FEAT FIRST WALL AND VERTICAL TARGET

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Beryllium and carbon erosion from the ITER-FEAT first wall (FW) and divertor vertical target (VT) during normal operation causes a deposition of films and dust on the in-vessel regions including the grooves (slots) between armor tiles and the gaps between adjacent plasma facing components. For tritium retention assessment and safety issues it is important to estimate possible amount of beryllium (Be), hydrocarbons (C-H) and carbon (C) films inside these grooves and gaps. Such films can form the dust and flakes with high surface area located partly on hot surfaces.

Based on an analysis of the design and operation conditions for FW and VT the calculation models and numerical codes are developed to simulate the erosion and re-deposition processes within the slots. The model of the FW slot allows to calculate the erosion rate with a dependence of erosion yield on the incident angle of charge-exchange (CX) neutrals.

The results on film growth on the FW slot wall are obtained for erosion rate of 0.1 nm/s on the Be plasma-faced surface. An erosion of Be surpasses its re-deposition at slot neck and on slot bottom. The maximum net deposition rate is ~6 nm/hour at distance of 2 mm from slot neck. A maximum film thickness of 20 μm, slot wall net erosion of 0.45 mm and slot bottom net erosion of 0.06 mm are found for 30000 cycles of the FW operations. Total amount of re-deposited Be inside the FW slots is 60 kg. This is potential source of the Be flakes and fine dust within the vacuum vessel. Only 4–8% of Be films in grooves may fall down onto the Dome which can be hot, the Be dust inventory generated by the FW slots does not exceed 5 kg on hot surfaces that is less than the defined safety limit of 6 kg.

The C-H and C sources for the VT slots are analysed. A source due to erosion of slot walls by ion hydrogen flux into slot seems to be dominant in view of possible film deposition within the slots. Chemical erosion within the slots results in radical formation and re-deposition. Three options of erosion spectra are calculated as boundary and the most credible estimations: only methyl; only heavy radicals; complex spectrum at C-H film etching by atomic hydrogen. Temperature profile down the VT grooves is found for normal operation mode.

Total number of the VT carbon-based mono-blocks is $9.4 \bullet 10^4$, total length of the slots between armour mono-blocks is 4500 m. Calculated rates of the C-H film formation of several nm/hour are similar to the rates measured in the experiments with methyl deposition. Maximum possible film thickness within the slots is calculated of 150 μm. Eroded carbon flux divides into deposition (12%) and escaping (88%) fluxes including a methane fraction. The most credible total amount of the deposited C-H films within slots is of 320 g during the VT lifetime (3000 cycles) including 283 g of carbon and 37 g of tritium. The boundary tritium amounts within the VT slots due to C-H radical co-deposition (25 g for methyl or 177 g for heavy radicals) are considerably less than found for pumping system (2.4 kg).

B-38

WALL CONDITIONING FOR WENDELSTEIN 7-X BY GLOW DISCHARGE

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Glow discharge conditioning (GDC) is an established method for wall conditioning in fusion devices. Although it is not compatible with magnetic fields and cyclotron resonance based methods are presently being investigated, GDC is for the time being the only in practice approved method for the superconducting stellarator WENDELSTEIN 7-X. Therefore, GDC can only be applied during shut-downs of the magnetic field, e.g. overnight or during breaks. Design of the GDC components has to consider the slender toroidal shape of the W7-X plasma vessel with a major radius of 5.5 m, a surface of 220 m² and a large geometrical aspect ratio of 5.5 which differ significantly from those of other fusion devices. In addition large areas of the wall will be covered by boron carbide featuring only a moderate electrical conductivity. Shape and position of the electrodes have to be adapted for effective cleaning. Thermal stability must be guaranteed under exposure to steady state stellarator discharges. 10 anodes will be distributed along the outer toroidal periphery of about 41 m for uniform cleaning and coating of the wall. Feeding of the total glow current through many electrodes allows a compact design of the anodes and to integrate them into the wall protection thus avoiding complex manipulators.

Experiments have been performed in the DEMO cryostat, a full size prototype of a 45°-sector of the W7-X cryostat, to test different electrodes and to study the discharges. The power supply used in the experiment is able to provide current densities of up to 10 µA/cm², typically applied for GDC. Start and control of the glow discharge as well as data acquisition is performed via a PC. Glow discharges in helium showed reproducible breakdown at elevated pressures of approx. $2x10^{-2}$ mbar and macroscopically stable discharges down to approx. $2x10^{-3}$ mbar. Simultaneous measurements of current and voltage during the discharge permit extrapolation to the expected power consumption in W7-X. First tests using a grounded boron carbide coated element do not indicate any problems for the discharge. Measurements of the discharge current with higher time resolution reveal unexpected non-harmonic oscillations at about 20 kHz. Similar observations have been made during glow discharges at W7-AS and ASDEX Upgrade. The oscillations are dependent on pressure and discharge current and are not related to the cycle of the switching power supply. External components can be effectively used to stabilise the discharge. To avoid water cooling graphite is a suitable material for the anodes of W7-X. According to estimates the temperature of specifically designed graphite electrodes would not exceed a temperature of 1200 K when exposed to steady state plasma radiation. The effect of the hot electrodes' thermal radiation on diagnostics is under discussion.

The presentation will give an overview of the basic design of the GDC and summarise the results of the experiments.

B-39

SUMMARY REVIEW OF ASPECTS OF THE ENGINEERING OF ITER DIVERTOR PFC-TO-CASSETTE ATTACHMENTS

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The ITER divertor cassettes comprise a stainless steel body, designed for the full lifetime of ITER, onto which are mounted 3 sacrificial Plasma Facing Components (PFC's). The latter are designed to be replaced a number of times during the lifetime of the machine to allow for component failure, erosion of the armour, or to implement alternative divertor geometries. This approach has been adopted chiefly to minimise the amount of activated waste to be sent to a repository, since, in general, the cassette bodies can be re-used for the next experimental campaign.

An important aspect of engineering replaceable PFC's is the scheme for attaching the PFC's to the cassette body. The main engineering challenges for this are the need for PFC replacement to be carried out remotely once a divertor assembly is removed from the reactor, and for the attachment to be firmly locked into the PFC and cassette in order to provide structural, thermal and electrical continuity between the attachments, PFC and cassette during reactor operation. During the ITER engineering design activities (EDA) there has been work on two schemes; initially on 'shear keys' and latterly on 'multi-links'. The work has concentrated on design for locking, investigating the effect of mechanical and electrical loading on locking, and investigating repeatability of locking on PFC replacement. Design for locking affects RH tools and activities as well as the attachments.

This paper summarises the knowledge gained on the more recent PFC-to-cassette multi-links attachment scheme during the EDA, covering key aspects of attachment design and the main results from tests on, and mathematical modelling of, attachments.

The work has provided useful theoretical and empirical information to be considered in the final stages of the design iterations for the multi-links scheme.

B-40

THERMAL STRUCTURAL DESIGN OF PLASMA FACING COMPONENTS IN FUSION POWER REACTOR A-SSTR2

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Plasma facing components of the advanced steady-state tokamak fusion power reactor A-SSTR2 to be conceptually designed in JAERI are subjected to high heat flux more than ITER. Blanket first wall is subjected to nominal heat flux 1MW/m^2 from plasma. Divertor plate is subjected to heat flux more than 10 MW/m^2 at the maximum heat point. This paper describes thermal structural design and analyses of the plasma facing components such as the blanket first wall and the divertor plate, using finite element analysis codes.

In the A-SSTR2, the blanket system is composed of many blanket modules put on a helium (He) gas duct. Helium gas flows in channels manufactured in a shell of blanket module, flows to the first wall at the top of module. First wall is made of SiC/SiC composite material. Thickness of the first wall is 3 mm. Helium coolant pressure is 10 MPa. Helium coolant temperature in the cooling channel is assumed to 700˚C. Heat conductivity of SiC/SiC material is assumed to be 15 W/m/K. The finite element analysis shows the maximum temperature less than its allowable temperature 1100˚C. But the analysis shows the maximum stress 247 MPa more than its allowable stress 200 MPa. This analysis requests a material developer to make a better SiC/SiC material in heat conductivity and mechanical strength.

Divertor plate made from only SiC/SiC composite material is too weak in strength, because heat flux beyond 10 MW/m^2 loads on the plate locally. Tungsten material is assumed as a structure of the first plate having cooling channels. Thickness of the tungsten divertor plate is 7.0 mm, and jointed to the other structure made of SiC/SiC material. Cooling channel is located at 2.5 mm depth from plate surface. Heat conductivity of the tungsten material is set to 121 W/m/K. Nuclear volumetric heat rate of the tungsten material is set to 20 MW/m^3. Heat transfer coefficient between tungsten material and cooling Helium gas is set to 10000 W/m^2/K. In the case of plasma heat flux 10 MW/m^2, temperature and stress are calculated within allowable limit of tungsten. But, in the case of plasma heat flux 20 MW/m^2, the maximum temperature of the first plate is calculated more than re-crystallizing temperature of the tungsten material.

This paper also describes the mechanical issues and solutions that the future tokamak fusion power reactor would have in the plasma facing components.

B-41

ELECTROMAGNETIC STUDY OF THE ITER DIVERTOR CASSETTE

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This article presents basic results of the electromagnetic studies for the ITER divertor cassette. These studies were performed for two types of the plasma behavior scenario: the fast and slow downward Vertical Displacement Episodes (VDE), since they both are considered to be most critical in respect of the transient electromagnetic loads applied to the divertor components.

For each of the two plasma behavior scenarios, two transient electromagnetic processes were analyzed separately. The first process is caused by the Halo currents and a variation of the toroidal magnetic flux, associated with plasma current. Variations in the plasma full toroidal current, shape and position generate the second transient process. Electromagnetic loads acting on the divertor components at the fast and slow downward VDE have been calculated for both transient processes and presented in terms of: 1) local and total loads due to Halo currents and a variation of the toroidal magnetic flux associated with the plasma current; 2) local and total loads due to variations of the toroidal plasma current, plasma shape and position; 3) a sum of two above.

A single calculational model has been used for all studies. It describes 1/54 part of the axially symmetrical double-walled vacuum vessel and one divertor cassette. The model of the divertor cassette reflects the cassette body, dome with reflector plates, inner and outer vertical targets. The divertor plasma facing components were modelled rather detailed: with separation on the dome assembly, inner and outer vertical targets. The computation model of each plasma facing component describes a steel support structure and so called poloidal elements which reflect the pipes, located at the plasma-facing surface of the support structure. The poloidal elements absorb the major part of the heat flux coming from the plasma. To intensify the thermal conductivity, they include bronze monoblocks with internal cooling channels. To suppress the eddy currents, all poloidal elements are separated in the toroidal direction with gaps. So, the plasma facing components have anisotropic electrical properties with a higher conductivity in the poloidal direction.

Time evolutions of total electromagnetic forces and moments, acting on the different divertor components: cassette body, dome with reflector plates, inner and outer vertical targets have been obtained. The peak values of loads have been defined. Using the space distribution of electromagnetic loads, distribution of equivalent nodal forces has been generated in ANSYS format suitable for subsequent dynamic structural analysis.

The electromagnetic analysis was performed with TYPHOON 3-D shell code, developed at the Efremov Institute.

B-42

MODULAR HE-COOLED DIVERTOR FOR POWER PLANT APPLICATION

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Designing high performance divertors for a power plant needs quite a different approach than known from experimental reactors as ITER. Whereas the design of the ITER divertor is based on water at low temperature as coolant and copper as heat sink material, a power plant divertor has to be operated at much higher temperatures suitable for high efficiency power conversion systems. Cooling divertor plates with water at temperatures below 200°C would waste some 10%–20% of the total power. Moreover, for safety reasons water would not be acceptable in connection with ceramic breeder blankets using large amounts beryllium. Hence a gas-cooled concept is required, allowing for high heat fluxes and coolant temperatures suitable for efficient use in the power conversion system.

Basis for the proposed divertor concept are a modular design as described in [1], and a helium cooled divertor concept with improved heat transfer proposed in [2]. The concept employs small tiles made of tungsten and brazed to a finger-like structure made of Mo-alloy (TZM). Design goal was a heat flux of at least 15 MW/m^2 and a minimum temperature of the structure > 600°C. The divertor is intended for steady state operation but has to survive a number of cycles between operating temperature and room temperature.

Design criteria are maximum structural temperature, tile temperature, primary and secondary (thermal) stresses. Various finite element analyses have been performed by the ABAQUS code to predict the temperature fields and stress fields with a fully 3D model. Maximum tungsten tile temperatures are roughly 2400°C, the maximum structural temperature (TZM) is below 1400°C. Reducing the height of the tungsten tile to 3 mm would reduce the maximum temperatures to 2000°C. Linear elastic analyses show that for a coolant pressure of 10 MPa the primary stresses are below 70 MPa as membrane stress with a local peak stress of less than 90 MPa at a temperature < 900°C.

Elastic-plastic analyses were performed to quantify the thermal stresses and total stresses. There are several assumption that have a considerable influence on the amount of plastic strains, e.g. the choice of a 'stress-free temperature' and inelastic material data. For this study properties of recrystallized materials were chosen as the most severe assumption, with the exception of the yield strength of TZM, where values in between the recrystallized and stress-released condition were used. Cyclic stress analyses were performed, i.e. a series of loading and unloading and temperature changes between operational conditions and RT. There are some regions around the TZM/tungtsen interface, that exhibit plastic straining. For any of these zones shakedown is observed (deformation stays elastic after one cycle of plastic straining) and no cyclic plasticity is found at any point. Hence structural design criteria as required by the ITER structural design code are met. To give some numbers, maximum plastic strains are 0.2% for the stress-free-temperature being at joining temperature.

[1] Pizutto, A., Riccardi, B., Preliminary critical heat flux assessment of the high effency thermal shield devices, 16th SOFE, 1995

[2] S. Malang, S. Hermsmeyer, Gas-Cooled high performance divertor for a power plant, Proceedings of ISFNT-6, San Diego, 2002.

B-43

EROSION PRODUCTS OF ITER MATERIALS UNDER PLASMA DISRUPTION SIMULATION

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Morphology and size distribution of erosion products for five different tungsten grades received under pulsed plasma irradiation simulating plasma disruptions were studied. The erosion products of redeposited W+Be layers and those of W and carbon-fibre composite (CFC) irradiated simultaneously were studied as well. The experiments were performed in the electrodynamic plasma accelerator MKT with the energy density of 300 kJ/m^2. The pulse duration was equal to 60 μs. Pulse number was from 2 to 10. The erosion products were collected on basalt filter and silicon collectors placed in different directions with respect to target and examined by SEM and TEM (for basalt filter) which allow to detect the particles with size < 100 nm. The metal erosion products are spherical droplets. But sometimes flakes were observed. The size distribution of erosion products depends on outlet direction. A number of particles were returned to the target by plasma pressure. At simultaneous irradiation of W and CFC the erosion increases as compared with their separate irradiation. Pieces of destroyed CFC with the size from 5 to 25 μm were observed on collectors and within melted tungsten surface layer. A model of droplet emission and their behavior in shielding plasma is presented.

B-44

BRITTLE DESTRUCTION OF CARBON BASED MATERIALS IN TRANSIENT HEAT LOAD TESTS

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During off-normal events in future fusion reactors significant amounts of thermal energy will be deposited on the plasma facing components. Beside vertical displacement events (VDE), plasma disruptions and edge localized modes (ELM) are the most critical loads. In these events energy densities ranging from about 1 MJm^{-2} to 60 MJm^{-2} may cause irreversible damage to the first wall and the divertor components. Metallic wall components such as beryllium or tungsten will melt; mobilization of the melt layer may result in an enhanced erosion and in the formation of metallic droplets which finally contaminate the plasma. Carbon based materials have been proposed to avoid the above mentioned erosion process via melt layer removal. However, graphites and carbon fiber composites (CFC) will undergo another particle generating process with strong impact on the component erosion, namely brittle destruction.

To investigate this phenomenon experimentally, isotropic fine grain graphites and carbon fiber composites (with and without Si-doping) have been exposed to intense transient thermal loads in the electron beam test facility JUDITH. For different pulse durations (1, 2, 5, 100, 5000 ms) the deposited energy density has been increased stepwise to determine the threshold for the brittle destruction process. Particle emission was diagnosed using time resolved measurements of the absorbed current and by digital photography. In addition different characterization methods such as weight loss, surface profilometry, optical and electron microscopy have been applied to test coupons following the thermal load test.

The generation of dust particles induced by intense electron beam pulses on graphites or CFCs shows a clear dependence on the applied pulse duration and, hence, on the resulting temperature gradient in the material surface. For lower gradients, i.e. for pulse durations in the 100 ms range and beyond mainly small dust particles are generated. Short beam pulses (1 ms range) will cause steep thermal gradients which are associated with the creation of larger units (probably agglomerates of graphite grains or clusters of carbon fibers). A clear correlation of the absorbed current and the onset of the particle emission processes has been established.

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ANALYSIS OF ELECTROMAGNETIC LOADS FOR THE KSTAR PLASMA FACING COMPONENTS*

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The Korea Superconducting Tokamak Advanced Research (KSTAR) necessitates plasma facing components (PFCs) in order not only to protect the vacuum vessel and diagnostics from the plasma particle and heat loads, but also to control the plasma. The KSTAR PFCs are mainly composed of divertor, inboard limiter and passive stabilizer and their supports. The divertor and inboard limiter are divided into 8 and 16 sectors in toroidal direction, respectively. These sectors are physically and electrically connected through only their supports. The passive stabilizer is composed of 16 sectors in toroidal direction, but they are electrically continuous. All PFCs are supported by the double-walled vacuum vessel structure. The aim of this work is to calculate electromagnetic (EM) loads on the PFCs, and to perform stress analyses to help design the PFCs and their supports.

Plasma disruptions are of great concern in tokamak designs. Disruptions can generate eddy and halo currents in the conducting vessel and in-vessel components. The resulting EM loads from these currents play an important role in the design of the vacuum vessel, PFCs, and their supports including the mechanical bridge of the passive plate. The EM loads on PFCs, a major design parameter, would be transferred to the vessel through the supports. There will be a large localized stress on the interface between the vessel and the PFC supports. The SPARK code was used to calculate 3-D magnetic field and EM loads on the PFCs and the vacuum vessel. Depending on the disruption scenarios, two models were defined for the calculation of EM loads. First, for the eddy current induced EM loads calculations, a 90^{o} model composed of the vessel inner wall and PFCs with the supports is chosen due to 180^{o} symmetry of the passive plate. Next, a 45^{o} model corresponding to the halo disruption scenarios was defined for the calculation of the poloidal halo current induced EM loads. Several halo current loops between the vessel and PFCs can be formed through divertor supports or passive plate supports. Since the present supporting structures of PFCs were assumed to be simple shell strips in the model, the resulting high stresses at the supports as well as at the vessel interface were not real stresses and more practical reaction forces applicable to the design of PFC interfaces were required. Therefore, realistic 3-D supports of PFCs were taken into account for better design of the supports.

Since large vertical loads as much as tens of tons are induced especially on the passive plates during vertical plasma disruption, the large localized stresses could be found at the passive plate vertical support as well as at the mechanical bridge. Also, during halo vertical disruption, it was found that a few tons are applied on a divertor support. In order to make these concentrated loads diffuse into the neighbour of the interface and to diminish the localized stresses, a special PFC supports were developed.

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B-47

POTENTIAL PERFORMANCES OF A DIVERTOR CONCEPT BASED ON LIQUID METAL COOLED SiC_f / SiC STRUCTURES

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In the frame of the Power Plant Conceptual Study (PPCS), a Self Cooled Lithium Lead blanket concept has been proposed for an advanced model of a fusion power reactor. This concept will associate low activation properties of SiC_f/SiC with liquid metal potential for high power density, low pressure and high coolant temperature suitable for high efficiency power conversion system. Between possible liquid metals Pb-17Li eutectic has been chosen to minimize energy confined in vessel component due to its low reactivity. In line with this blanket concept, a Pb-17Li cooled divertor, using SiC_f/SiC as structural material could allow recovering of the heat deposited in the divertor, so increasing the plant efficiency.

In this paper, an improved design for such a type of divertor is proposed. This design consists of SiC_f/SiC castellated square tubes, poloidally oriented and using tungsten alloy as armour material. In each channel is inserted a T flow separator which assumes, in the region nearest to the plasma a comb form, thus creating a kind of toroidal channels. The liquid metal flows poloidally (v ~1 ms^{-1}) in one half of the channel, acting like inlet header, then it is forced to pass through the short toroidal channels to cool the high flux region and finally it is routed back to the other side of the poloidal channel, acting like outlet header. This design is able to extract relatively high surface heat flux without exceeding temperature limits and thermal mechanical criteria. Furthermore, quite elevate average outlet Pb-17Li temperatures can be attained (650 °C), with reasonable Pb-17Li velocity.

Assuming an extrapolated thermal conductivity through the thickness of 20 $Wm^{-1}K^{-1}$, which is about a factor five larger than the today SiC_f/SiC value, this design is able to withstand a maximum heat flux of 5 MWm^{-2}. Beyond the thermal conductivity, other identified specific issues are the feasibility of the relatively complicated geometry and the high pressure drops which occur because of the high required lithium-lead velocity despite the low electrical conductivity of SiC/SiC.

B-48

FUSION TECHNOLOGY ENGINEERING R&D AT JET

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Research activities that can provide adequate contributions or solutions to technological issues of importance for both JET and ITER are carried out at Culham and in the European national laboratories. They are closely related to the EFDA Technology Workprogramme. This programme of Fusion Technology at JET (FT) covers a wide range of research areas such as tritium in the tokamak, tritium processes and waste management, plasma facing components, other fusion engineering aspects and safety. This paper presents the status and the programme of those fusion technology activities at JET not directly related to tritium.

The characterisation of first wall material using various surface analytical techniques (SIMS, TOF-ERDA, RBS, NRA, IBA) is carried out on tiles removed from JET. The results are analysed by taking into account the actual integrated ion flux on the plasma exposed surfaces. To investigate the erosion properties of tungsten, a poloidal set of 6 carbon tiles has been covered with tungsten layers and were installed in the JET machine during the 2001 shutdown. A programme using long-term samples (LTS) for erosion rate measurements and sticking monitors to determine sticking coefficients of hydrocarbon radicals is in progress.

A few Mark II Divertor tiles will be tested in the Neutral Beam (NB) test bed facility, simulating different JET plasma heat loads, in order to validate a model for plasma facing components behaviour and the mechanism of first wall material transport.

Optic fibres under qualification for ITER use will be tested on JET as well as an in-vessel viewing system based on laser technology in order to increase the quality and the completeness of the images.

Data of the operating experience of the major JET systems as well data on operating experience with Remote Handling facilities during the JET shutdowns are collected and elaborated to support the ITER design with an improved failure rate data base and operating experience.

Nuclear activation and shutdown dose rate calculation models are benchmarked on JET and some models have been significantly improved.

B-49

THE DESIGN OF A NEW JET DIVERTOR FOR HIGH TRIANGULARITYAND HIGH CURRENT SCENARIOS

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A new Mark-II class divertor has been designed in the frame of an extended enhancement project, which was initiated in order to consolidate the preparation of the ITER operating scenarios. The extension of the present operating domain of ELMy H-modes (the reference mode of operation for ITER) to higher, ITER-like, triangularities at high plasma currents (_=0.5 up to Ip = 4 MA) with plasma pressures of _N~2 and consequentially high heating powers (40–50 MW) determined this new divertor concept. Hence the divertor target geometry was optimised to allow for extreme high triangularity configurations, so that the maximum triangularity at high plasma currents is limited by the stresses in the TF coils and/or disruption forces.

The main features of the design are the following:

1. A design optimised for high bottom triangularities (_~0.56), which features an asymmetric horizontal target, and vertical targets moved to the High-Field-Side as far as possible.
2. A power handling capability (with strike point sweeping) which permits 40MW to be injected into the plasma for 10s (nominal specifications) in configurations with forward and reversed magnetic field (but single helicity); this generally requires the configurations to be compatible with sweeping amplitudes up to 12–14 cm along the target tiles within the current limits of the divertor coils.
3. A compliance allowing to run Advanced Tokamak scenarios
4. A geometry for the outer vertical and horizontal target tiles which maximises the effective pumping within the limits of this open configuration.

Those design requirements were tested against 18 plasma equilibria, which have been chosen to cover a broad range of configurations compatible with future high priority scenarios. The Carbon tiles covering the divertor have been profiled to handle the power with a 1800°C surface temperature limit. The Carriers have been optimised for high electromechanical loads and a casted Inconel 625 solution has been developed. The use of fast a prototyping technique has allowed to visualise the carrier volumes and to adjust the interfaces for the implemented diagnostics.

The design work takes the greatest possible advantage of the experience gained in JET with previous divertor designs and adheres, as much as possible, to technical solutions which have proved effective (for example inertially cooled high conductivity CFC tile or Inconel structures). The proposed design therefore represents a conservative approach to the solution of a technical objective that maximizes the chances for ultimate technical success. Nevertheless ITER relevant technologies such as a casted carriers, have been introduced to optimise the manufacturing.

Due to financial constraints it has recently been decided not to implement this project. Nevertheless the design work conducted from April 2000 to January 2002 shows a number of features which can be relevant in preparation of a new divertor such as the ITER one.

B-50

MAGNUM-PSI, A NEW LINEAR PLASMA GENERATOR FOR PLASMA-SURFACE INTERACTION STUDIES IN ITER RELEVANT CONDITIONS

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The aim: design parameters of the new plasma generator

The FOM-Institute for Plasma physics, in collaboration with the Eindhoven University of Technology and the TEC-partners, is opening a new research line with plasma-surface interaction in conditions typical of the ITER divertor as the central theme. A linear plasma generator is being designed aiming at a hydrogen ion flux greater than 10^{23} $m^{-2}s^{-1}$, energy flux up to 10 MW/m^2, in a magnetic field of 5 T, with a target area placed under a small angle with the magnetic field and of sufficient size to allow the study of redeposition. Chemical erosion, redeposition and hydrogen retention are the key issues to be addressed in this laboratory experiment. The device is to be equipped with comprehensive diagnostics of the plasma and - in situ - of the surface processes. Various tools to manipulate the plasma beam by external heating or cooling (gas injection) are being considered.

In parallel with the experimental programme, computational studies of the plasma and plasma-chemistry are started. The research programme is called ' Magnum-psi' (Magnetized plasma generator and numerical modelling for plasma surface interaction studies).

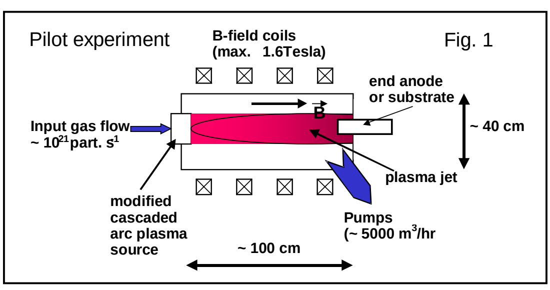
The pilot experiment

To optimize the source and study the properties of the emerging beam, a pilot experiment has been constructed. The arc plasma expands in a vacuum vessel of about 1 m long and surrounded by 5 magnetic field coils. The maximum magnetic field is 1.6 Tesla during about 2 s, the duration of the experiment being limited by the cooling of the coils. Operation at 0.4 T can be sustained for several minutes. We will focus on attaining very high flux densities of hydrogen ions at relatively low temperatures (0.1– 4eV). Figure 1 shows the set up.

We measured the temperature and density profiles with a radially movable double probe, at different position from the plasma source nozzle. Thomson-scattering will be installed for accurate measurements of n_e and T_e in the presence of a magnetic field. With an arc current of 50 to 80 A we measured a plasma with $n_e \sim 10^{21}$ m^{-3} and $T_e \sim 1$ eV in an Ar/H2 mixture.

In a pure argon plasma at B = 0.4 T we attained an ion flux of 10^{22} m^{-2} s^{-1}, in hydrogen the ion flux is an order of magnitude lower. Various strategies will be applied to increase the hydrogen ion flux.

The paper will present the results of the pilot experiment as well as the design of the in-visioned Magnum-psi set up.



B-51

INVESTIGATION OF EDDY CURRENTS IN THE COMPONENTS OF THE DYNAMIC ERGODIC DIVERTOR AT TEXTOR USING ANALYTICAL AND NUMERICAL APPROACHES

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Analytical and numerical approaches for the calculation of eddy currents in mechanical structures of the TEXTOR tokamak in view of operating the Dynamic Ergodic Divertor (DED) coil system fed with AC current up to 15 kA at frequencies up to 10 kHz are described. The design of the in-vessel components located closed to the DED coils requires detailed investigation of eddy current effects to avoid unacceptable heating and forces.

Different approaches depending on skin-layer depths compared with the body dimensions are analysed. The applied algorithms are based on analytical and simplified numerical methods. Precision and application range of these algorithms have been checked by some powerful numerical codes.

To allow for parameter studies the simplified numerical methods together with analytical formulas have been implemented in an easy to use code that applies cylindrical or rectangular prisms in longitudinal or transversal magnetic field. The simplified technique is rather effective for first step engineering estimation and gives a good feeling and understanding for the problem. In a certain parameter range it results in even precise values and can be used for design optimisation of the structures without huge efforts in numerical modelling.

The models and algorithms have been developed for calculation of eddy currents, power losses, temperature rise, mechanical stresses and shielding effect of the conductive structures located in the vicinity of the DED coils. The mechanical structures supporting the DED coils and the divertor target tiles as well as the bellows interconnecting the sections of the vacuum vessel have been studied.

After modification of the component's shape prototypes have been manufactured and successfully tested in a full scale model under the real DED field.

The design recommendations resulting from the eddy current studies contributed significantly to the optimised lay out of the DED in-vessel components.

B-52

THE ITER LIMITER SUPPORT SYSTEM WITH CONTROLLED POSITION OF THE PLASMA FACING MODULES

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Two ITER limiters, located in two diametrically opposite equatorial ports, perform the specific functions of defining the plasma boundary and protecting other regions of the first wall and antennas from direct contact with the plasma during start-up and ramp-down. The considerable thermal load in these phases, and the expected high loads during abnormal events, call for a design using a special movable and separable part of the limiter assembly, comprising high-heat-flux components (the limiter module) that require very accurate alignment to the field contour in order to reduce the heat flux peaking. Additionally the design should allow relatively simple remote handling replacement and refurbishment of the module. Furthermore, uncertainties on the optimum plasma configuration require that the plasma facing surface can be adjusted to match different plasma scenarios.

The supporting and alignment system should be capable of remotely positioning the limiter modules to their nominal position within 1mm relative to each other, and to displace each module radially within a range of ± 20 mm from this position in order to increase plasma operational flexibility (e.g. larger plasmas). It should also permit small rotations around poloidal and toroidal axes in order to precisely align the front surface of the limiter module to the magnetic field surfaces. At the same time, it should support the limiter module against the high electromagnetic loads produced during disruptions, seismic loads and dead weight, as well as providing all the connections. The stringent requirements for accuracy and loading capacity, together with operating and access conditions, necessitate the use of special design solutions. The vacuum environment makes the problem technically even more complex.

An engineering design of the limiter port plug with a remotely controlled mechanical supporting and alignment system has been developed to meet all the requirements, providing safe and reliable operation with good margins. Supports with a changeable geometry are used to give variable functionality: they define the position and orientation of the limiter module and provide support. Position and orientation changes of the limiter module which is connected rigidly to the driving beam are provided by the out-of-vacuum actively driven supports, placed in a sealed container far from the plasma, that act at two points at the rear of the beam. The in-vacuum passive supports are self positioned. The possibility to dismantle the whole system by remote handling (considering also rescue repair schemes) was an important criteria for the selection of the design. The heavily loaded rear part of the limiter plug integrates the alignment system, supports the whole assembly, and provides also the functions of vacuum boundary and neutron shield.

The paper describes the design and layout of the system and its main components, the principles of operating and assembly and presents the main results of analyses.

B-53

INFLUENCE OF TARGET INCLINATION ON METAL RESPONSE IN DISRUPTION SIMULATION EXPERIMENTS

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The behaviour of metal melt layer under external pressure of impacting high heat plasma fluxes determines the metallic PFCs erosion during off-normal events such as plasma disruptions and ELMs in Tokamak and needs in the further studying.

The paper is focused on the results of the experimental study of some metals (Al, Ti, SS, W) response on target inclination (45^o and 70^o) at high heat long-pulse (P_{irr} ~ 100 GW/m^2, t_p = 0.36 ms – pulse duration) plasma irradiation on the VIKA facility [1]. The irradiations were performed in the presence of ~3 T magnetic field.

In contrast to normal irradiation (0^o), where 3 T magnetic field is able to suppress sufficiently the molten metal (Al) flowing along the surface under high plasma pressure (~1 Mpa), the intensive flowing was observed on tilted targets (downstream). Besides the molten metal motion in one perpendicular direction was observed too. Both effects result in the sufficient spreading of the erosion imprint outside of the area of the geometrical projection of the incident beam cross-section on the tilted targets. The degree of these effects is proportional to the target inclination and depends also on the kind of metal. The possible causes of these phenomena are discussed.

The results are discussed also from the aspect of the applicability to the problem of the predictions of the life-time of ITER divertor dump targets.

REFERENCE

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B-54

IN-PILE TESTING OF WATER-COOLED FUSION COMPONENTS AT RELEVANT THERMAL LOADING CONDITIONS

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Experiments on thermal cycling of actively cooled PFC elements simultaneously with neutron irradiation inside of a nuclear reactor core are very important due to its high correspondence to real PFC operational conditions. Both fluence and thermal gradient based damages inside of the PFC armor and heat sink body are evolving in such tests at the same time, in contrast to more prevailing experiments, where fluence accumulation and thermal screening or fatigue tests are consecutive.

The paper presents a technical specification of different ways allowing setting-up of integrated in-pile/thermo-cycling experiments with models of water-cooled plasma faced components. This one gives a review of authors experience in installations of such integrated experiments, collected at the time of the ITER EDA phase, and it accentuates on physical principles and on designing-technological aspects of the experimental rigs design and manufacturing. Two successful test campaigns on in-pile thermo-cycling of small-scale water-cooled mock-ups of divertor elements are described. First of these experiments was based on loading of the mock-up surface with a heat from irradiative – heated tantalum source, transferred by hydrogen dissociation. Method of direct heat transfer through 0.3 mm thick “papyex” soft contacting interlayer was used in the second experiment. A few massive tungsten blocks, warmed up by thermal neutrons and gamma absorption inside of reactor core were used as heat sources in this case. Thermal loading of tested mock-ups during of both experiments was performed in cycling regime. Values of heat flux onto PFC armor, achieved with these tests are: ~ 3.5 MW/m^2 for the method of hydrogen dissociation and up to 8.5 MW/m^2 for the second experiment using direct heat transfer. Some physical and technological reasons preventing to creation of higher heat fluxes are with these two methods are discussed.

Besides the paper contains some results of developmental work purposed for further setting-up of in-pile thermal cycling experiments using electrical (resistive) method of heat flux production, promising more intensive thermal loading of tested mock-ups.

Good performances of described test campaigns allow hoping for further successful development of in-pile thermocycling technology as of important and tolerant testing instrument for fusion purposes application.

B-55

APPLICATION OF A DIFFUSION BONDING METHODOLOGY TO DEVELOP A Be/Cu HIP BOND SUITABLE FOR THE ITER BLANKET

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Due to its low atomic mass and high thermal conductivity beryllium is a candidate material for the plasma facing material of the ITER Primary First Wall (PFW). Given these favourable characteristics one of the most challenging obstacles to the use of beryllium is that a method must be found to intimately bond the tiles to the copper alloy heat sink of the blanket. HIP bonding and brazing have successfully been used in the past to produce good bonds that can withstand the expected heat flux conditions. Both however, have the drawback that they employ temperatures above the ageing heat treatment temperature of an attractive copper alloy for the underlying heat sink, namely Copper Chrome Zirconium (CuCrZr). In addition the existing HIP bonding technique utilises high temperatures that allowed excessive amounts of intermetallics to form resulting in a strong but brittle bond. To address these problems, the mechanisms involved in the production of a diffusion bond have been studied and a low temperature bond developed. As part of the study it was possible to produce a methodology generally applicable to the development of a wide range of diffusion bonds.

This paper outlines the methodology and demonstrates its application to the bonding of beryllium tiles to the heat sensitive CuCrZr alloy. The methodology comprises three main parts; structural analysis to determine the stresses and strains introduced by the HIP cycle, diffusion calculations to arrive at a suitable temperature regime and small scale testing to prove the conditions determined. Structural analysis is achieved through the use of the finite element method. Diffusion calculations solve the appropriate equations that describe the behaviour of the metals present. Small scale testing involves the production of a sample using mechanically applied pressure in an oven to replicate as closely as practical the HIP conditions. The strength of the bonds is assessed using a torque shear test. Ultimately heat flux testing carried out on PFW mock-ups proves the integrity of the bonds, under the conditions expected for ITER.

From the shear testing of the small samples and the heat flux testing of earlier DS copper mock-ups it appears promising that a strong durable bond can be produced using a low temperature HIP compatible with the use of CuCrZr. This process is now being used on a CuCrZr PFW mock-up to prove the compatibility in the context of a full size component [1].

[1] A. T. Peacock et al. Development of Be/Cu and SS/Cu joining techniques to allow the use of CuCrZr for ITER blanket applications, this conference.

B-56

MELT LAYER EROSION UNDER PLASMA HEAT FLUXES TYPICAL FOR ITER HARD DISRUPTION

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During tokamak plasma disruptions, the divertor components in ITER will be exposed to an intense flow of hot plasma. Disruption heat fluxes will cause severe melting of metallic plasma facing components. Melt motion under action of various forces as well as droplet splashing due to melt boiling and due to hydrodynamic instabilities in liquid metal will result in melt loss that could significantly limit a lifetime of the divertor components.

Melt layer erosion has been studied in disruption simulation experiment at the MK-200 plasma gun facility. Aluminum, copper, titanium and tungsten have been tested. The targets were exposed to perpendicular and inclined incidence of hydrogen plasma stream with directed ion energy up to 3 keV, plasma temperature of 1 keV, plasma density of 10^{21} m^{-3}, stream diameter of 6 cm and plasma stream energy density of 15 MJ/m^2. The magnetic field B = 2T was parallel to the plasma stream axis. Experiment was aimed at the investigation and quantification of the erosion mechanisms and their contribution to the net melt layer erosion. Size and velocity of the droplets, which are emitted from the melt layer, have been studied as well.

Under the action of a plasma stream, the melt moves along the target surface in radial direction from the plasma stream axis to the periphery. The radial melt motion results in formation of an erosion crater and of melt mountains around the crater edge. A depth of the erosion crater and a height of the mountains grow linearly with the number of the plasma shots. The net volume of the mountains is practically equal to the net volume of the erosion crater i.e. there is a balance of the material eroded from the crater and the material accumulated in the mountains.

Pulsed energetic plasma causes a volumetric boiling in the melt layer. Intense boiling starts after a certain number of the plasma exposures. Bubble collapsing produces open cavities at the exposed metal surface. Because of melt boiling and formation of surface waves in the liquid metal, the exposed surface shows a rather large roughness (up to 100 microns).

The obtained results indicate that the melt motion is a dominant mechanism of the melt layer erosion. Material evaporation and emission of the droplets contributes only little to the resultant erosion when the exposed target is large enough and it overlaps fully the plasma stream.

B-57

ACCUMULATION OF DEUTERIUM IN SPUTTERED AND RE-DEPOSITED LAYERS OF Be AND W UNDER THEIR SIMULTANEOUS IRRADIATION BY DEUTERONS

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To analyze joint operation of different plasma facing materials of the International Thermonuclear Experimental Reactor (ITER), the targets made of Be and W were bombarded by deuteron fluxes with beam intensity of $\sim 3\times10^{21}$ $m^{-2}s^{-1}$ in a facility MAGRAS modeling the conditions expected in ITER. The goal of the study was the measurement of the chemical composition of re-deposited and sputtered Be and W layers, in particular of deuterium accumulation in them, by Rutherford backscattering of He-ions in a Van-de-Graaf accelerator.

D-ion energies were 200-600 eV. Average dose was $\sim 4\times10^{25}$ m^{-2}. To simulate scattering and return of sputtered Be and W particles, experiments were performed at relatively high D-pressure (5–10 Pa). Compound targets (40 mm in diameter) fabricated from polished Be and W as four central sectors were located on the magnetron cathode. At such conditions, the sputtered atoms predominantly returned back to the target surface. Annular forms of the electrical discharge in the magnetron and of the targets gave an opportunity to study characteristics of both sputtered target zones and re-deposited material layers, resulted from D-ion exposure, in a single experimental run. Ratios of Be-sectors area to W-sectors area in compound targets varied from 3:1 to 1:3.

The experiments have shown that in the layers re-deposited on W-sectors the chemical composition (in particular D-accumulation) is conditioned by the re-depositing Be-atoms. Therefore, the chemical composition of the re-deposited layers on Be and W sectors of a compound target is practically the same. D-content in the re-deposited layers both on Be and W sectors is from 4.5 to 8 at.%. The atomic ratio D/BeO is up to 0.1. It is determined by a high (~40 at.%) O-content in the re-deposited Be-layers. A similar D/BeO ratio was obtained on the homogeneous Be-target. The D-concentration in the sputtering zones of Be-sectors is equal to 2.5-4 at. % (the atomic D/Be ratio is 0.03–0.045). The D-concentration in the sputtering zones of W-sectors is 0.03–0.04 at. %. The depth distribution of D-atom concentration both in the re-deposited layers of all sectors and in the sputtering zones of Be-sectors of the compound and homogeneous targets has the table-like shape and is practically uniform what may be attributable to the high ion dose implantation. The depth distribution of other elements (Be, O, W, C) both in the re-deposited layers and in sputtering zones was also determined. W-content in layers re-deposited on Be-sectors is ≤ 1 at.% and practically does not depend on the ratio of Be and W sector areas. Considerations explaining these findings are presented.

B-58

DEVELOPMENT OF THE EXPERIMENTAL DEVICE FOR ANALYSIS OF THE INTERACTION WITH PLASMA AND THE PEBBLE FLOW

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In the future high-power density fusion reactor, the heat removal and the erosion of divertor plates are important issues. The pebble divertor concept has been proposed for such a future fusion reactor. In order to overcome these difficult issues, this innovative divertor concept has been studied to form the pebble curtain by a large number of falling pebbles to intersect the divertor plasma flow due to removal of high heat load and exhaust gases. The operating window of this divertor system has been estimated from the point of heat load to the pebble and the pumping performance and the development of the pebble dropping system has been performed. The formation of the stable and dense pebble flow in the plasma is very important for removal of heat flux and particle flux. The behaviour of the pebble flow depends on various mechanical properties of pebble-pebble interaction for the pebble flow formation and the pebble-plasma interaction after ejecting to the free space of divertor. In the previous research, we have studied the formation of pebble flow from the pebble-dropping device and we propose the estimation method of analysing the pebble flow rate from the pebble-dropping device. In the next stage, we have to estimate the plasma-pebble interaction experimentally and analytically.

The plasma-pebble interaction is considered as follows. The pebble is charged up instantly in the plasma, however, at the same time the electric sheath is quickly formed surround the pebble. The sheath formation time is negligible short time in comparison with the pebble movement time through the plasma. The electric charge neutrality is almost established at the range including sheath near the pebble. Therefor the pebbles will not be effected by the electrostatic force each other and by the electromagnetic force in the plasma. We consider the influence of plasma flow on the pebble surface including the sheath, and the influence of heat load on the pebble through the sheath. We take notice particularly to the deviation of the pebble movement by the momentum of the plasma flow i n this study. In the first step of the research for the plasma-pebble interaction, we will investigate the behaviour of the single pebble. In the second step, we have the schedule of the research from the single pebble to the pebble curtain with multi-layer pebble flow.

From the viewpoint of measuring the pebble movement deviation by the momentum of plasma flow, the possible deviation is estimated from the calculation. The single pebble (1 mm of diameter) is assumed to be dropped from the point just above the plasma. The width of the plasma flow is assumed to be 1 cm. When the deviation of more than 1 mm is considered as the available measurement range, the desirable ion flux is more than 5×10^{20} /m^{2}sec at the ion energy of 2 keV or more than 4×10^{21} /m^{2}sec at the ion energy of 200 eV. In this presentation, the parameter of the experimental device will be decided for the investigation of the behaviour of the pebble. And the conceptual design of the experimental device is reported for analogical studies of the pebble behaviour in the practical divertor.

B-59

THE THERMOPHYSICAL AND MECHANICAL PROPERTIES OF THE COPPER HEAT SINK MATERIAL INTENDED FOR USE IN ITER

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A key technological challenge during ITER operation will be the satisfactory performance of the high heat flux components in the first wall, divertor and limiter of the machine. Such components will consist of an outer plasma facing armour material (beryllium, carbon or tungsten) bonded to a water cooled copper-alloy heat sink, itself attached to a 316L-IG austenitic stainless steel structure through a hot isostatic pressing (HIPping) process.

Two copper-based alloys are currently under consideration as the heat sink material, namely the dispersion strengthened Cu-25Al alloy (GlidCop®), and the age hardened CuCrZr alloy. The pulsed operation of ITER will subject the high heat flux components and particularly the copper-alloy heat sink materials to cyclic stresses and periods of steady loading, where creep-fatigue life and fracture toughness may be important in limiting operational lifetime. There is, therefore, a need for detailed and accurate mechanical and thermophysical property characterisation data to guide the design efforts.

The specific heat, thermal expansion and thermal diffusivity/thermal conductivity of the two alloys have been measured from room temperature up to 400°C. The densities of the alloys are slightly lower than that of pure fully-dense copper at room temperature and the specific heat and thermal expansion are practically the same as pure copper. The thermal conductivity of the two alloys increases with temperature (over the range 20 to 400°C) and at 400°C the thermal conductivity of the Cu-25Al is similar to that of pure copper. The thermal conductivity of the Cu-25Al alloy is 8% higher than that of the CuCrZr alloy at room temperature and 3% higher at 400°C.

Key mechanical properties for both alloys have been measured. These include: tensile tests (together with Poison's ratio and Young's modulus) from ambient to 350°C; constant amplitude strain controlled fatigue tests at 300°C from 0.3 to 2%; creep deformation to rupture tests at 300°C with duration up to 10000h; and constant amplitude strain controlled fatigue tests with a 300 second creep dwell at 300°C to match the fatigue tests. Best-fit equations are provided with the experimental data for engineering design purposes. Results indicate that CuCrZr is superior to Cu-25Al in all mechanical properties tests performed.

This work was jointly funded by Euratom and the UK Department of Trade and Industry.

B-60

FINAL DESIGN OF W7-X DIVERTOR PLASMA FACING COMPONENTS – TESTS AND THERMOMECHANICAL ANALYSIS OF BAFFLE PROTOTYPES

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Different activities have been carried out in the frame of the construction of the water-cooled Plasma Facing Components (PFCs) of the Stellarator WENDELSTEIN 7-X. These include the target and baffle plates and the part of wall protection covered with graphite.

The target areas, which covered about 30 m^2, are designed to withstand 10 MW/m^2. The selection of the material combination of the target elements is now fixed: 3D carbon reinforced carbon composite (CFC) grade Sepcarb® NB31 is joined to a heat sink made of CuCrZr copper alloy.

Mainly neutral particles and thermal radiation from the plasma load the surface of the 32 m^2 baffle plates of the divertor and 15 m^2 of wall protection with a stationary heat flux up to 500 kW/m^2. Due to the geometry of the W7-X plasma vessel and the restricted distance between plasma and vessel, these PFCs must be flat. The PFC structure consists of fine grain graphite tiles clamped on Cu alloy (CuCrZr) structures with brazed stainless steel cooling tubes. A new fine grain graphite with improved thermomechanical behaviour allows the substitution of the formerly planned CFC.

In addition, 30 m^2 of the central wall protection are loaded up to a maximum of 250 kW/m^2. For this region, cooling elements with clamped stainless steel tubes may be sufficient for the heat removal.

Various prototypes of these latter PFCs were extensively tested in the FIWATKA facility at Karlsruhe. Stationary and cyclic testing investigate the long-term behaviour of the heat transfer between graphite and cooling structure and between heat sink and clamped cooling tube. The experimental results are compared with finite element calculations of the temperature distributions in the elements for both cooling concepts, brazed and clamped tube. The metallographical examinations and the measurements of physical properties of the tested prototypes confirm the design of baffle and first wall protection elements.

Based on these activities, the fabrication of the W7-X divertor PFCs and the graphite covered wall protection for W7-X can be initiated.

B-61

A REPETITIVE TWO-STAGE GAS GUN WITH POROUS CELL CRYOSTAT PELLET GENERATOR FOR CONTINUOUS HIGH SPEED FUELLING

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In order to respond to next generation fusion device needs, PELIN Laboratory in Saint Petersburg, CEA-DRFC in Cadarache and CEA-SBT in Grenoble started in 2000 a tripartite collaboration to promote new pellet injector technologies. The aim was to merge the laboratory knowledge on specific technologies to provide pellet injection systems able to deliver pellets either at low and medium velocities (0.2–1 km/s) and high frequencies (up to 20 Hz) or at high velocities (in the 3 km/s range) and low frequency (0.1 Hz) in steady state mode. Research and development are carried out at Saint Petersburg for the high frequency low speed system and at Grenoble for the high-speed low frequency system. This paper reports the first results obtained in Grenoble on a repetitive two-stage gas gun coupled to a porous cell pellet generator.

PELIN Laboratory designed and built the tested porous cell cryostat. A copper cell is fixed by two flanges across the barrel. A stack of meshes enclosed inside the copper cell acts as a cryogenic valve and regulates the deuterium feed inside the barrel. A helium gas cooling circuit surrounds the copper piece and a heater, mounted on the deuterium entrance circuit, generates a temperature jump. The temperature is increased from 9 K to 23 K to allow the deuterium slush to go through the meshes and decreased to 9 K to solidify it for the shot. The cycle period is linked to the available cooling power allowing a 10 s cycle period with a single stage gas gun with a 300 K propellant gas. Pellets reach speeds of 1500 to 1600 m/s with 90 % reliability in a steady state regime. To allow access to higher speeds a two-stage gas gun (TSGG) technology is needed. Such a TSGG was developed in 1992 by SBT for the single shot mode. Several modifications have now been made to reach the repetitive mode (e.g. safety loop control on piston motion). The TSGG breech peak pressure ranges from 20 MPa to 120 MPa. The cell was able to support these pressures after stiffening of the barrel flanges. Pellet speeds from 1600 to 2600 m/s were performed with a cycle periods from 15 s to 34 s. The pellet reliability was studied as a function of the pellet speed and cell temperature, and measured in the range respectively from 80 to 40%.

This first campaign has produced promising results, firstly on the porous cell pellet generator technology that can provide permanent deuterium fuelling, and secondly on the two-stage gas gun used in the repetitive mode. Improvements in cryostat design should be undertaken to reach lower cell temperature to improve ice resistance and allow higher pellet speeds.

G-1

A NEW PELLET INJECTOR FOR STEADY STATE FUELLING IN TORE SUPRA

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A major upgrade of the RF power and pellet injection systems, the CIMES (*"**C**omposants pour l'**I**njection de **M**atière et d'**E**nergie **S**tationnaire"*) project, is in progress for Tore Supra. This is an in-depth transformation of the RF H&CD systems, together with the pellet injector, and all the necessary auxiliaries in order to produce high performance plasmas with discharge durations up to 1000s. This paper presents the pellet injector part of the project with a detailed description of the injector capabilities and its implementation on the machine.

The new pellet injector is based on the screw extruder technology developed by the PELIN Laboratory. This feed system, well suited for steady state operation, is coupled to a pellet cutting system and a fast valve for pneumatic pellet acceleration. Over 15,000 of 1.6, 2 or 2.3 mm hydrogen or deuterium pellets can be produced continuously at a frequency up to 15 Hz and accelerated at velocities up to 900 m/s with a reliability better than 99% at 10 Hz. The pellet size can be adjusted in real-time during a plasma discharge using data from the feedback control. Novel fast valves, built both by PELIN and by CEA-SBT, have been optimised to reduce the propellant gas load below 15 mbar.L, corresponding to about twice the pellet matter content, per fired pellet. This low gas inventory can be managed with a reasonably sized pumping system.

The implementation on Tore Supra requires the design of a line to pump this propellant gas and a set of guide-tubes to convey the pellets towards the plasma in different configurations of injection. It will be possible to inject either directly from the Low Field Side of the torus at high velocity, or vertically through a top port, or from the High Field Side to compare the respective pellet penetration depths and fuelling efficiencies. Two guide-tubes are available for HFS injection at different poloidal locations (equatorial plane and 38° above). The corresponding maximum pellet velocities (respectively 170 m/s and 200 m/s) have been predicted using two different propagation models and confirmed experimentally in the PELIN Laboratory. To select the configuration, a fast four-way switch has been designed and included in the pumping line.

A new pellet injector able to continuously fuel the plasma with a high level of reliability is in preparation for Tore Supra in the framework of the *CIMES* project. This injector, which is fully ITER relevant apart from tritium aspects, will be installed on Tore Supra in early 2003 and will be available for the 2003 campaign experiments.

G-2

REPETITIVE FUELING PELLET INJECTION IN LARGE HELICAL DEVICE

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Fueling is a primary element of particle control in a fusion research. High temperature plasmas with large dimension in the recent large-scale experiments have posed an issue of inevitability of gas-puffing since a thick and hot scrape-off layer prevents the penetration of neutrals and consequently the fueling efficiency is degraded seriously. Excess neutrals deteriorate confinement performance and enhance sputtering of the plasma facing components. These problems are highlighted in the long pulse operation towards the steady-state. Hydrogen ice pellet injection has a potential to resolve these problems; however, a full-scale experiments for steady-state plasma operation by pellet injection have not be done yet. Large Helical Device (LHD) is a largest stellarator in the world. Intrinsic physical advantage of a heliotron concept and employment of a superconducting magnet system provide capability of steady-state operation of LHD. Long pulse operation of high temperature plasmas of 1.5 keV has been already demonstrated for more than 2 minutes. The prolonged discharge exceeding 10^4 seconds is in plan. LHD provides an excellent platform for the demonstration of the steady-state fueling by pellet injection. The repetitive fueling pellet injector has been developed to fulfil the experimental requirement in LHD. The injector operation is based on the screw extrusion concept, proposed by Mitsubishi Heavy Industry [1] and developed by PELIN Laboratory [2]. Hydrogen is solidified continuously in the cryogenic screw extruder with the maximum rate of 15mg/s. There is no limitation of the operational duration in principle. Refrigeration is done by two GM-type cryo-coolers (3 W at 4.2 K in total). Any cooling media like liquid helium are not used in the system and the cryo-system is completely closed. This concept has realized a high reliability and an easy maintenance. Also the control of temperature is facilitated and the turn-around time of an operation and a standstill are shortened. The pellet injection is available 6 hours after the start of cooling down from the room temperature. The solidified hydrogen rod is extruded with a diameter of 2.0mm or 2.5mm and consequently the extrusion of 55mm/s or 35mm/s is provided, respectively. Then the pellet is cut and ejected by a propellant gas. The velocity of the pellet can be controlled from 150m/s to 500m/s. The amount of gas is suppressed to less than $2Pam^3$ per one pellet launch. The first specification on the repetition rate is 10 Hz for 1000s. A high repetition rate and a long operational duration have been demonstrated close to this specification. This repetitive pellet injection is planed to be employed in the next experimental campaign in LHD which starts in September, 2002. Fueling should be well coupled with pumping for a particle control. Compatibility with the intensive pumping by the local island divertor system is also discussed.

[1] Y. Oda et al. Proc. 18th Symp. on Fusion Technology (1994) North Holland, Vol. 1, p. 661.
[2] I. Viniar et al. Fusion Engineering and Design **58–59** (2001) 295.

G-3

PELLET INJECTOR DEVELOPMENTS FOR STEADY STATE PLASMA FUELLING

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The next step experiments on the existing (LHD, TORE SUPRA) or future (W7-X, ITER) long-pulse machines should be capable of exploring the requirements for steady state operation which can be considered as an ultimate goal of the fusion devices development programme. One of the key tasks of the programme is to develop fuelling systems operating in steady state at a rate matched to the exhaust plasma capabilities. Fuelling is to be provided by a combination of both gas and ice pellet injection. It has been shown that pellet injection can achieve high density plasmas close to the Greenwald density limit with good confinement.

An efficient pellet injector mock-up for steady state fuelling has been developed in the PELIN Laboratory in 1999. A screw extruder for solid hydrogen and deuterium ice fabrication has been developed and successfully applied for continuous pellet injection. In principle, there is no limitation of the injection duration. More than 12,000 2 mm deuterium pellets were produced and accelerated at 10 Hz to 0.5 km/s with a reliability of 99% for over 1200 s. Thanks to its reliable performance during long run operations, the screw extruder has been accepted as a basis for pellet injector developments for ITER, TORE SUPRA and LHD:

- Two injectors have been developed in the framework of the ITER programme. A pneumatic tritium pellet injector TPI-1 has been tested with deuterium in an experimental tritium closed loop in 2001. Pellets of 3.5 mm diameter and 3–4 mm in length were produced continuously during 1500 s at 1–6 Hz. Tritium experiments are foreseen in 2002. A novel centrifuge injector TPI-2 with a screw extruder is being produced now and first experiments are to be made in 2002. An original system allowing to introduce the pellets in the centrifuge launcher just at the rotation axis in order to minimize the synchronism difficulties has been designed.
- A pneumatic deuterium pellet injector prototype has been developed for TORE SUPRA and tested for continuous injection at rates of 10–16 Hz for over 1000 s. A new more efficient pellet injector is being developed now in the framework of the TORE SUPRA *CIMES* project. A dedicated paper will be presented in this symposium [A. Geraud et al.].
- A pneumatic pellet injector for LHD has been already produced and tested for continuous injection at rates 1–11 Hz. For hydrogen pellets of 2 and 2.5 mm fabrication, the injector used two cryo-refrigerators (3 W at 4.2 K in total) instead of liquid helium. The pellet injection is available 6 hours after the start of cooling down. The pellet velocity can be remote controlled from 150m/s to 500m/s. This pellet injector is planned to be used at LHD in September, 2002.

Pellet injectors, using the screw extrusion technology for the fuel feed system, are in intensive study and implementation in the PELIN Laboratory for LHD, TORE SUPRA and ITER. They will be able to continuously fuel the plasmas for this new generation of long-pulse machines, with expected reliabilities close to 100%.

G-4

PELLET MASS TRANSFER IN LONG GUIDING SYSTEMS IN LARGE TOKAMAK DEVICES

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In recent years the breakthrough in pellet refueling performance triggered by magnetic high field side (HFS) pellet injection has initiated more systematic work on geometrical and vacuum characteristics of pellet guiding systems. Work focussed on achieving higher pellet injection speeds to improve pellet penetration and mass deposition to increase the range of refueling scenarios in large tokamaks. Consequently, new tracks have been designed and built in devices such as JET, ASDEX Upgrade and testing has been done at ORNL, where a mock-up for the new JET guiding track was tested. The system at JET is 15 m long and uses a large radius, s-type geometry with a circular tube cross-section. The track at ASDEX Upgrade uses a looping geometry with large radii (2–5 m) and a rectangular cross-section. There for the first time pellets can now be reliably injected at speeds close to 900 m/s from the HFS.

However, mass conservation in such long, curved guiding systems has not been investigated as well and is far from ideal with mass losses exceeding 50% at velocities $v > 500$ m/s. Therefore, work reported here has been concentrating on numerical and in-situ experimental studies of mass loss as a function of pellet velocity, guiding track geometry and cross-section, vacuum pumping configuration and pellet ice composition and quality. The model for pellet mass losses in a guiding track, which was the basis for the initial design of some of the systems, uses the Leidenfrost effect. An evaporating gas layer forms beneath the pellet during long contact (10–50 ms) of the cold pellet with the hot tube mainly due the centrifugal load on the pellets in track bends. At small speeds the gas layer efficiently inhibits the heat wave travelling into the pellet. The effect of the high speed pellet motion would require complicated 3D simulation. Instead a simple approach has been chosen by assuming that the pellet renews the gas layer each time it moves forward by its length. Numerical estimates of the heat exchange and evaporation rates as functions of pellet speed are then performed to evaluate the pumping efficiency of the new JET and ASDEX Upgrade configurations. Finally, the calculations are tested against experimental results from the large tokamak devices and ORNL tests. Measurements are compared for a number of pellet velocities ranging from 150–900 m/s, for different types of geometry and vacuum configurations and for a circular track cross-section compared to a rectangular one.

The first conclusion from experiments at ASDEX Upgrade is a substantial decrease of mass losses in well pumped guiding tracks. Further an increased erosion of pellet surface in a rectangular tube as opposed to a circular one, a non-linear decrease of the thermal insulation of the pellet with increasing pellet speed and a noticeable improvement of all characteristics by doping the pellet ice with 0.5% N_2 have been found. First results of calculations show that the assumption of a renewed gas layer seems to break down at very high speeds. The results will be used to optimise the pumping configurations at the JET and ASDEX Upgrade guiding track systems.

G-5

PRELIMINARY TESTS OF THE PELLET-INJECTOR TPI-1 IN THE OPERATING CONDITIONS OF THE ITER FUEL CLOSED CYCLE

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Realized by way of the injection of the accelerated solid pellets of D, T, and DT, the system of fuel cycle ranks among the main functional systems of the thermonuclear reactors. With this system the fuel mixture density and the mode of burning in the central zone of the plasma column are inspected. The injector of the pellets is to be adequate of the safety exacting requirements, and to have (when applied to ITER project) the following characteristics:

– inherent size of the fuel pellets, mm –	3–7
– pellets velocity near the input into the discharge chamber, km/s –	0.5–1.0
– injection frequency, Hz –	2–50
– operating cycle duration, s –	to 1000
– reliability during the operating cycle, –	> 0.99

The continual action tritium pellet-injector (TPI-1) model corresponds to above-listed requirements at a lower level of the parameters. It has been executed on a basis of the screw extruder and pneumatic accelerator of the fuel pellets; its design features are confirmed by the preliminary tests. However the complex tritium experiments and resource tests of model only can give the ultimate answer on the feasibility and efficiency of the accepted technical resolutions. For this purpose injector is pooled to Research Institute of Experimental Physics (Sarov) experimental closed loop destined for fuel cycle ITER simulation.

The principle scheme, main characteristics, and the results of the first stage extruder tests are presented. The duration of the uninterrupted extrusion of the solid deuterium rode was in the range of 200 s; maximum velocity of the extrusion was close to 20 mm/s. Four working cycles have been carried out; the overall consumption of the liquid He did not exceed 40 l. The pellets formed from the rode were accelerated in the pneumatic accelerator and injected in the diagnostic chamber, where its parameters were measured. After the pellets sublimation gas was arrived from the diagnostic chamber to uranium traps and then to the block of the gas supply. Thus, the concept of the closed fuel's circulation was realized in the test that was taken as the fundamental for ITER.

The programs of the further experiments and tests of the injector are presented.

G-6

COLD TRAP AND CRYOGENIC MOLECULAR SIEVE ADSORBER: COMPONENTS FOR THE TRITIUM EXTRACTION FROM THE PURGE GAS OF THE HCPB-BREEDER BLANKET FOR ITER

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The helium purge gas loop for tritium extraction from the ceramic breeder blanket (Helium Cooled Pebble Bed, HCPB) of ITER includes a liquid nitrogen cooled cold trap for the retention of tritiated water produced in the breeder. Including 0.1% hydrogen in the He purge gas enhances the tritium extraction from the blanket as it allows an isotopic exchange process to take place between the protium in the gas phase and the tritium generated by nuclear reactions and fixed in the breeder. Purging the breeder with hydrogen inevitably produces some tritiated water which can be removed very efficiently by the mean of a cold trap working at -100°C. The final He purification requires a liquid nitrogen cooled molecular sieve adsorber bed to be included in the loop to extract the various isotopic species of hydrogen (H_2, T_2, HT) as well as N_2 and O_2.

The existing cold trap has already been operated in a preliminary test rig under once through conditions to obtain layout data for the final design. The matrix of operation parameters and measured humidities in the helium at the outlet of the trap allow us to define the cooled surface needed for higher flow rates. The results of these humidity measurements confirm the efficiency of the apparatus.

To demonstrate the efficiency of the hydrogen extraction from the helium at the working temperature of 78 K, the test rig will also house a liquid nitrogen cooled adsorber bed, designed and presently under construction at the Forschungszentrum Karlsruhe. To permit a selective regeneration mode between hydrogen and the co-adsorbed impurities (mainly nitrogen and oxygen) the molecular sieve's desorption process should be achieved in two separate stages by raising the temperature first to 150 K and then to 573 K. In the first stage only hydrogen isotopes will be released and in the second the remaining impurities will be released by further heating. The first stage will be controlled by a flow of evaporated liquid nitrogen heated up to 150 K and passed through a tube spiral placed around the molecular sieve container. Complete regeneration is achieved by the mean of electrical heaters installed along the adsorber bed.

As next step, a test facility on a technical scale (one sixth of the flow rate of the ITER-HCPB) will be built in a glovebox in the Tritium Laboratory Karlsruhe (TLK) for tritium experiments with the two main components of the extraction process, cold trap and molecular sieve bed.

G-7

USE OF MICRO GAS CHROMATOGRAPHY IN THE FUEL CYCLE OF FUSION REACTORS

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The exhaust gases of future fusion devices such as ITER can contain Q_2, He, Ar, Ne, N_2 up to 100%; CO and CO_2 up to 50%; CQ_4 up to 20%; higher hydrocarbons (C_2Q_6 to C_4Q_{10}) each up to 5% and Q_2O up to 20%. After processing of these gases with special catalysts and permeators, the composition of the gases changes drastically. The bleed gases of the permeators can contain CO, CO_2, CQ_4, N_2, H_2O, He, Ar and Ne at concentrations up to 100% with very small hydrogen concentrations, whereas the permeate gases will contain almost only hydrogen isotopes with small amounts of the tritium decay product ^{3}He.

Various analytical tools such as laser Raman spectroscopy, mass spectrometry and gas chromatography are nowadays available to detect the gas species mentioned above. A comparison of the three techniques showed that gas chromatography is the simplest, the most used, by far the cheapest, requires minimal space, can be placed fully inside a glove box, does not need high or ultra high vacuum pumps, does not use any type of window material and is easy to operate and maintain. Furthermore, due to recent improvements of micro-gas chromatography* in Japan, the previously long retention times of more than 30 minutes analysis time for the eluting hydrogen molecules can be reduced to less than 3 minutes. As a consequence, three of the five gas chromatographs (GCs) proposed for the ITER Analytical System (ANS) are micro-GCs for the analysis of hydrogen-helium mixtures.

Hydrogen isotope mixtures are analysed with a special micro-GC modified for the connection of an external capillary column which achieves the required separation of the six hydrogen molecules when cooled in liquid nitrogen. The micro-GC is equipped with a back-flush system which allows a pre-separation of helium/hydrogen from all other gas species. This ensures that only hydrogen and helium are injected into the low temperature column, whereas all other gases are flushed to an exhaust of the micro-GC. In this way trapping of impurities in the low temperature column with the resulting rapid deterioration of the separation capability of the gases to be analysed is avoided. Chromatograms obtained with micro-GCs for various impurity compositions are compared with experimental data obtained with conventional gas chromatography. Furthermore, an experimental set-up using only micro-gas chromatography is proposed for the simultaneous measurements of all exhaust gas species (with the possible exception of water) of a fusion reactor.

This paper will present chromatograms measured with micro- and conventional GCs at the Tritium Laboratory Karlsruhe for typical inactive exhaust gas mixtures to be handled in the fuel cycle of fusion reactors and discuss the advantages/disadvantages of this new technique in comparison to conventional gas chromatography.

*Y. Kawamura, S. Konishi and M Nishi: Fusion Eng. Des. 58–59 (2001) 389-394.

G-8

EFFECT OF WATER VAPOR ON TRITIUM RELEASE FROM CERAMIC BREEDER MATERIAL

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In most current designs of D-T fusion reactor blankets employing ceramic breeder materials such as Li_2O, $LiAlO_2$, Li_2ZrO_3, Li_2TiO_3 and Li_4SiO_4, the use of a Helium sweep gas containing 0.1% of hydrogen is contemplated to extract tritium efficiently via isotopic exchange reactions. However, at lower temperatures, the release process of tritium from the breeders is dominated by the desorption of tritiated water and are therefore rather slow since the rate of isotope exchange reactions is considerably low. For this reason, there is still a need to develop techniques that contribute to the acceleration of the recovery of bred tritium.

With this background, the authors investigated the effect of water vapor on the releases of tritium from a lithium silicate ceramic breeder material. Out of pile tritium release experiments were conducted using the ceramic breeder irradiated in a research reactor. Tritium was released from the lithium silicate breeder material using a nitrogen sweep gas with 0.1 % of water vapor. As a result, it was found that the addition of water vapor to the sweep gas greatly enhances the release rate of tritium from the ceramic breeder. To obtain an improved recovery of tritium from a blanket over wide ranges of temperature, the effect of catalytic active metal additives, such as platinum and palladium, on the heterogeneous isotope exchange reactions at the breeder–sweep gas interface was also examined. Pt or Pd was deposited on the lithium silicate ceramic breeder material, which is so-called “Catalytic Breeder Material.” Out of pile tritium release experiments were conducted using the catalytic breeder materials irradiated in the research reactor. Comparison of the results for experiments with the humid sweep gas and that with the catalytic breeder materials reveals that the addition of water vapor to the sweep gas gives the almost same effect on the enhancement of tritium release rate as that the catalyst metals give. These are probably caused by the acceleration of the exchange reaction that takes place on the surface of the breeder material.

The result of tritium release experiment with the wet sweep gas was analyzed using a numerical model. The surface effect was eliminated from the numerical model, since the surface reactions were thought being significantly enhanced by the wet gas purge. The diffusivity of tritium in the lithium silicate breeder material was evaluated using the numerical model. However, the result indicates that the analysis with the assumption of only diffusion control cannot reproduce the experimental result. Thus, the trapping of tritium caused by trapping sites was also considered in the numerical model. Then, it was found that the model with the trapping site effect well reproduce the experimental tritium release curve.

G-9

STUDY ACTIVITY OF THE PT ON SDBC RUSSIAN CATALYST OF ISOTOPE EXCHANGE REACTIONS BETWEEN HYDROGEN GAS AND WATER VAPOR

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The problems of tritium removal from heavy and light water and upgrading tritiated heavy water wastes are issue of the day as before. They can be accomplished by various hydrogen isotope separation techniques but those based on isotopic exchange reaction between hydrogen and water are considered as the most attractive. The large-scale studies of combine electrolysis and catalytic exchange (CECE) process in Petersburg Nuclear Physics Institute, Russia, showed complicated influence of various factors on the process. In general it is due to the presence of two simultaneous isotope exchange sub-processes: phase and catalytic exchange. It was decided to segregate the sub-processes and study them independently in order to understand the separation performance of the available column better. That is necessary to make optimal design of the industrial process at the Detritiation Plant which is being developed in Petersburg Nuclear Physics Institute for tritium and protium extraction from heavy water of PIK reactor.

The paper presents some experimental results of the process study of isotope exchange between hydrogen gas and water vapor over a wide range of deuterium concentration on the wetproofed catalyst developed at D.Mendeleev University of Chemical Technology of Russia (0.8% wt. platinum deposited on styrene-divinylbenzene copolymer). A laboratory scale scheme of glass-made apparatuses was established. The scheme allows investigating phase and catalytic isotope exchange apart since the catalyst and scrubbing layers are physically separated out. The catalyst layer temperature has been varied from 60 to 100°C at atmospheric pressure. Temperature, gas relative humidity and oxygen addition to hydrogen dependences on the catalyst activity have been studied by both deuterium and tritium analysis of input and output flows. The deuterium mass transfer rate of 150 mol/(m^3s) was achieved at 90°C.

The method and technique for catalyst activity study have been developed. The using of this technique allows defining catalytic activity under a wide range parameter changes for a short term. Deuterium and tritium catalytic mass transfer rates as well as reverse reaction rate constants of isotope exchange between hydrogen gas and water vapor, and activation energy constant of 0.8 wt.% Pt on SDBC were obtained. The results have an important meaning in the estimation of an upper limit contribution of the catalyst exchange sub-process into overall CECE process.

G-10

HEAVY WATER DETRITIATION BY COMBINED ELECTROLYSIS CATALYTIC EXCHANGE AT AN EXPERIMENTAL INDUSTRIAL PLANT

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An experimental industrial plant has been in operation in Petersburg Nuclear Physics Institute to develop the combined electrolysis catalytic exchange (CECE) process for hydrogen isotope separation since 1995. The plant is also used for processing tritium heavy water waste; several tons of reactor quality heavy water have been obtained. Last year the plant was updated to provide a means for heavy water detritiation with high detritiation factor. This paper describes operating experience and the experimental results for heavy water detritiation.

The separation height of the chemical exchange column was increased up to 5.9 m, so the overall height of the column was increased up to 7.5 m. Some quantity of wetproofed catalyst developed by the D. Mendeleev University of Chemical Technology of Russia was added in the column. Especial silica gel drier has been used to remove tritiated heavy water vapour from the oxygen produced by the alkaline electrolysis cell and fed to the gas-phase recombiner.

During experimental heavy water purification from tritium detritiation factor about 10^3 was achieved. Heavy water with reduced contents of tritium less than 10^5 Bq/kg was withdrawn at the rate of 4.5 kg per day as a top product. The sample points located along the separation column allowed obtaining tritium concentration profiles within the column. The experimental data are compared with the ones predicted by computer simulation.

The experience gained during heavy water detritiation shows the high efficiency of the CECE process. CECE process is attractive for use in the heavy water hi-flux PIK reactor detritiation plant. This process can find an application at the ITER fusion facility.

G-11

FUEL CYCLE BASED ON AN ELECTRIC PULSED HYDRIDE INJECTOR FOR TOKAMAK T-11M

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The main drawback of the fuel gas puffing systems into the thermonuclear installations is a need of the fuel storage, especially tritium, with the pressure that is some times greater as an atmosphere one. It is stipulated for using piezo valves for obtaining the needed gas flow rate. An Electric Pulsed Hydride Injector (EPHI) may be an alternative to piezo valve. The EPHI allows keeping the hydrogen isotopes in a chemically bound state on the elements containing titanium and generates a gas by heating the elements with an electric current. In EPHI one can choose the gas flow cross-section in such way that the needed gas flow rate will be ensured at the gas pressure approximately 10^4 times as low as that in piezo valve.

Tokamak T-11M has the steel toroidal chamber of a circular cross-section with a major radius R = 0.70 m and a minor radius a = 0.23 m. The plasma volume is 0.7 m^3. The current pulse duration is 0.16 s approximately. The ordinary gas puffing system contains the vessel with deuterium at the pressure of 0.2–0.3 MPa. It is connected with two piezo valves installed in two horizontal ports of the tokamak chamber, respectively, which are placed on the opposite sides of a major diameter. The new gas puffing system contains EPHI with the volume of $1.6 \cdot 10^{-3}$ m^3, the system of saturating the elements by deuterium, sources of electrical power and control system. EPHI is connected to the flange of one of above-mentioned horizontal ports. The plasma characteristics (plasma current, electron density n_e), the gas flow lines intensity H_α and other parameters have been obtained by deuterium puffing using piezo valve and EPHI. Analysis of experimental results shows that the use of EPHI allows obtaining the smooth changes in time of the lines intensities H_α, a fast (0.027 s) growth of n_e up to $2.3 \cdot 10^{19}$ m^{-3} and a long (0.06–0.13 s) maintenance of the maximum value of $n_e \approx 3.5 \cdot 10^{19}$ m^{-3}. In using piezo valve the change of n_e in time has a form of a triangle with the maximum of $4 \cdot 10^{19}$ m^{-3} at the time of 0.067 s.

The conclusions about advantages of EPHI in comparison with piezo valves in gas puffing systems of the thermonuclear installations, published earlier including SOFT, have been confirmed on practice in this work. The use of EPHI carries less of perturbations in plasma and promotes to an improvement of its characteristics.

G-12

THE INNER DEUTERIUM / TRITIUM FUEL CYCLE OF ITER

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The use of tritium in future nuclear fusion reactors calls for a careful design of all tritium exposed parts of the reactor, particularly for the entire fuel cycle to minimise the inventory and to accomplish safe handling of tritium. Not only in view of the control of effluents and releases, but also for economic incentives as much tritium as possible needs to be recovered for reuse from all off-gases and waste streams within the Tritium Plant of ITER.

The major subsystems of the inner tritium fuel cycle of the ITER Tritium Plant are the Storage and Delivery System (SDS) including the Long Term Storage (LTS) vault, the Tokamak Exhaust Processing (TEP) system and the Isotope Separation System (ISS) as well as the Analytical System (ANS). The design basis operation scenario is a short burn pulse of 450 s. A long burn pulse scenario of 3000 s was assessed to determine the design implementations. In both cases the duty factor is specified to be 25%, which leads to repetition times of 1800 s and 12,000 s respectively. The main cryo-pumps of the tokamak will not be regenerated during the short burn pulse. Consequently the complete amount of fuel, about 120 g of tritium for a DT shot at the specified flow rate of 200 Pam^3s^{-1}, must be supplied by the SDS and will only be recovered during the following dwell period. On the other hand, the long burn pulse requires a semi-continuous recovery in accordance with the regeneration scheme of the cryo-pumps and recycle of the unburned fuel adjusted to the required flow rate and composition after purification and isotopic separation, including the removal of protium.

Besides the supply of the deuterium / tritium fuel and other gases to the tokamak and the recovery of tritium from all off-gases of the machine the inner fuel cycle within the Tritium Plant has quite a number of additional duties. The ISS needs to be continuously operated to process the tritiated hydrogen gas stream from the Water Detritiation System. The SDS/LTS handles both incoming and outgoing tritium shipments to and from ITER and is equipped for tritium accountancy and ^3He recovery. The TEP system comprises a process ring manifold and a vacuum and a high vacuum manifold to recover tritium from miscellaneous tritiated gas streams. The operation of all of the subsystems of the inner tritium fuel cycle is strongly interlinked, and the required availability is much higher than the actual duty factor of the tokamak.

The current design of the inner fuel cycle of ITER will be described as a whole. Above and beyond the discussion of the fuel cycle operation during burn and dwell and the tritium inventories in transit and within the different subsystems the philosophy for tritium recovery from various side streams will be explained in detail to illustrate fuel economy and waste minimisation. General safety measures including the recovery of tritium during off-normal operation will be described.

G-13

EU CONTRIBUTION TO ITER CTA FUEL CYCLE DESIGN AND R&D

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Design and supporting R&D are being carried out in the EU in preparation for the construction of ITER. As the Tritium Plant and portions of the Vacuum Pumping systems are located in one of the "time-critical" buildings the existing equipment design and layout are being extended by the detailed design of piping, ducting, cabling and services to ensure compatibility with the allocated building space, taking into account fire prevention, access/egress routes, redundancy and diversity requirements. Necessary design changes in the pumping system to accommodate the reduced number of divertor level ducts and the potential for integration of cold traps for hydrocarbon radicals are being assessed. Studies to standardise, to the maximum extent possible, pump designs and components for different applications – torus exhaust, neutral beam and cryostat pumping – are in progress.

Vacuum pumping R&D is focussed on the continuation of parametric tests with a full range of (non-tritiated) gas mixtures, including elevated temperature runs to determine performance in leak detection mode. The results of cycling and poisoning tests, and the determination of regeneration conditions for all gas species, which are underway, will be included in the paper. Development of a mechanical backing pump with the required high leak tightness, achieved by the use of a magnetic drive coupling between the motor and pump, and minimisation of cross-contamination between pumped gas and the oil in the timing gears, achieved by incorporation of ferrofluidic seals between the process and oil filled cavities of the pump body, are proceeding well. Tests on a specially adapted roots blower are expected to substantiate initial results on a prototype ferrofluidic seal.

Performance tests with the integrated CAPER loop for torus exhaust gas processing (TEP) are being made to confirm that the highly demanding overall detritiation factor (DF) of 10^8 can be achieved, thus substantiating results already obtained for the three process steps tested separately. The design and fabrication of an ITER scale fast delivery, self-assaying tritium storage bed, using ZrCo intermetallic as gettering material is nearing completion and available test results are presented. The parametric testing of analytical devices based on micro gas chromatography – the reference technique for gas analysis in the ITER Tritium Plant – demonstrates the required capability to provide rapid analytical determinations with very small gas samples, and full characterisation of performance for all anticipated gas compositions is being carried out.

Tests of full scale ITER components, including a PERMCAT (final isotopic exchange step in TEP) and a cryosorption panel are being prepared for on-line testing at JET with gas from the torus exhaust during plasma operation.

G-14

COMPARATIVE INVESTIGATION OF PALLADIUM ALLOYS STRUCTURE AND PROPERTIES EVOLUTION WITHIN TRITIUM MEDIUM AT 400°C

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Investigation of binary (Pd20Ag) and V1 (Pd15Ag0.6Pt0.6Ru1.0Au0.2Al) alloys, as a material for palladium diffuser in the ITER Fuel Clean-Up System was performed. Tritium exposure duration was as long as 2292 h. Calculated He-3 concentrations in the alloys have reached respectively 600 and 500 appm. It was shown that that after the initial increase of the alloy lattice parameter the further decrease of it took place. Positron spectroscopy data have showed that simple defect concentration in the alloys stood stable but Helium bubble size and its concentration rose significantly during Tritium exposure. By microstructure investigation it was shown that Helium bubbles nucleation was uniform through the grain volume but bubble growth was observed predominantly along dislocations and grain boundaries. Tensile property investigations at high temperature (400°C) have revealed significant elasticity decrease of the alloys under study.

G-15

TRITIUM TESTS WITH A TECHNICAL PERMCAT FOR FINAL CLEAN-UP OF ITER EXHAUST GASES

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The minimisation of tritium containing effluents and releases in accordance to the ALARA principle (As Low As Reasonably Achievable) is one of the key issues for future nuclear fusion reactors. For both safety and fuel economy purposes as much tritium as possible need to be recovered for reuse within the tritium fuel cycle of ITER.

In the course of a DT plasma burn tritium is fuelled into the ITER tokamak at a rate of almost 1 kgh^{-1}. Since the design target for ITER is to loose not more than 10^{-5} gh^{-1} into the Vent Detritiation System of the Tritium Plant, the plasma exhaust gas need to be processed such that a tritium removal efficiency of about 10^{8} is achieved. Expressed in terms of tritium concentrations this corresponds to a decontamination from about 130 gm^{-3} down to about 10^{-4} gm^{-3} (0.1 Cim^{-3}) or 50 ppb (parts per billion).

In view of the fact that the element hydrogen is ubiquitous and protium undergoes exchange with tritium, it is in principal necessary to remove any tritium containing species from the off-gases to very low levels. However, it is rather difficult to reduce for example the water vapour concentration in a gaseous mixture to levels of a few ten ppb. Thus a very low tritium concentration can almost not be achieved without isotopic dilution employing deuterium or protium. The final step of the three stage CAPER process developed for the recovery of tritium from all off-gases within the Tritium Plant of ITER is therefore based on isotopic swamping with protium in a so-called PERMCAT. The PERMCAT (permeator catalyst) reactor is a direct combination of a palladium / silver permeation membrane and a catalyst bed and was developed for final clean-up of gases containing up to about 1% of tritium in different chemical forms such as water, methane or molecular hydrogen. The PERMCAT unit is based on isotopic swamping in a counter current mode to optimise the tritium removal process and to keep the load to the Isotope Separation System of ITER reasonably low.

In previous tritium experiments employing laboratory scale PERMCAT reactors decontamination factors as high as 10^{5} have been measured at tritium inlet concentrations of 1%. These experimental observations are well in agreement with results from mathematical modelling. To characterise PERMCAT reactors at throughputs relevant for ITER two technical scale units have been manufactured at the main workshop of the Forschungszentrum Karlsruhe. One unit has been shipped to JET for tests in their Active Gas Handling System, the other is being intensively tested at the Tritium Laboratory Karlsruhe (TLK).

The results from first tritium experiments with a technical scale PERMCAT reactor at the TLK will be reported. The design principals and future potentials will be discussed and the decontamination factors measured at different flow rates and gas compositions will be presented.

G-16

SELF RADIOLYSIS OF TRITIATED WATER

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Increasing attention has been directed to reduce tritiated emissions from tritium facilities. The common practice for gas clean-up system is based on catalytic oxidation of tritium into water. Detritiation facilities produce various qualities of tritiated water. Tritiated water is usually stored in polyethylene vessel or stainless steel made container, depending on its activity. For radiological reasons, the most active water is adsorbed on molecular sieve. For all kind of containers, the inner pressure increases due to ^3He production by radioactive decay of tritium and chemical species production due to effect of beta radiation on water molecules. If the production of ^3He is well known, the processes taking place in self radiolysis of tritiated water are not well established especially for ionic polluted water. The purpose of this study is to understand the mechanisms leading to gas production in tritiated water and to establish a predictive model of the rise up of pressure in containers.

During the radiolysis of chemically pure water, molecular products generated are molecular hydrogen and hydrogen peroxide. The production of oxygen is due to subsequent decomposition of hydrogen peroxide.

The synthesis of chemically pure tritiated water (1800 Ci/L) has been realized by isotopic exchange between gaseous tritium and liquid water (18 MΩ). We analysed both the composition of the gaseous phase and the proportion of H_2O_2 in the water for several weeks. Thus, we have determined the radiolitic yield of molecular hydrogen and hydrogen peroxide, (GH_2 and GH_2O_2). In order to determine the chemical reactions that occur in the system, we simulated the production of radiolitic species in chemically pure water with the CHEMSIMUL code. This code, developed at Risø National Laboratory (Denmark), is a computer program used for numerical simulation of chemical reaction. Then, we fitted the simulation to the experimental observations. These experimental results and comparison with chemical simulations are presented in this paper.

We plan to use the same experimental procedure for the study of impure tritiated water. First, we will experimentally find out the radiolitic yield of molecular products with water containing ionic species. Regarding to the experimental results, we will fill out the previous chemical equations system and propose a predictive model to improve the safety of stored tritiated water.

G-17

GLOVEBOXES ATMOSPHERE DETRITIATION PROCESS USING GAS SEPARATION MEMBRANES

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The tritium facilities consist of a multiple containment for a safety handling. The gloveboxes constitute the second barrier of containment. The conventional tritium removal system, performing the elimination of tritiated species present in low concentration in the gloveboxes atmosphere, is based on the catalytic oxidation of tritium into tritiated water followed by an adsorption on molecular sieve. This method has demonstrated its high reliability but the main disadvantage lies on the size of the equipment necessary for its application to gloveboxes atmosphere detritiation. The use of gas separation membranes in tritium removal systems has been studied to reduce the size of facilities or to process a larger contaminated gas volume. According to previous studies, the separation process using hollow fibre polyimide membranes appears to be advantageously applicable to detritiation systems. Our investigation is focussed on the selection of the most effective membrane for this application.

Three experimental apparatus have been designed in order to investigate the hydrogen, deuterium and tritium recovery performances of various membrane modules synthesised from various polymers and characterised beforehand chemically and physically. Experiments are performed with small quantities of hydrogen, deuterium or tritium in the range representative of the radioactive contamination level in tritium handling gloveboxes. The hydrogen or deuterium removal experiments are performed with an experimental device which enables to reproduce transient or steady state conditions. The results obtained with this test loop allow a better understanding of the permeation of hydrogen and deuterium in low contents (some ppm) and to choose a membrane module for tritium separation experiments. Tritium removal experiments are operated on a second device set up in a glovebox. In order to evaluate the suitability of the membrane for detritiation process, the separation performances obtained with hydrogen (ppm) and tritium (ppb) have been compared. The effect of tritium on the separation is studied observing the evolution of performances of a membrane exposed to contaminated atmosphere during several months. To interpret the experimental data, a numerical simulation code of the separation of multicomponent gas across asymmetric hollow fibre membranes with various flow patterns has been specifically developed. The pure component permeability rates injected in the simulation tool are acquired by a third device.

This study allows to select the most suited membrane for this application investigating the effect of the type of polymer, the pressure, the flow rate, the concentration on the separation performances. The conclusions of this work will be used to design and to optimize an industrial membrane detritiation process.

G-18

FRENCH EXPERIENCE IN TRITIATED WATER MANAGEMENT

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Tritiated water is predictably produced in tritium plants. Most common atmosphere clean up systems, based on catalytic oxidation, produce weakly active water (few Ci/L). ITER detritiation plants might generate important volumes of such water containing very low tritium inventory. On the other hand, direct reaction between tritium and traces of oxygen present in processes can form highly active tritiated water (above 200 000 Ci/L, $7.40 \cdot 10^{15}$ Bq/L). Thus, a significant amount of tritium could be mobilized into oxidized form, which has a major radiological impact. That is why, both from a safety and from an economic point of view, tritiated water management appears to be a key point in the exploitation scheme of tritium plants. This paper aims to present the French know-how concerning this topic.

Detritiation processes, consisting in catalytic oxidation of tritium into water and subsequent trapping on molecular sieves, generate important amount of liquid containing very low tritium inventory. ITER General Site Safety Report considers, in normal operation conditions, the production of weakly active water with tritium content below 10 Ci/L. This water is planed to be treated by the Water Detritiation System, which is a Combined Electrolysis and Catalytic Exchange process (CECE). However, our experience proved that specific operations, as maintenance or dismantling of tritium processes for example, could lead to the production of more active water, up to 500 Ci/L. For the treatment of tritiated water, we propose the Bipolar Electrolysis process that is based on isotopic separation by electrochemical permeation through a series of Pd-Ag alloy membranes. This process, which is suitable for activity up to hundreds Ci/L, allows tritiated water enrichment, together with an important volume reduction, and a negligible atmospheric reject. Subsequent enriched water could then be treated as highly active water.

Most processes used for gaseous tritium treatment (chemical purification or isotopic separation) are water sensitive. That is why highly active water potentially formed in processes should be trapped immediately. Molecular sieves, providing safe storage conditions are commonly used. Tritium recovery from this water is run in two steps. Tritiated water is first removed from molecular sieves using a hot inert gas sweeping and cryogenic trapping. Water is then vaporized and is reduced in tritiated hydrogen. Several metals can be used for this operation; their advantages will be discussed. This reduction produces high purity hydrogen containing tritium percentage above 6% that is compatible with standard isotopic separation processes.

G-19

A METHOD OF MEASURING THE DENSITY OF NEUTRAL HYDROGEN ATOMS USING A PHOTOEMISSIVE DIODE

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As it is known, hydrogen isotopes have found a wide use in the setups of controlled fusion. In practice, it is necessary to control neutral hydrogen pressure (concentration) and this requires suitable measuring techniques. Thus, for instance, it seems desirable to know about the hydrogen pressure in the chamber of the hydride injector, when the latter is used for gas-filling hydrogen isotopes in tokamak.

The device for pressure measurements must have small dimensions and be capable of measuring the pressure in a wide range of magnitudes. The suggested method enables the device to be designed meeting these requirements. In addition, the method is distinguished with its simplicity and allows a good time resolution to be achieved. The device based on this method can consist of a photocathode emitting small (of the order of ten microamperes) electron currents and a receiving anode. In the simplest case the potential difference on the electrodes and the current value are chosen so that to avoid electric breakdown and the effect of a spatial charge. In these conditions the value of current flowing through the device is inversely proportional to the concentration of neutral hydrogen atoms. Analytical and numerical investigations of the current value as a function of neutral hydrogen concentration, electrode spacing and current density of photoemission from the cathode have been conducted. It is shown that high efficiency of measurements can be achieved at a quite low illumination of the photocathode (of the order of tens of milliwatts when usual cathodes used). Pressure intervals are determined depending on the cathode-anode spacing where the measurements of neutral hydrogen concentration with the device proposed prove to be the most effective.

G-20

MAINTENANCE OF THE JET ACTIVE GAS HANDLING SYSTEM

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The JET Active Gas Handling System (AGHS) has been in operation in conjunction with the JET machine since spring 1997. The tritium levels within the vessel have remained sufficiently high, 6.2g at the end of the DTE1 experiment and 1.5g currently, that the AGHS has been required to operate continuously to detritiate gases liberated during D-D operations and to maintain discharges to the environment to ALARP. Maintaining the system to ensure continued operation has been a key factor in guaranteeing the continued availability of the essential sub-systems.

Extensive work was carried out to remove and replace failed roughing pumps in the Mechanical Forevacuum (MF) System. The pumps were contained in a sealed containment room known as the MF Casemate. This room had some historic tritium contamination that was subject to a clean up campaign prior to the pump removal. The complicated task, of removing the pumps, involved a thorough assessment of tritium contamination levels in the process pipe work followed by decontamination and measurement of actual levels. A significant decrease in the tritium concentration in the pipe work was achieved after several iterations using different decontamination techniques. The failed pumps were breached using isolators and pressurised suits to provide suitable protection for the operators. One wall of the MF Casemate had to be dismantled in order to remove the pumps. The pumps, the largest of which weighs 5 tonnes, were transported on skids and then craned to a holding position in the AGHS main process area pending further work to investigate the precise mode of failure and possible repair. Replacement pumps were installed and leak tested and the MF Casemate re-sealed.

The inlet filter of the Exhaust Detritiation System showed signs of becoming blocked during the 2001 shutdown. It is thought most likely that this was due to oil clogging the filter from an incorrectly connected leak detector. The filter was bypassed and the off gassing measured to determine it's potential tritium inventory. Following this assessment radiological protection measures were specified which included local isolators, a tent and pressurised suits. The filter was successfully removed and replaced with no spread of contamination and minimal dose to the operators whilst maintaining the availability of the EDS to provide detritiating ventilation to the Torus and the neutral injection boxes.

This paper will detail the experience gained from carrying out these, and other, maintenance activities in the JET AGHS whilst ensuring that the plant remained available for operations.

This work was performed under the European Fusion Development Agreement.

G-21

TRITIUM RELATED STUDIES AT THE JET FACILITIES

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JET is a unique facility in the preparation of ITER in that it is the largest operating fusion device and provides the closest plasma parameters to ITER. Furthermore, it offers the largest variety of heating possibilities, it has Be handling capability, and was built from the beginning to perform DT experiments and will be the only magnetic fusion device for the next decade able to study tritium related issues. Various research activities of importance for JET and ITER are performed within the EFDA JET Fusion Technology Task Force (FT-TF) involving the JET Operator (UKAEA) and several EU Laboratories. Some of the tritium related issues studied are reported here:

Tritium in flakes and tiles: During the 1997 Deuterium-Tritium Experiment (DTE1) tritium accountancy performed within the Active Gas Handling System (AGHS) revealed clearly that a large fraction (up to 40%) of the tritium injected into the tokamak was trapped in the machine. At the end of the tritium clean-up campaign the tritium release rates were very small, but there were still 6 g tritium (17%) remaining in the machine. Inspection of the machine revealed the presence of flaking co-deposited carbon-hydrogen layers mainly at the inner water-cooled louvres. These flakes have a very high specific tritium activity and represent the main trap for tritium, as the tritium content measured in tiles was small in comparison. During the 2001 shutdown flakes from the sub-divertor region of the tokamak were collected remotely to help reconcile the tritium balance at JET.

Detritiation of tritium contaminated materials: Due to the injection of 36 g tritium into the tokamak during DTE1 and the multiple reprocessing of 20 g tritium available in AGHS, components which were in contact with tritium are now contaminated. Special detritiation techniques are developed for components replaced during operations to allow waste disposal under economically sound and environmentally safe conditions.

Development of water detritiation*:* Two batches of tritiated water collected at JET have been sent to Canada for reprocessing. As this route might not be available in the future, design studies of a water detritiation plant at JET are very important. Laboratory size plants with liquid phase catalytic exchange (LPCE) column have been built to test the performance of various packing materials and catalysts and to determine mass transfer for D and T between water and gas.

Use of AGHS as a test bed for ITER: Special ITER components such as a PERMCAT reactor which is one of the main components of the ITER Tritium Exhaust Processing System, and cryo-panels of the same design as those to be used in the ITER cryo-pumps will be tested in the AGHS at JET under ITER-like conditions.

G-22

AN OVERVIEW OF PROCESS INSTRUMENTATION, PROTECTIVE SAFETY INTERLOCKS AND ALARM SYSTEM AT THE JET FACILITIES ACTIVE GAS HANDLING PLANT

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The JET Facilities Active Gas Handling System (AGHS) consists of a number of interlinked processing sub-systems for the supply and recovery of tritium gas used in the JET machine.

In order to carry out a multiplicity of monitoring and control tasks, a diverse range of process instrumentation is employed to support tritium plant operations. Overall there are approximately 500 process measurements within the AGHS. Instrumentation falls into three broad categories including general process measurements, ionisation chambers and special on-line analytical measurements. Where possible commercially available instrumentation is used that meets both the functional and quality requirements needed to ensure full compatibility with a tritium handling facility. Standard commercially available instrumentation have been specifically adapted for mounting in a secondary containment environment that are subject to variable vacuum and temperature conditions. Custom designed instrumentation including ionisation chambers and oxygen monitor give measurements in tritium applications where the commercially available options have not been suitable. In addition a dedicated hardwired interlock and alarm system provides an essential safety function and forms part of the safety case for operation with tritium. In the event of failure modes, each hardwired interlock will back-up software interlocks and shutdown areas of plant to a failsafe condition. Hardwired alarms indicate the status of equipment and are divided into three groups with different levels of severity. Overall there are 159 hardwired interlocks used in the AGHS.

This overview presents the different types and design of process instrument used throughout the AGHS with specially adapted or custom-built instruments highlighted. Design of the interlock and alarm system is outlined and general methodology described. Practical experience gained during plant operations concerning the maintenance of instrumentation and protection systems is summarised. The methods employed for routine functional testing of essential instrument systems needed for continuous safe operation of the plant is explained. Prospects for further enhancements to AGHS instrumentation are discussed.

G-23

TRITIATED GAS WASTE ARISEN FROM FUSION REACTOR OPERATIONS AND POSSIBILITY OF ITS ISOTOPE SEPARATION IN TRITIUM SYSTEMS OF THE REACTORS

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The intensive development of thermonuclear power greatly depends upon the solution to problems associated with the safe operation of the reactors. An important part of the problem is radioactive waste because fusion reactors based on the use of radioactive isotope tritium will produce solid, liquid and gaseous waste. Although, a total amount of the waste and its potential hazard will be much less then fission power planets because of fissionable materials absence and operations will be more simpler and reliable, nevertheless, the fusion waste problem exists and must be solved taking into account some decontamination modes of operations and a decommissioning stage.

By using ITER parameters main modes of fusion reactor operations are discussed including maintenance, replacing of some plasma facing components and decommissioning. Some estimations of a gas waste based on initial proposals for decontamination processes were carried out.

The waste must be treated in tritium system where chemical impurities are removed and protium-deiterium-tritium mixture goes to an isotope separation system (ISS). The ISS is designed to treat D-T plasma exhaust gas and blanket gas from tritium breeding zones. Additional amount gas to ISS from decontamination processes may increase drastically protium and deuterium concentration in separated gas and ISS will be more complex.

Hydrogen isotope separation of the gas mixture using a cascade of cryogenic distillation columns and recycle operations to decrease amount of columns and numerous computer codes were developed.

Possibility of process protium-deiterium-tritium mixtures with a few tens percents protium and ration D/T $\gg 1$ has been discussed.

G-24

EXPERIMENTAL STUDY ABOUT HYDROGEN ISOTOPES STORAGE ON TITANIUM BED

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As knew, the Nuclear Power Plant Cernavoda equipped with a Canadian reactor, CANDU type, is the most powerful tritium source from Europe. On long term, due to a 6*10 exp16 Bq/year, Cernavoda area will be contaminated as increasing the tritium quantity. Also, the continuous contamination of heavy water from the reactor, induce a diminish of moderation's capacity. Therefore, it consider that is improper the use of heavy water after 40 Ci/kg in the moderator and 2 Ci/kg in the cooling fluid. For these reasons, we have developed a detritiation technology, based on catalytic isotopic exchange and cryogenic distillation. Tritium will be removed from the tritiated heavy water, so it appears the necessity of storage of tritium in a special vessel that can provide a high level of protection and safety of environment and personal.

There was tested several metals as storage beds for hydrogen isotopes. One of the reference materials used for storage of hydrogen isotopes is Uranium, a material with a great storage capacity, but unfortunately it's a radioactive metal and also can react with the impurities from storage gas. Other metals and alloys as ZrCo, Ti, FeTi are also adequate as storage beds at normal temperature. The paper presents studies about the reaction between hydrogen and titanium used as storage bed for the hydrogen isotopes resulted after the detritiation of tritiated heavy water. The experiments that were made use as process gas protium and deuterium at different storage parameters.

The storage devices for hydrogen isotopes that use as storage beds titanium, can be useful in safety conditions for storage and also for hydrogen isotopes transport particulary for tritium. The isotopes of hydrogen that it can be storage on titanium bed, are stable and it may be removed at very high purity.

G-25

PERFORMANCE TESTS OF THE ITER MODEL PUMP

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The point design of the ITER high vacuum system is based on cryosorption pumps located inside the vacuum vessel in the divertor ducts. The cryopump design evolved from an earlier concept and some redirection has been needed, especially to come to a more compact design in terms of valve integration and housing. However, the model pump, which is currently being extensively tested in the TIMO test facility at Forschungszentrum Karlsruhe, is still believed to be a good representation of the ITER pump.

The most demanding scenario for operation of the pump system is the long pulse quasi-steady state DT operation at a fuelling rate of 200 (Pa·m^3)/s, necessitating several cryopumps in staggered operation to provide for quasi continuous pumping. With the cryopumps being accumulation pumps each will have to be regenerated after a certain amount of time, which is dictated by explosion avoidance safety aspects.

Within the first fundamental test campaign with limited cooling power but operation conditions typical for the ITER pumping phase, the principal interrelation of valve position, (limited) gas throughput and resulting pumping speed for different gases and gas mixtures could be determined. In the meantime, the test facility has been upgraded so as to replicate ITER relevant total cycle times. This paper presents the operation data of the model pump under ITER cycle time conditions for the complete plasma exhaust operating envelope, i.e. regeneration including fast heating, desorption, pump out and fast re-cooling, and pumping including full ITER relevant gas flow. The influence of coolant supply rate on the steady-state temperature distribution of the cryopanel system and on rapid cool-down is assessed. It is demonstrated, to what extent the latter issue is affected by the temperature level chosen for partial regeneration, which itself has to be adapted to the composition of the gas mixture to be pumped. The possibility to operate the cryopanels at elevated temperatures and the consequences of such a revised concept are also discussed.

G-26

A LARGE SCALE CRYOPANEL TEST ARRANGEMENT FOR TRITIUM PUMPING

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The design of the ITER high vacuum system is based on a number of supercritical helium-cooled cryosorption pumps providing a high pumping speed and capacity, and fast on-line regeneration. To pump helium, which cannot be condensed at the available 5 K cooling conditions, and to help pump hydrogens, the cryopanels are coated on both sides with activated charcoal granules. A comprehensive programme is currently under way to test a near ITER scale model pump with about 50% of the pumping surface proposed for the full scale pump, in the TIMO test bed (Test Facility for ITER Model Pump) at Forschungszentrum Karlsruhe. The experimental campaign is designed to reveal design weaknesses and covers all aspects of operation. The only cryopump aspect which cannot be considered in TIMO is the performance under tritium exposure. In this field of cryosorption, only one small scale experiment limited to liquid nitrogen temperatures has been performed until now.

Therefore, a cryopanel test arrangement will be installed in one of the cryovacuum modules of the AGHS (Active Gas Handling System) at JET. The panel surface area corresponds to one ITER panel, 28 of which are installed in each pump. The active surface is by a factor 300 larger than for the former small scale experiment.

This task will be performed within the JET Task Force Fusion Technology. The main objectives of this programme are to clarify the pumping mechanism of tritium when compared to the other hydrogen isotopes, to study the regeneration behaviour of cryosorption panels loaded with tritiated gases, and to estimate the different tritium inventories. This is a mandatory input to come to an optimized regeneration scheme, which is expected to involve warming up to different temperature levels to release different gas species at different times. For this concept, the results at JET will be combined with the corresponding ones for non active gases from TIMO, in order to fully account for any isotopic effect.

This paper presents the actual status of the task and outlines how it is planned to be performed at JET. The design of the test arrangement is fully described. The paper demonstrates how the AGHS is used as a benchmark test bed for an ITER component.

G-27

HYDROGEN FROSTING SCENARIOS WITH THE ASDEX UPGRADE IN-VESSEL CRYO PUMP

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The ASDEX Upgrade vacuum vessel is equipped with 14 turbo molecular pumps (TMPs) and a cryo pump (CP). The available deuterium (D_2) pumping speeds are S_{TP_D2} = 10 m^3/s for the TMPs and S_{CP_D2o} = 100 m^3/s for the cryo pump (CP). For the CP the ideal pumping speed referred to the chevron opening was quoted. It is obtained with good accuracy as long as the vessel pressure (p_v) is two orders of magnitude larger than the saturation vapour pressure (p_s) of the pumped species referred to cryo panel conditions. At zero pumping speed the vacuum vessel end pressure p_e is reached that can be obtained with a cryo pump. Its level lays roughly one order of magnitude above the saturation vapour pressure. The cryo panel is cooled by 2-phase helium (He) passing through a sub-cooler. Under standard conditions the sub-cooler works at ambient pressure (p_a). Allowing also for the frictional pressure drop required for transferring helium through the CP, a panel saturation temperature of 4.3 K results. Then the ideal pumping speed is reached with good approximation for a vessel pressure of $p_v > 10^{-8}$ mbar for D_2 and $p_v > 10^{-4}$ mbar for hydrogen (H_2). Ideal H_2 cryo pumping is thus only achieved during a plasma discharge with its typical neutral pressures above 10^{-3} mbar. In between discharges, however, residual H_2 inventory on the cryo panel would limit the vessel pressure to nearly the CP end pressure $p_{eH2} \approx 10^{-5}$ mbar, due to the small pumping speed ratio $S_{TP}/S_{CP} \approx 1/10$.

A lower hydrogen end pressure of the cryo pump would be desirable for vessel conditioning during pulse intervals and during standby. Also for species investigations a major contribution of the CP would be desirable for speeding up the transition from hydrogen to deuterium. To achieve these goals the cryo panel temperature has to be reduced as far as possible. The helium supply circuit of the cryo pump was therefore equipped with a suction fan, designed to pump down the sub-cooler at least to a boiling pressure of $p_b \approx 0.5$ bar. With the resulting cryo panel temperature of ≈3.5 K the end pressure becomes then $p_{eH2} \approx 10^{-7}$ mbar. The penalty to be paid consists in larger helium losses. Initially, evaporation is required for dropping the He boiling temperature of the sub-cooler inventory. In addition sizeable throttling losses are linked with the liquid transfer from the storage vessel at p_a to the sub-cooler at p_b due to the increase in pressure difference. These losses, together with technical constraints of the suction fan, thus limit the minimum boiling pressure that can be reached.

The dependence of the H_2 saturation vapour pressure (p_{sH2}) on temperature is well established down to temperatures as low as 2.5 K. Also the relation between the He boiling pressure and temperature is precisely given. Thus a relation between the boiling pressure in the sub-cooler and the resulting vessel pressure can be derived. The prediction of the vessel pressure is rather precise, since the supply with 2-phase helium permits to predict the cryo panel temperature within narrow limits (≈1/10 K). This relation provides therefore a good tool for calibrating the pressure gauges of the vacuum vessel over a wide pressure range. Therefore boiling pressures above ambient will also be assessed in order to reach a vessel end pressure above or near 10^{-4} mbar. Then also baratrons could be calibrated.

G-28

EVALUATION OF SUPER CRITICAL HELIUM AS A COOLANT FOR DIII–D TYPE CRYOPUMPS*

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DIII-D uses three cryocondensation pumps for plasma density control. The DIII–D pump consists of a series of concentric stainless steel tubes assembled together. The pumping surface consists of 25 mm stainless steel tube with two phase helium flowing inside. In order to increase the flow velocity, the heat transfer coefficient and the flow stability of the two phase flow, an annular inset of 19 mm is used to restrict the flow channel. The three pumps inside DIII–D have performed as expected for the last several years. The pumping surface of each of the three cryopumps is about 1 m^2 in area and is maintained below 5 K by cooling with two phase helium (1.3 atm, 4.35 K). The two phase helium (TPH) was chosen for DIII–D because it is available at the DIII–D site and is used for neutral beam and other applications. The pumping speed is about 30000 l/s per pump. The helium flow rate is about 5 g/s for each pump.

Superconducting machines under construction such as KSTAR and SST-1, have super-critical (sc) helium available on site and it is desired to use it for cooling the cryopumps. The typical conditions of the helium are 0.4 Mpa (3.94 atm) pressure and 4.2 K.

The design of the DIII–D cryopump is simple, robust and reliable. In order to evaluate this cryopump, two phase helium and supercritical helium are evaluated for the GA design of the cryopump.

It is concluded that with supercritical helium a flow rate of 50 to 60 g/s (compared to 5 g/s with two phase helium) will be required to achieve a similar performance. The co-axial insert used in the DIII–D helium panel will not be required with sc helium.

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G-29

SIMULTANEOUS TRITIUM AND DEUTERIUM TRANSFER IN A WATER DETRITIATION CECE FACILITY AT TLK

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A Combined Electrolysis Catalytic Exchange (CECE) facility is running at the Tritium Laboratory Karlsruhe (TLK) in order to investigate the simultaneous transfer of tritium and deuterium between various molecular hydrogen isotopes in gas and water, employing a Liquid Phase Catalytic Exchange (LPCE) column. Experiments are performed to determine the height equivalent of theoretical plate (HETP) and the mass transfer coefficients during simultaneous deuterium and tritium exchange between deuteriated and tritiated water and gaseous hydrogen isotopes. The experiments are aimed to provide the data required for the design of large isotopic exchange columns needed for the recovery of tritium from waste water generated during the operation of the tritium facilities of fusion machines.

The exchange column is filled with a mixture of catalyst (platinum on charcoal and PTFE) and ordered packing layered in the column. Pre-heated water at the working temperature is fed into the column from the top and tritium-deuterium-hydrogen mixtures saturated with water vapour from the bottom. Hydrogen depleted of deuterium and tritium is removed from the top of the column. The column has several extraction points for both liquid and gas phases for analysing their compositions. Deuterium and tritium distribution along the column will provide information about possible differences in the HETP along the column and the differences in the mass transfer coefficients.

The analytics foreseen for aqueous flows in this facility are a FT- infrared spectrometer for determining the deuterium content and LSC for determining the tritium content. The IR spectrometer is connected in-line with the CECE facility, samples being extracted with the aid of a micro-pump. In the gas phase the deuterium content will be measured with an Omegatron mass spectrometer. Quantitative measurements on known gas mixtures have revealed that reactive collisions between ions and molecules, that leads to the formations of trimers H_3^+, D_3^+ and ions such as HeH^+, N_2H^+ occurs at higher rates than predicted by theory. For calibration, several gas mixtures were prepared, comparative measurements being done with a Quadrupole mass spectrometer. For quasi-continuous measurements of concentrations along the LPCE column the Omegatron was linked with the CECE facility.

For safety reasons related to the tritium inventory of tritiated water, a solid polymer type electrolyser is used in the experimental facility. The experiments allowed to determine the enrichment factor between gas and liquid for both deuterium and tritium. Preliminary tests used a Vapour Phase Catalytic Exchange (VPCE) system instead of an electrolyser to prepare the required composition of hydrogen-deuterium-tritium.

G-30

ITER PROGRAM-VITON EXPANSION JOINT

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The Cryostat main chamber of the ITER reactor is connected to a Vacuum Vessel through a duct. In order to compensate the mounting displacements (during the final assembly operations of Cryostat and Vacuum Vessel) and movements (between Cryostat and Vacuum Vessel) during the service and maintenance operations the installation of a rectangular expansion joint (2.2 m of width and 3.4 m of height) is foreseen. For this reason, in the frame of the ITER program ENEA and EFDA requested Idrosapiens to design, develop and manufacture an expansion joint prototype that fulfills the intended use. The utilization of a metallic expansion joint was not considered due to its excessive space requirement. It was therefore decided to design and manufacture an expansion joint from a rubber material.

The selected rubber material was Viton® A401-C which is a fluoroelastomer dipolymer developed by DuPont Dow Industries. This choice was made because this dipolymer, obtained from two monomers vinylidene fluoride (VF2) and hexafluoropropylene (HFP), has the ability to better withstand high temperatures in comparison with other rubber materials. Moreover, Viton® A401-C has good radiation resistance and out gassing properties. Viton® A401-C was utilized to produce the flexible element of the joint, which comprised a single U shaped bellows manufactured by a forming die and press process. The function of the flexible element is to assure the capability of the joint to compensate the required relative displacements between the ITER structures assuring, at the same time, perfect tightness between the chambers. The flexible element is then linked to the duct through a rigid frame with flanges and bolts in order to achieve the required tightness. An internal structural shield provides additional support to the bellows in order to resist pressure stresses on the rubber sheet when is subjected to a delta pressure.

This project has been executed through the following steps:

1. Stress analysis of the Viton® flexible element using thin shell pressure formulae;
2. Stress analysis of the frame which supports the flexible element by F.E.M.;
3. Definition of the mixing, moulding and curing process of the Viton® in order to obtain the U shaped flexible element starting from Viton® powder and moulding additives;
4. Manufacture of 3 square expansion joints at reduced- scale (1.3 m x 1.3 m);
5. Helium leakage tests to determine the ability of the joint between the rubber sheet and the rigid frame to assure the required leak rate (10^{-8} [Pa m^3/s]).
6. Pressure tests in order to verify the ability of the elastomeric bellows to withstand a delta pressure of 150% of the required maximum in-service pressure (3 [bar]).

The executed tests provided evidence that the bellows could achieve a leakage rate between $8.2 \cdot 10^{-6}$ [Pa m^3/s] and $8.5 \cdot 10^{-9}$ [Pa m^3/s] and pressures up to 3 [bar].

H-1

CARRIER AND BORE TOOLS FOR 4" BENT PIPES

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Removing and installing Vacuum Vessel components remotely during maintenance of a fusion reactor often requires cutting, welding and inspection of cooling pipes. To allow the replacement of these components while minimising the space requirements, bore tools are preferred to orbital tools for accessibility reasons.

Requirements following ITER’s developments were defined in 1998. A general standard for the cooling pipes inside the cryostat was assumed to be stainless steel 4" bent pipes (100 mm ID) with a bend radius greater than 400 mm and a 6mm thickness. Cutting / welding is required up to 10 meters away from the tool insertion point. Following these requirements, the work performed under the EFDA work program aimed to demonstrate the feasibility to operate with bore tools by means of an innovative carrier concept and to study the associated mechanisms.

The carrier’s design is based on a modular configuration during the navigation through the pipes which is changed to a rigid beam when the working area is reached. The carrier is composed of standard elements for the general functions and dedicated tooling for performing the machining operations.

Each carrier configuration includes the following capabilities :

- Set up the tool from pipe entry point to the working zone (10 meters),
- Position the tool at the required location,
- Generate stress in pipe :
  - clamping inside the pipe,
  - compensate internal pipe stress after cutting (100 daN load),
  - align two pipe sections before welding (20 mm axial stroke , 10 mm radial stroke under a 100 daN load)
- Provide necessary rescue functions.

A complete remote maintenance sequence is made of the following operations: pipe cutting, tack welding, butt-welding, non-destructive testing to check the weld bead quality.

This paper presents the whole carrier concept and the results of the test campaigns on the prototypes since the beginning of the fusion task.

H-2

TECHNOLOGY AND CONTROL FOR HYDRAULIC MANIPULATOR

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Hydraulic manipulators are candidate for fusion reactor maintenance. Their main advantages are their large payload with respect to volume and mass, their reliability and their robustness. However, due to their force control limitations, they are disqualified for precise manipulation. For the same reason, they are dangerous for the environment and themselves in case of undesired contact or unexpected collision with the environment. CEA, in collaboration with CYBERNETIX and IFREMER has developed the advanced hydraulic robot MAESTRO. Force and hybrid control has been developed in order to avoid these problems.

The MAESTRO arm actuating technology is based on an intensive use of “flow” servo-valves. It means that the response of the servo valve to an electrical current step is a precise flow of oil in the joint. For force monitoring of the arm, pressure sensors are used in the control loop to set an equivalence between the pressure in the joint’s chambers and the flow provided by the valve. Using «pressure» servo-valve instead of «flow» servo-valve makes a real simplification of the control loop. In that case the response to a current step is a pressure step and no more extra sensors are needed for monitoring the hydraulic joint in force control mode. Using this kind of valves makes also big safety improvements, for the environment and for the arm itself.

The feasibility of such a control methodology was proven with existing industrial pressure servo valves on a mock-up. IN-LHC, designed and manufactured a prototype of valve that fits the performances and space constraints of the Maestro arm. A characterization of this new product was made on a mock-up and a set of these prototypes integrated in the Maestro slave-arm. A comparison between the two actuating technologies was made and showed that the performances of the pressure servo-valves can lead to generalize its use.

H-3

VACUUM VESSEL MAINTENANCE:
REVIEW ON LOGISTICS ASPECTS

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The logistic studies here presented aim at identifying and classifying the many possible configurations for the periodical Vacuum Vessel maintenance. The in-vessel components maintenance consists in their removal, transporting to the Hot Cell, refurbishing (or replacing) transporting back to the Vacuum Vessel and repositioning. Each operation has its own procedure, time duration and constraints, which must be taken into account when defining the maintenance process flow and schedule. Moreover, the process definition strongly depends on the resources which are devoted to it, in terms of number of casks, number of Work Stations and Test Tanks, docking stations at the Hot Cell, Hot Cell storage capability.

Based on different maintenance scenarios the following were estimated:

Cask transfer logic:
- cask transfer logic to minimise cask transfer time, including the possibility of parallel/series cask transfer operations
- minimum required number of transport casks, cask paths (main and alternate) and maximum number of casks required at any one time
- estimation of cask transfer time
- definition of the building space needed to accommodate cask circulation including the need for cask lay-by zones to avoid cask traffic jams
- identification of possible parallel/series transfer operations

Hot Cell
- specification of the number of cask docking stations
- estimation of number of and time required for docking/undocking operations and cask loading/unloading operations.
- number and type of components to be temporarily stored in the hot cell

The present study describes the various possibilities for the divertor and blanket maintenance process as relations between the needed resources and the obtainable performance.

H-4

BLANKET HANDLING CONCEPTS FOR FUTURE FUSION POWER PLANTS

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In the frame of the Power Plant Conceptual Studies (PPCS) launched by the European Commission, two main blanket handling concepts have been investigated with respect to engineering feasibility and the impact on the plant availability and on cost: the large module handling concept (LMHC) and the large sector handling concept (LSHC). The LMHC has been considered as the reference handling concept while the LSHC has been considered as an attractive alternative to the LMHC due to its potential of smaller replacement times and hence increasing the plant availability.

As a study case, a fusion power plant having a helium-cooled pebble bed (HCPB) blanket and 18 TF coils has been considered.

In the LMHC the blanket modules are handled and replaced by in-vessel remote handling devices moving toroidally on the divertor rails. The dimensions of the large modules are optimised in a way that they can be transported through the horizontal ports.

For the LSHC it is assumed that the power core consisting of combined blanket and divertor elements is divided in toroidal direction into several sectors equal to the number of TF coils. Each of the large sectors are replaced through the corresponding horizontal ports of the vacuum vessel having the size of a full sector.

The plant availability resulting from blanket replacement has been estimated assuming a lifetime of the blanket of 5 years and of the divertors of 2 years. Therefore in case of the LSHC the sectors have to be replaced every 2 years and refurbished in the hot cell according to the individual needs of blanket and divertor, respectively.

As a results of the investigations, although no feasibility issue has been identified, a number of engineering issues have been highlighted for the LSHC that would require considerable efforts for their resolution. An availability of about 80% has been estimated based on a replacement time for all the internals of about 90 days, which is only slightly better than the availability estimated considering the reference LMHC.

However, the increase in the investment cost mainly due to the larger port openings, to the larger PF coils, to the larger buildings and the relevant remote handling devices should reduce the attractiveness of the LSHC resulting from the higher availability.

H-5

WATER HYDRAULIC ACTUATORS FOR ITER MAINTENANCE DEVICES

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The characteristic advantages of hydraulics (high power density, simple construction and reliability) together with the characteristics of water as the pressure medium (fire and environmentally safe, chemically neutral, not activated nor affected by radiation) are highlighted in critical applications such as remote handling operations in ITER. However, lack of commercial selection of water hydraulic components, common design expertise and known application experiences prevents wide use of water hydraulics. However, since 1994 IHA has designed and manufactured water hydraulic actuated prototypic tools for ITER divertor maintenance. Experiences have been very good. Therefore IHA is developing water hydraulic component selection and preparing design guidelines for water hydraulics to be rapidly applied in any forthcoming system where water hydraulics is foreseen to provide advantage.

Aim of the project is to develop set of power units, control components and actuators and design guidelines and selection criteria for their use. By that way designers are able to apply water hydraulics where advantageous.

Under work are hydraulic cylinders with selectable sealing properties for heavy-duty applications and for low-friction force control applications. For that, the sealing studies are the main task. Also existing rotary actuators and their sealing solutions are studied. The need for new type of rotary actuators is considered. In the paper, sealing studies and developed tests systems will be explained and the results obtained are presented.

Water hydraulic power units exists for typical use. Under development work are compact portable power unit which can be integrated into manipulator tools. Also under development is power unit for variable fluid consumption, since small consumption typically causes fluid warming with traditional pump types. In the paper, pump designs are presented and their characteristics and measuring results are presented.

The existing control valves are tested and modified if necessary, and new control valves are developed to meet the requirements of ITER specific demands. The valves are characterised and also control electronics and algorithms are developed to meet the valve properties. In the paper the valve types, their characteristics and results obtained so far are presented.

H-7

MODEL-BASED REMOTE HANDLING WITH THE MAESTRO HYDRAULIC MANIPULATOR

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A powerful hydraulic manipulator may be a useful tool to deal efficiently with the remote handling dexterous operations required by the Divertor maintenance. Its main advantages are a good weight capacity, the large range of tasks that can be addressed and its capability to recover from unforeseen situations. On the other hand, it implies intensive video monitoring and it is well known that hydraulic technology is not currently suited to nuclear environments. This paper describes the developments and experiments jointly performed by CEA/LIST, IHA and ENEA in the frame of task T329-5 to enhance the benefits of computer-assisted hydraulic remote operation, while minimising its shortcomings.

A major difficulty that must be overcome is the degradation of video feedback due to the level of radiation encountered inside the reactor vessel. A relevant answer is a supervisory control scheme relying on a virtual computer model duplicating the real environment both to provide visual feedback and to support high-level target-oriented motions. The key points of this approach are a supervision interface implementing a robot graphical control language, the calibration of the manipulator arm and the registration of the environment objects wrt. the robot reference frame. Besides this main goal, significant efforts were also dedicated to three related topics : (i) water hydraulics that would hold more easily the contamination constraints of ITER, (ii) long distance teleoperation in order to allow an operator to control an intervention system located several hundred kilometres away and (iii) the development of a reliable hydraulic arm better suited to the operational needs than the current prototype.

These developments led to several feasibility demonstrations. The most striking was a representative dextrous maintenance operation performed with the full benefit of a virtual environment model. For this purpose, a MAESTRO master-slave hydraulic manipulator having a load capacity of 100 kg (equivalent to its weight) has been integrated inside the DTP (Divertor Test Platform) located at the ENEA Brasimone site. The complete mission was successfully performed at the end of December 2001. It shows that, although completely "blind" remote operation is not yet feasible, the need for camera feedback is significantly decreased, and new developments for improving this result are outlined. Other worthwhile results were the successful evaluation of a water hydraulic one-joint mockup developed by IHA, a demonstration of long distance (1000 km) remote operation and the specification of a new hydraulic arm that takes into account the first experimental lessons.

H-8

ITER HOT CELL

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The ITER plant layout will include a hot cell and waste treatment facility to allow the repair, refurbishment and waste processing of active components during machine operations. The hot cell building is designed for use also during the initial tokamak's internal component installation period, to provide a beryllium controlled area and a facility for loading components into transfer casks. The hot cell building is conceived such that it can be expanded to meet future demand for use in the decommissioning phase of ITER.

The hot cell facility is divided into specific areas: cask docking and components receiving/delivery area, repair/refurbishment and testing area, waste processing and storage area, remote handling equipment test stand area. The hot cell processing systems and their associated equipment are housed within the hot cell building which provides space, support, confinement and services such as lighting, power, HVAC, shielding, access, etc. The hot cell building is located on the site to facilitate the layout of the transportation system connecting the tokamak pit with the hot cell building.

A dedicated atmosphere detritiation system is permanently connected to the areas where chronic tritium offgassing from components occurs and can be automatically connected to other areas in case tritium is detected. The function of the system is to limit the tritium concentration in the hot cell atmosphere to low levels. After removal of components from rooms, the tritium content is reduced to very low levels in a short time so that personnel access, in full airline suits, is feasible if required for repair work. To limit the generation of tritiated water the amount of air in-leakage is reduced to a minimum by careful design of all penetrations into the hot cell. A vent detritiation system is used to maintain under pressure compared with ambient in the hot cell and to limit the release of tritium into the environment to a minimum.

The paper provides also a description of the ITER hot cell building and equipment design basis. This takes into account the logistics requirements based on a reference flow rate of components during maintenance periods. It is assumed that during these periods, full replacement of the divertor cassette system as well as the replacement of a number of shield blanket modules and port mounted components will take place. A brief description of the refurbishment concept for the divertor and blanket components is also provided.

H-9

THE USE OF VIRTUAL REALITY FOR PREPARATION AND IMPLEMENTATION OF JET REMOTE HANDLING OPERATIONS

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This paper discusses the experiences gained at JET in the selection and use of a Virtual Reality system for the support of remote handling operations. The use of real time 3-D computer graphic models for preparation and support of remote handling operations has been in use at JET since the mid 1980's, even before the now common term Virtual Reality (VR) was contrived. Recently, a complete review has been undertaken at JET of the functional requirements and benefits of VR for remote handling and a subsequent market survey of the present state-of-the-art of VR systems has resulted in the implementation of a new system at JET.

The VR system is used in two discrete modes:
In On-Line mode the remote handling equipment electro-mechanical hardware is connected to the VR system and provides input for the VR system to update a real time 3-D display of the equipment inside the torus. This mode is used during operations to supplement the video camera system information and for assisting with camera control and warnings of impending or potential collisions.
In Off-line mode the VR system model is manipulated by the operator using mouse and keyboard with no connections to the remote handling equipment. This mode is used during preparation of RH operational strategies, checking of operational feasibility and substantiation of operations procedures.

Various potential VR systems were evaluated against a detailed technical specification which covered visualisation function and performance, user interface design and base model input/creation capabilities. The cheapest of those systems which satisfied the technical requirements was selected.

The paper will describe the technical requirements of the new JET remote handling VR system, the rationale for selection of the VR system and the benefits that have been obtained from its use during both preparation for operations and during the remote handling operations.

This work was performed under the European Fusion Development Agreement.

H-10

DEVELOPMENT OF AN IMAGE RECOGNITION SYSTEM FOR RFX FIRST WALL MAINTENANCE

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The RFX first wall is covered with 2016 graphite tiles individually clamped onto the 72 vessel stiffening rings. Since the largest ports are the 12 vacuum pumping ports with a 150 mm diameter, a remote handling system was built to carry out maintenance tasks inside the vessel, such as tile replacement, first wall inspection, graphite dust and fragment removal. It consists of a 5 degree of freedom articulated arm with two end-effectors equipped with a TV-camera and a sliding transporter to insert the tile into (or extract from) the vessel. The system has been used many times: in particular, two complete inspections of the vessel inside have been carried out and more than 200 tiles have been replaced. In principle, the system could perform the tile replacement in a full automatic mode; as a matter of fact, the deviations of the relative positioning of the manipulator arm with respect to the vessel from the nominal conditions often required to complete the tile handling phase in a teleoperated mode. In order to speed up this phase and to increase its reliability and safety further, it was envisaged to develop a monoscopic vision system capable of recognizing the tile clamping key and the bush inserted in the vessel rings, i.e. the two objects on which the end-effector must act to dismount/mount a tile. The aim is to estimate the object pose with respect to the end-effector to drive it correctly to its target and to perform the handling task automatically.

When dismounting a tile, the object features to be detected are the central circular hole and the slits machined on the graphite cap of the tile clamping key, when mounting a tile, the feature is the key insertion hole in the vessel stiffening ring. The detection is carried out by means of an algorithm based on the Deformable Template technique. Active contour models are used, consisting of a curve that deforms and adjusts its shape to match the contour of the object to be recognized according to a minimun energy principle. The algorithm robustness and accuracy is improved by including a priori knowledge of the object shape and optical distortion compensation. Intrinsic difficulties of this application, such as possibly damaged edges and reflecting surfaces have been tackled and quite satisfactory results have been obtained. A very accurate calibration rig was built and extensive tests carried out to assess the achievable accuracy in the pose measurement. In conditions similar to the operating ones the maximum errors were in the order of 0.5 mm and below 1° in the position and in the orientation estimate, respectively, well within the specifications to perform the handling tasks successfully.

In the original end-effector assembly, TV-camera and lamp were in a fixed position, complying with the arm maximum diameter to allow the insertion/extraction of the manipulator through the pumping ports; a movable camera and light source are now mandatory to frame and light the target objects meeting the vision system requirements: the design of the new hand has already been completed and the modifications are going to be accomplished.

The paper will report on the used algorithms, the calibration rig and the hand modifications; test results and achieved accuracy will be discussed.

H-11

AM LASER SYSTEM (IVVS) FOR THE ITER IN VESSEL VIEWING AND RANGING

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In vessel inspection (viewing & ranging) systems, able to withstand the ITER very severe operating conditions, are fundamental to detect plasma facing components in order to properly programme ordinary and extraordinary maintenance activities. In this field different schemes have been proposed during the EDA ITER phase each of them showing problems in ranging and/or in radiation hardening and/or in additional illumination source… For this reason ENEA proposed the Laser In Vessel Viewing & Ranging System (IVVS) based on an Amplitude Modulated (AM) laser beam that scans the target surface being deflected on it by a mechanical scanning head (prism) in both toroidal and poloidal directions. The backscattered beam intensity is properly detected and the target image is then reconstructed. At the moment IVVS seems the only system under testing able to perform, at the same time, viewing & ranging under the ITER operating conditions (Table 1).

Table 1. IVVS Main Features

Characteristics	**Value**	**Dimension**
Laser beam wavelength	1550	nm
Laser beam amplitude modulation frequency	> 80	MHz
Field of Viewing and Ranging	2 – 7 (under the assumption to use 6 probes for the entire ITER)	m
Scanning accuracy	0,005	degrees
Viewing resolution	< 1	mm
Ranging accuracy	Better than 10^{-4} (i.e. 0.5 mm @ 5 mm)	
Viewing Time	30	μsecx pixel
Ranging Time	300	μsecx pixel
Operating temperature	250	°C
Operating vacuum	10^{-9}	mbar
Operating radiation level	30 $2x10^4$	MSv(total) Sv / h

IVVS is an hybrid laser radar including a coherent and an incoherent part. The laser beam is transmitted from the source (an 80 mW diode laser, linearly polarized beam modulated by means of an acoustooptic modulator, located outside the bioshield) through a coherent optical fibre into the IVVS probe integrating both launching and receiving optics in a very reduced space (Φ ~160 mm, L = 800 mm). The beam is focused on the target through the scanning prism can be rotated at a variable speed simultaneously in both toroidal and poloidal directions. The backscattered beam incomes through the scanning prism (coaxial optics) and it is collected by a large aperture (50 mm) multispeckle receiver and transmitted trough a bundle of coherent silica fibres. In this scheme the typical high intensity of a large aperture optics is jointed to a coherent speckle transmission, fundamental for ranging. The IVVS testing is scheduled to be completed within July 2002 at ENEA's Frascati Labs. The present paper will included the IVVS design principle and the main results of the optics scheme that is the essential and innovative part of the probe.

H-12

RADIATION ENHANCED DEGRADATION OF ALUMINIUM MIRRORS FOR REMOTE HANDLING APPLICATIONS: EFFECT OF HUMIDITY

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High quality mirrors for the optical UV - visible - IR range will be required in ITER for both remote handling applications and diagnostic systems. The reflectivity of the mirror surface may be degraded by many different mechanisms ranging from simple contamination, to sputtering / erosion due to particle bombardment, and enhanced radiation effects which may modify the metal or overcoating surfaces. The work reported here is concerned with the radiation enhanced degradation of aluminium surfaces in a humid environment, and is related to an earlier JA home team report on corrosion of aluminium mirrors. The possibility of mirrors being irradiated in a humid environment must be assessed due to loss of coolant events.

To examine the possibility of radiation enhanced corosion in a humid environment, initial tests were carried out irradiating uncoated polished aluminium discs at 300 K, 13 Gy/s up to 50 MGy in the CIEMAT ^{60}Co pool facility. Irradiations were done in humid air (80% RH), dry air (<10% RH), and nitrogen gas. Irradiation in humid air was observed to cause severe radiation enhanced corrosion due to the formation of white traces of $Al(OH)_3$. In contrast the discs irradiated in dry air and nitrogen showed no degradation. Unirradiated discs maintained in high humidity (saturated) air for the same period of time also showed no corrosion nor visible surface degradation. Clearly some type of radiation enhanced corrosion occurs in the presence of water vapour. Most mirrors will however be protected by a thin film coating, preferably a transparent oxide, but to date there is no data about their behaviour under irradiation.

Tests have been made on high quality overcoated mirrors ("Ealing" and "Coherent" research quality Al on Pyrex glass, SiO overcoated). Reflectivity measurements were made from 250 to 2500 nm. Irradiations have been performed at 323, 443, and 523 K, 11 and 13 Gy/s to 100 MGy in nitrogen gas. The Pyrex glass support turned black through radiation coloration for the 323 and 443 K irradiations, and dark brown at 523 K. For all these irradiations very limited degradation was observed in the mirror reflectivity. This will be discussed elsewhere. Finally one irradiation was performed at 523 K, 13 Gy/s, to 2 MGy in humid air, as above for the unprotected discs. It has been observed that humidity is a key factor in the radiation enhanced degradation process and leads to severe damage of the mirror reflectivity, even for the SiO protected mirror. In this case the aluminium surface corrodes even below the SiO protecting layer. SEM observation and XPS analysis of chemical species formed indicate $Al(OH)_3$ suggesting radiation enhanced diffusion of OH species through the SiO layer. The corrosion, which appears to correspond to a filiform corrosion mechanism, is sufficient to lift the aluminium and the protecting layer off the Pyrex glass surface.

H-13

UTILIZATION OF VIRTUAL PROTOTYPING IN DEVELOPMENT OF CMM

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The characteristic advantages of hydraulics (high power density, simple construction and reliability) together with the characteristics of water as the pressure medium (fire and environmentally safe, chemically neutral, not activated nor affected by radiation) are highlighted in critical applications such as remote handling operations in ITER. However, component cost and lack of wide selection of water hydraulic components makes it difficult to build and test complex water hydraulic systems. The use of virtual prototyping for the development of water hydraulic tools can be used to address this problem. Rapidly increased computational power has created conditions for extensive numerical calculations, enabling computer aided virtual prototyping to replace physical prototype phases in product development.

The task to be analysed is the transportation of the cassette through the service port using the CMM (Cassette Multifunctional Mover). The clearance between the cassette and the service port floor and ceiling is about 20 mm and collision between the cassette and the service port has to be avoided. The dynamic characteristics of the water hydraulic cylinders and valves are studied by building a non-linear hydraulic model and a simplified mechanical model (no flexibilities considered) and performing several simulations. In addition to the simplified model for the mechanical system of the cassette and the mover, a more complex model including FE-description of the cassette and flexible parts of the mover is built within the research. The complete model of the system including hydraulic and mechanical flexibilities is set up by employing MBS-simulation software and control system software. By means of this simulator the total dynamic behaviour including properties of hydraulics, mechanics and control system can be studied.

Simulation results are displayed using 3D models of the components which allows visualizing the interactions between the different components in a graphical way. The simulation can be performed in real time when using the simplified mechanical model. The 3D graphical user interface can also be used for optimizing and analysing the behaviour of the CMM. The simulation results allow analysing the coupling effects between the cylinders and the cassette oscillations due to sudden valve opening and closing. The amplitude and frequency of the oscillation will depend on several hydraulic parameters. As a result of the simulation, a set of suitable hydraulic parameters that fulfil the task requirements is obtained. The real water hydraulic components will be selected with the characteristics defined by the set of hydraulic parameters obtained with the aid of the simulation.

H-14

DEVELOPMENT OF A LONG REACH ARTICULATED MANIPULATOR FOR ITER IN VESSEL INSPECTION UNDER VACUUM AND TEMPERATURE

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This project takes place in the Remote Handling (RH) activities for the fusion reactor ITER. The aim of the R&D program is to demonstrate the feasibility of in-vessel RH intervention by a long reach, limited payload manipulator which penetrates the first wall using the six Ø150mm penetrations evenly distributed around the machine for access of the In-Vessel Viewing System (IVVS). Potential in-vessel activities for this device include close inspection of the plasma facing surfaces and leak detection.

The work performed under EFDA workprogramme includes the design, manufacture and testing of a demonstrator articulated manipulator called the In Vessel Penetrator (IVP).

The first part of this work concerned the analysis of the requirements to perform a realistic operation inside the Vacuum Vessel with access through the IVVS penetrations. This phase resulted in the development of the conceptual design of the overall manipulator based on three mechanical design concepts. The most promising of these designs, comprising a 5 module, 11 degrees of freedom robot based on a parallelogram structure (Ø150mm, 8.2m long) was further developed at assembly drawing level together with the detailed design of one representative segment. A scale one mock up of this segment was manufactured and tested during 2001, focusing mainly on the mechanical performance of the system and its ability to withstand the most critical loading conditions.

In parallel, a feasibility study of IVP operation under vacuum was made and provided recommendations to modify the design for intervention under vacuum and temperature. Some technologies were selected and analysed to determine their suitability to the IVP application and items identified for further validation.

This paper presents the whole robot concept, the results of the test campaign on the prototype demonstrator and the vacuum and temperature technologies study.

H-15

RADIATION TOLERANCE QUALIFICATION FOR MAINTENANCE TASKS IN THE FUTURE FUSION REACTORS: FROM FIBRE-OPTIC COMPONENTS TO ROBUST DATA LINKS

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The periodic maintenance operations of the future International Thermonuclear Experimental Reactor (ITER) will have to be performed in a severe nuclear environment, exposing operating tools to estimated gamma dose rates of about 10 kGy/h and temperatures ranging from 50°C to 200°C, with total doses that can reach 100 MGy [1, 2]. Remote-handling technology will therefore play a major role during these maintenance tasks. Moreover, both gamma rays and neutrons are considered, since equipment could be stored inside the bioshield during the reactor plasma burn. Connecting the remotely operated actuators and sensors with their control unit requires bulky and shielded umbilicals. Their management could be eased by applying radiation resistant communication links with multiplexing aptitudes.

The data link architectures envisaged up to now for ITER instrumentation are based on existing electronic technologies, using digital data transmission and time domain multiplexing. This approach requires the use of a large amount of radiation-sensitive electronics. Fibre-optic technology is being considered as a potential solution, since its intrinsic passive wavelength-division multiplexing (WDM) capabilities have shown promising results up to the required MGy dose levels [3]. However, the radiation tolerance assessment of all components is required before their use in instrumentation links for ITER. We describe our methodology to find the most suitable components with regard to these specific ITER requirements, considering fibre optics as a potential radiation tolerant alternative to conventional electronic components and systems [3]. We illustrate this with recent irradiation test results obtained for optical fibres, in-fibre wavelength-selective filters (Bragg gratings) and vertical cavity surface-emitting lasers (VCSEL) as case studies. Our approach aims to identify and describe the main failure mechanisms, to build a radiation-tolerant component database and to develop component validation procedures. Next, we propose a new analogue fibre-optic data link design based on COTS components [3]. It takes advantage of the multiplexing and demultiplexing resources using, respectively, fibre-optic wavelength-flattened and wavelength-selective couplers. In principle, it could reduce the required amount of radiation-hardened electronics to driving circuits for the optical emitter and detector. We compare the analogue and digital link architecture in terms of radiation-acceptance level, intrinsic performance and complexity.

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H-16

LASER MÉGAJOULES CRYOGENIC TARGET DEVICES AND THE ONE SCALE CRYOSTAT DEMONSTRATOR

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LMJ program claims to obtain Deuterium-Tritium (DT) mixture combustion leading to a fusion gain of ten. The cryogenic targets for inertial confinement, driven by 240 laser beams, are hollow spheres. Their internal shell are covered with a solid cryogenic fuel layer. The success of DT combustion depends on quality of the fuel layer geometry. Cryogenic targets must be cooled and kept at temperatures near the triple point (19 K) with a very good stability (0.2 mK) for many hours.

Besides that sharp thermal characteristics, the Cryogenic Target Carrier (CTC) displays others technical challenges like very strong mechanical specifications. The CTC deals with the target handling and positioning in the center of the 5 m radius experimental target chamber with a precision of few microns. This is done by an hexapode machine at the front end of a 7 meters long remote controlled carbon boom, with regards to the strong mechanical stability required.

Moreover, the mass and the composition of the materials used in the CTC nose have to be optimized in front of the heavy environment conditions generated during the laser shot (X rays, neutrons and charged particles radiations and high velocity flyer remains).

In order to validate our current CTC conception, we have computed X rays and 14 MeV neutrons effects on CTC structure to estimate potential performances decay and total occupational dose for workers. We are also manufacturing a One Scale Cryostat Demonstrator to confirm all CTC thermal challenges, such as sharp thermal regulation, cooling autonomy, cryogenic target transfers …

Some environmental calculations of shocks effects on CTC will be presented, as well as a general description of the new One Scale Cryostat Demonstrator in term of global characteristics and experiment schedule for the next three years.

L-1

ELEMENTARY TRITIUM CONTRIBUTION TO THE DOSES BY INGESTION AND RE-EMISSION FROM RELEASES IN INERTIAL FUSION REACTORS

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The evaluation of the radiological environmental impact of elementary tritium emission to the atmosphere on IFE reactors has different phases. The factors influencing the Primary phase (atmospheric releases) are boundary conditions such as atmospheric and geometric grid from the emission point. The Second phase occurs when the tritium is deposited in the ground, and it suffers the transformation rapidly to tritiated water (less than one week). This process depends on the kind of soils (clay, loam or sand), relative humidity and microporosity of soil. This phase is decisive in the dosimetric effects in elementary tritium and in the chronic and collective doses to population.

This Second phase is the goal of this work for selected IFE reactors (HYLIFE II, OSIRIS, SOMBRERO and CASCADE), by using MACCS2, UFOTRI and HOTSPOT8 codes for severe accidents and NORMTRI for normal operational conditions. For a realistic estimation of tritium deposition on the surface, we have performed computation using a radial grid of 100 km around the emission point with a population density of 100 persons.km^{-2}. The time dependent responses for the Second phase have been obtained along one year in steps of one hour, in accident and normal conditions.

This work will conclude that the two chemical forms (HT and HTO) contribute in a different way to the effective and equivalent individual dose (MEI-EDE), and to the chronic doses. Elementary tritium contributes as large as 90–98% to the total dose from ingestion of foods/natural agriculture and meat (HTO only 40%) and the rest of doses are from inhalation generated by the re-emission to the atmosphere. For comparison, HTO contributes 20–25% from inhalation and skin absorption to the total dose. We conclude that the elementary tritium is incorporated to the plants in a very efficient way and penetrates deeper in the soil when the porosity is large. The work also presents conclusions on the period in which the release is produced, indicating than the maximum toxicity appears when the accident happens in the growth season of the vegetables.

L-2

TIME-DEPENDENT DAMAGE IN STRUCTURAL WALL OF INERTIAL FUSION REACTORS

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Neutron pulses are a key characteristic of materials under irradiation in Inertial Fusion Energy (IFE) reactors. That peculiarity can generate a specific response, different from that of steady-state irradiation, when determining materials damage and activation.

New results on neutron intensities and energy spectra versus time after target emission will be presented showing differences between two protections (LiPb, Flibe), due to the differences in density, moderation and threshold reactions such as (n, 2n) and (n, 3n). The primary damage responses and impurities generation will also be presented versus time in a given pulse emission.

Using one of those time-dependent neutron irradiations in the structural wall, we present a multiscale modelling study of pulse irradiation in Fe, 1 and 10 Hz, up to the level of defect microscopic characterisation depending on time irradiation. A well established procedure is able to include with the appropriate dose rate (0.01 and 0.1 dpa/s) the cascades generated in time in a kinetic MonteCarlo (KMC) lattice (microscopic) to analyse the defects diffusion, clustering and disintegration. Final responses of the microscopic structure after irradiation will be given. A discussion on the cascades that are used in the calculation versus the established data bases will be given, showing the importance of the statistic in the generation of that input data for KMC. Different impurity concentrations are considered and its effect analysed. New approaches for description of diffusion coefficients and binding energies for impurities are included and their effect discussed. We will remark the established differences with a continuous irradiation for a still low fluence of irradiation. The experimental validation of the full methodology of Multiscale Modelling is a key issue and we will present experiments using Fe ions in ultra high purity Fe compared with calculations at the level of Transmission Electron Microscopy.

The significance of SiC as low activation material is well recognised but still is lacking an approach at the level of microscopy for Multiscale Modelling of this material. The most important reason is the absence of an appropriate description of existing defects and its diffusion. We consider as a first task the generation of a good data-base of defects energetic. Classical Molecular Dynamics has shown its weakness for that purpose. We are presenting new results conducting to calculation of defects energetic in SiC, by using a new code based on tight-binding molecular dynamics.

L-3

ADDRESSING THE ISSUES OF TARGET FABRICATION AND INJECTION FOR INERTIAL FUSION ENERGY

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Addressing the issues associated with target fabrication and injection is a major part of an international program to establish the feasibility of Inertial Fusion Energy (IFE), both for laser-driven and heavy-ion driven concepts. A description of the unique materials science and chemistry research associated with supplying targets for an IFE power system are presented, as well as analyses of the target response during injection, and cost modeling of the target supply process.

The "Target Fabrication Facility" of an IFE power plant must supply about 500,000 targets per day. Target fabrication has concentrated on investigating and developing the various materials needed by the target designs and on fabrication techniques that could eventually scale to low cost and high production rate. After manufacture, the target is injected into the target chamber at a rate of 5–10 Hz. The DT layer must survive the exposure to the extremely high heat flux and remain highly symmetric, have a smooth inner ice surface finish, and reach the chamber center at a temperature of about 18.5 K. Models of the thermo-mechanical effects on the advanced materials during injection have been developed. Fundamental measurements of the properties and response of DT under these unique conditions are being carried out. An experimental injection and tracking system is being constructed to develop technologies and to demonstrate meeting these challenging requirements.

We must also understand the issues that affect low-cost target production – studies have consistently shown that a cost reduction of at least four orders of magnitude from current technologies will be needed for future electricity production. This issue is often considered to be one of the major feasibility issues for future commercial application of inertial fusion. We have now, for the first time, prepared an engineering analysis of all of the process steps needed to mass-produce direct drive targets having gains suitable for commercial fusion in a laser driven system. We have performed this cost analysis based on a commercial process plant environment. This modeling of target fabrication includes process flows, mass-energy balances, plant utilities, raw materials, quality control, waste handling and recycle, capital equipment cost amortization, and staffing requirements. Inputs to the model show that the future cost goals of less than about $0.25 per target can be met. This highly encouraging result is a major breakthrough in addressing a major technical feasibility issue of IFE.

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L-4

NEW CONSTRAINTS FOR PLASMA DIAGNOSTICS DEVELOPMENT DUE TO THE HARSH ENVIRONMENT OF MJ CLASS LASERS

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Up to now all the diagnostics developed for laser produced plasmas were mainly designed without taking into account direct effects of radiated or nuclear energies emitted by the plasma itself on the diagnostic active components. The designs of plasma diagnostics for the future MJ class lasers (LMJ in France or NIF in USA) will be dramatically affected by these new environmental effects. These facilities, are able to focalize up to 1.8 MJ UV laser light (and even equivalent unconverted light for NIF only) into a volume of a few mm^3. This high energy density focused on a small target will drive a large amount of x-rays, debris, shrapnel and nuclear particles (neutrons and gamma rays) when DT filled glass microballon implosion experiments are performed. Unfortunately this colossal energy increase with respect to the old laser facilities (Phébus in France or NOVA in the US – respectively 6 kJ and 30 kJ of UV laser light) cannot be compensated by an equivalent increases of the emitting region dimension, which remains nearly unchanged (core diameter of a few tens of microns). The spatial resolution of main core diagnostics must be conserved and the diagnostic will always need to be nearly as close to the plasma as in the past. The harsh environmental background, fixed by the close proximity of the diagnostics, must be taken into account not only for the DT ignition or gain target (a gain of 10 produces a neutron emission of about 10^{19} n/4π) but for many other ignition and gain preparation experiments where neutron level are within 10^{14} up to 10^{16} n/4π. For these types of preliminary experiments all the main diagnostics used in present day experiments must continue to operate and accurately measure without being affected by this new environment. This diagnostic survivability goal is not obvious and a lot of new studies must be conducted to verify which diagnostic measurement principle can be conserved, adapted or dramatically changed under these environmental conditions.

Simulations and experiments conducted at CEA/DAM related to the preliminary studies of this harsh environment will be presented. Recent experiments performed at the Omega laser facility at the University of Rochester (USA), including neutron effects on CCD readout and EMP measurements, will be fruitful for the future MJ class laser facilities diagnostic design.

In addition some new measurement strategies especially developed to overcome these environmental difficulties for diagnostic design (grounding, shielding, high bandwidth signal transfer, remote imaging devices, new detector studies, …) will be briefly presented.

L-5

TUESDAY, 10th September

Session 4: Plasma Facing Components
Chairman: A. Pizzuto

Session 5: Plasma Heating and Current Drive
Chairman: H. Zohm

Session 6: Fuel Cycle and Remote Handling
Chairman: S. Gross

Poster session P2
Chairmen: Prof. Tran, M.Q. and Dr. Karttunen, S.

Poster presentations

EXPERIENCE GAINED FROM SERIES MANUFACTURING OF ACTIVELY COOLED PLASMA FACING COMPONENTS AND THEIR OPERATION ON TORE SUPRA

Tore Supra Team, presented by J.J. Cordier

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The design and fabrication of large areas of actively cooled Plasma Facing Components (PFCs) is a major issue for the next generation Tokamaks. Tore Supra is currently the only large fusion device which has implemented actively cooled PFCs from the beginning of operation in 1988, while maintaining continuous development activities to improve their performances and reliability.

Following a difficult start-up using graphite brazed material on complex shaped metallic heat sinks, R&D, in tight collaboration with European industry, led to a variety of promising new PFC concepts. This R&D was aimed primarily on developing a reliable bonding joint between the tile material and the metallic heat sink. The subsequent generation of PFCs, used carbon fiber carbon (CFC) flat tiles on a flat and modular stainless steel heat sink with an improved brazing technology, including systematic inspection methods during and beyond manufacturing. This design remained a medium heat flux PFC (1–2 MW/m^2) and behave satisfactory during plasma discharges supported by the Inner First Wall (2 min. duration record discharge obtained in 1996 on Tore Supra with 280 MJ of injected energy).

Since 1992 high heat flux PFCs based on copper heat sink structures have been developed in order to enable a large increase of power extraction capacity. The result is a high heat flux “finger” element (capable of removing up to 10 MW/m^2) used first for the RF antennas edge limiters and then for the Toroidal Pump Limiter (TPL), which is the main part of the Tore Supra upgrade of in-vessel components (CIEL project). This structure is made of about 620 actively cooled elements. Reliability problems appeared during series manufacturing of a so large number of high heat flux elements that finally led to the improvement of a tile attachment repair process. The final delivery of the TPL fingers has now been completed and the whole limiter has been recently installed inside the Tore Supra inner vessel.

A first experimental campaign has been performed in 2001 using three limiter sectors (one fourth of the full set of TPL fingers). In 2002, Tore Supra operation is aiming 1 GJ injected energy discharges through the completed actively cooled TPL structure (i.e.3.5 MW of injected power, 300s at least).

The paper reviews the improvements achieved during the successive development phases of the actively cooled PFC technologies employed in Tore Supra, the different problems which were solved and consequences on based components. Hereby non-destructive inspection methods played an important role for component control. Test beds which have been set up for this purpose are reported. The recently completed CIEL in-vessel components are described. PFCs behavior results obtained in 2001 and 2002 during plasma operation in the Tore Supra new configuration is also reported. Lessons for future realizations from these more than ten years developments, in particular for ITER, are discussed.

IN-7

EUROPEAN CONTRIBUTION TO THE DEVELOPMENT OF THE ITER DIVERTOR

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The present paper summarises the number of activities carried out by the EU Participating Team in support of the development of the divertor, which is one of the most challenging components of the next step ITER machine.

One of the most impressive achievements was the development of suitable technologies for the production of high heat flux components with both CfC and tungsten armour joined onto a copper alloy heat sink, namely CuCrZr. This long lasting effort culminated with the manufacturing of near full-scale prototypes for the vertical target and the dome/liner. Two possible joining technologies have been developed, hot isostatic pressing and brazing. The main issues, which needed to be addressed, were the large thermal expansion mismatch between the armour and the heat sink, as well as the preservation of the thermomechanical properties of CuCrZr. A number of high heat flux tests have been carried out in electron beam test facilities on mock-ups and prototypes to demonstrate the capability of the developed technologies to meet or exceed the ITER requirements. In parallel to that, suitable non-destructive inspection techniques have been developed to check the quality of the joints during manufacturing and on the final components. Tests on calibrated joint defects have been performed to determine the acceptance criteria. An adequate Quality Assurance scheme has also been implemented throughout the entire production. CfC composite materials, with a 3D fibre structure, have been developed with high thermal conductivity and an optimised mechanical strength. The R&D efforts are now focusing on the optimisation of the armour to heat sink joining technologies to improve the reliability and decrease costs as well as on the development of suitable repairing methods. The European industry is also being involved in the assessment of the proposed design solutions for the divertor structures with the aim of identifying and optimising possible manufacturing routes.

Four neutron irradiation campaigns have been performed. Small-scale mock-ups and material samples have been irradiated up to 0.35 dpa at 350 and 700 °C as well as up to 0.2 and 1.0 dpa at 200 °C. Tests are carried out before and after irradiation to study the evolution of the joints and of the material properties.

A vast database on critical heat flux has been established on different cooling schemes using both metallic and prototypical mock-ups. The most suitable schemes proved to be the twisted tape insert (twist ratio 2) for the inner and outer vertical target and the hypervapotron cooling for the dome. These design choices ensure a margin against the critical heat flux event above 1.4 in the worst case. An integration and hydraulic test on full-scale divertor components was also carried out to evaluate the pressure drop and flow distribution in a representative cooling configuration.

Design activities have and are being carried out and include neutronic and electromagnetic analyses and thermal and mechanical calculations. The aim is to define a comprehensive design of the ITER divertor taking into account all the prescribed requirements.

BO-2

ENGINEERING DESIGN OF THE KSTAR PLASMA FACING COMPONENTS

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The PFCs (Plasma Facing Components) of KSTAR (Korea Superconducting Tokamak Advanced Research) consist of the divertor, inboard limiter, passive stabilizer, neutral beam armor, poloidal limiter, and in-vessel cryopump. Basically the PFCs are designed for the KSTAR baseline operation which is characterized by 20 sec pulse length and 16 MW power, however, inboard limiter and passive stabilizer can accommodate upgrade operation conditions of 300 sec pulse length with 27.5 MW power.

Each component, except for in-vessel cryopump, has bolted graphite or carbon-fiber-composite (CFC) tiles supported by SS 316LN (for the divertor, inboard limiter, neutral beam armor, and poloidal limiter) and CuCrZrMg (for passive stabilizer) backing plates. The backing plates are attached to the vacuum vessel inner wall through the PFC supports except for the poloidal limiter which resides on the mechanical support of passive stabilizer. The baseline PFCs are water-cooled during plasma operation to maintain the surface temperatures of graphite and CFC tiles to be less than 600°C and 1200°C, respectively. The baking temperature of the PFCs, 300°C, can be achieved within 24 hours. Baking medium is nitrogen gas. Coolant and baking gas requirements have been obtained and the baking/cooling channel design has been done. Structural analyses for the situations of plasma disruption, coolant/baking gas pressure, bakeout, and earthquake have been carried out using ANSYS code. Thermal analyses on the carbon tiles have also been performed to determine the tile size and the required thermo-mechanical properties. All components are divided into small sectors and connected by bolts to be delivered into and to be assembled in the vacuum vessel through the main entry port of 788 mm diameter. In-vessel cryopump of over 50 Torr•l/s at 1 mTorr will be installed in the divertor pumping plenum. The cryo-surface temperature of less than 4.3 K is maintained with 3.7 K two-phase liquid helium and regeneration will be done within 20 minutes for 20 sec pulse length of baseline operation.

So far the engineering design and analyses have been carried out. The interface issues relevant to in-vessel control coil, diagnostics, fuelling, pumping, heating, etc. have been solved. The prototype of one sector of inboard limiter, which is the only one component of PFCs to be installed by the first plasma planned at the end of 2005, was fabricated and tested.

BO-3

STUDIES OF IMPURITY DEPOSITION/IMPLANTATION IN JET DIVERTOR TILES USING SIMS AND ION BEAM TECHNIQUES

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Deposition in JET divertor tiles has been observed to be asymmetric; i.e. heavy deposition occurs in the SOL at the inner divertor whereas there is little deposition at the outer divertor. Heavy deposition at the inner divertor has lead to flaking on the water-cooled louvres and after the DTE1 tritium experiment at JET it was observed that majority of the retained tritium is in the flakes that have spallen from the louvres. Erosion/deposition issues have thus a great impact on the tritium retention and lifetime of first wall components in present and future fusion devices.

The asymmetry in the deposition implies a drift in the SOL. To investigate material transport and SOL flows observed at JET, $^{13}CH_4$ and SiH_4 were injected into the plasma boundary in the last day of discharges prior to the JET 2001 shutdown. Methane was puffed at the top of the vessel, and the silane at the outer divertor wall. Silicon was observed spectroscopically as an impurity in the plasma. A set of special tiles was installed in the Mk IIGB divertor during the 1999 shutdown. The tiles were coated with poloidal stripes of a boron doped carbon layer on a rhenium interlayer. These tiles were removed in the 2001 shutdown, and have been analysed in 2002 to determine the erosion/deposition profile, and ^{13}C and ^{28}Si distribution over the entire divertor cross-section. The analysis techniques used have been Secondary Ion Mass Spectrometry (SIMS) and the Ion Beam Analysis (IBA) methods Time of Flight Elastic Recoil Detection Analysis (TOF-ERDA) and Rutherford backscattering (RBS).

^{13}C could be found in large amounts on the surfaces of the inner divertor tiles whereas in the outer divertor it was hardly detected. Under the ^{13}C layer a deuterium rich carbon layer was found on the inner divertor tiles. Thick deposits of beryllium were buried under the carbon layer. Also the silicon distribution was recorded. The observed ^{13}C and Si distributions are discussed in relation to the position of strike point and the ion fluxes measured with divertor Langmuir probes.

This work has been conducted under the European Fusion Development Agreement and is partly funded by EURATOM.

BO-4

PROGRESS IN GYROTRON DEVELOPMENT

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Gyrotrons are microwave oscillators based on the Electron Cyclotron Maser (ECM) instability. The free energy is the rotational energy of weakly relativistic electrons ($1 < \gamma \leq 1.2$) in a longitudinal magnetic cavity field. In contrast to klystrons the interaction circuit is a high-order-mode cavity allowing higher power at higher frequencies.

At present, gyrotrons are mainly used as high power microwave sources for various applications in tokamak and stellarator plasmas for controlled thermonuclear fusion research. Long-pulse (a few sec) gyrotrons utilizing open-ended cylindrical resonators which generate output powers of 100–960 kW per unit, at frequencies between 8 and 159 GHz, have been used very successfully for plasma formation, electron cyclotron resonance heating (ECRH), noninductive electron cyclotron (ECCD) and lower hybrid current drive (LHCD), plasma stability control and active plasma diagnostics at system power levels up to 4 MW.

As fusion machines become larger and operate at higher magnetic fields ($B \cong 6T$) and higher plasma densities in steady state, it is necessary to develop CW gyrotrons that operate at both higher frequencies and higher mm-wave output powers. The requirements of the stellarator W7-X and the tokamak ITER are between 10 and 40 MW at the frequencies 140 GHz and 170 GHz, respectively. This suggests that mm-wave gyrotrons that generate output power of at least 1 MW, CW, per unit are required. Since efficient plasma applications need axisymmetric, narrow, pencil-like mm-wave beams with well defined polarization (linear or elliptical), pure-mode gyrotron emission is necessary in order to generate a TEM00 Gaussian beam mode. Single-mode 110–170 GHz gyrotrons with conventional cylindrical cavity, capable of 1 MW per tube, CW are currently under development in several scientific and industrial laboratories. The maximum pulse length of 1 MW gyrotrons (110 GHz, 140 GHz, 170 GHz) with synthetic diamond window is 10 s, with efficiencies slightly above 30%. The energy world record of 90 MJ (0.64 MW at 140 s pulse length and 140 GHz) has been achieved by the European FZK-CRPP-CEA-TED collaboration. Total efficiencies around 50% have been obtained using a single-stage depressed collector (SDC). The present state-of-the-art will be discussed in this invited paper.

To reduce the costs of the ECRH system on ITER and to make its poloidal launcher for neoclassical-tearing-mode stabilization more compact, a further increase of the output power per gyrotron is desirable. To achieve output powers in excess of around 2 MW, CW at the ITER reference frequency 170 GHz it is necessary to switch to coaxial cavity geometry. A maximum output power of 2.2 MW at 165 GHz (1ms pulse length) was obtained at FZK with an efficiency of 28%. At the nominal output power of 1.5 MW the efficiency increases from 30% to 48% in operation with a SDC. The availability of sources with fast frequency tunability would permit the use of a simple non-steerable mirror antenna at the plasma torus for local current drive experiments. This work also reports on the status of the development of advanced coaxial cavity gyrotrons and step-wise frequency tunable gyrotrons.

IN-8

HIGH POWER DENSITY AND LONG PULSE OPERATION WITH TORE SUPRA ICRF FACILITY

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Ten years after the first Ion Cyclotron Range of Frequency (ICRF) heated Tore Supra (TS) shot, and as TS facility, now equipped with most of the new CIEL high power handling internal components, is progressing towards long pulse performances, the main results and experience gained from the ICRF system are reviewed.

One of the specific features of TS ICRF system is that each of the 2 straps of the 3 plug-in antennas is inserted in a compact resonant circuit made with two vacuum sealed variable capacitors from COMET. The main interest is to limit the volume where high RF voltages occur, thereby reducing the risk of voltage limitations in general and minimising the power losses and voltage stand-off requirements in the feed line, especially at the vacuum feedthrough. The experimental result is indeed that high power density can be achieved, up to 16 MW/m^2 (while taking the FS surface) or 10 MW/m^2 (while taking the port surface). Not only the voltage limitations issue in the feed line is driven away, but also the voltage standoff in the resonant part of the antenna is quite high: many multisecond pulses were performed with antenna voltage in the range of 40 kV, and up to 60 kV for some short pulses. From the large TS ICRF shot database, with a wide range of coupling conditions, both a global and performance focused analysis will be reported.

Other key features of TS ICRF facility will be shortly reminded, while focusing on the main modifications made from the initial plant and associated results, in particular in the area of feedback control.

Finally, in the objective of extended pulse length in Tore supra, main present limitations will be reviewed and recent high energy pulses performed will be reported.

CO-5

TRITIUM INVENTORY CONTROL – THE EXPERIENCE WITH DT TOKAMAKS AND ITS RELEVANCE FOR FUTURE MACHINES

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At present, the commercial use of tritium is relatively small scale. The main source of supply is as a by-product of heavy water moderated fission reactors and the products are mainly discrete sources or tracers with activity typically in the GBq range. There are in general no restrictions on the use of tritium other than those which would normally apply to the use of radioactive material.

The future use of tritium as intermediate fuel for a fusion power plant series will involve an increase by several orders of magnitude in the industrial use of tritium and may increase concerns relating to safety, transport and waste disposal. In addition, the use of tritium in fusion power will be unable to be satisfied by current sources of supply and tritium re-generation will be essential. Power plant studies have however shown that these issues can be satisfactorily addressed and that in particular there is no possibility of a severe accident. In addition the values for clearance of tritiated materials in a number of countries are consistent with the low environmental impact of disposal of tritiated waste.

There are however many practical operational and regulatory problems which will need to be solved in the context of the current experimental programmes.

The current regulations for control and accountancy of tritium inventory, as applied internationally and in relevant countries, are reviewed and their influence on the DT fuel cycle considered. The effect of safety case limits on the need for control of tritium inventory in TFTR, JET and ITER is analysed. The sensitivity of the fuel cycle to tritium inventory is considered.

The experience of controlling tritium inventory in TFTR and JET is reviewed and the latest results from JET presented. This takes into account the limits and constraints, the differing requirements for tritium processing, in-vessel retention, the needs for waste management and decommissioning including detritiation, and techniques for measurement. Comparisons with ITER requirements are made and the desirable areas for further development identified.

Issues of tritium inventory control which could influence the viability of the DT fuel cycle are discussed. and priorities for action proposed.

This work was performed under the European Fusion Development Agreement.

IN-9

REMOTE OPERATIONAL TRIALS WITH THE ITER FDR DIVERTOR HANDLING EQUIPMENT

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The successful result of a major initiative over the period 1996–2000, the ITER 1998 FDR design Divertor Test Platform is located at ENEA's Research Centre in Brasimone, Northern Italy, with the conception, design, manufacture, installation and commissioning of the divertor cassette handling equipment already well reported. Following the introduction of the smaller dimensioned and lower cost ITER FEAT design in 1999, many major aspects of this equipment have subsequently changed. However, the nature and principles of the remote handling equipment are still very similar, and hence there are still many valuable lessons to be learned for the future from the existing equipment. In particular, the ability of the installed systems to perform their planned function in a true remote environment is perhaps the final and most appropriate work which remained to be tested in the DTP at Brasimone.

This paper will describe and document a series of three, discrete, true remote handling trials carried out using most of the major DTP subsystems. Chronologically these are:

- Installation and removal of captive bolts and rail sections using the Duct Vehicle System
- Basic installation and removal of the second cassette using the Second Cassette Carrier (SCC) and Tractor (TRC) units
- Detailed installation and removal of a distant regular cassette using the Cassette Toroidal Mover (CTM), Radial Cassette Carrier (RCC) and TRC, together with the MAESTRO hydraulic servo-manipulator arm.

In each case, a draft procedure representative of the tasks to be performed by the system under test was defined, followed by preparations and local trials to ensure the procedure was achievable hands-on. Then the task environment was organised to allow the work to be performed remotely, including selection and reasonable placement of video cameras – for example, no fixed, in-vessel mounted cameras were allowed. The operator environment was then reviewed and optimised before carrying out dry runs of the procedure remotely, using all the facilities of the remote environment, but with local support if necessary. From that point on, no operator access to the work environment – including direct vision – was allowed, unless absolutely essential, e.g. a failure which without local intervention would prevent further progress. The predefined trials were finally performed by one or two operators, just once if it was complex, or repetitively where the task was simple. At the same time, a record was kept by an observer, describing in detail the task execution and timing, problems (in particular failures) which arose and their solutions if resolvable (including local intervention), and suggestions for improvements to the equipment or procedure.

The paper describes in detail each of the trials, with preparations and results, and presents an overview of the conclusions and suggestions for future development of ITER cassette remote handling equipment.

HO-6

PROTECTION SYSTEM FOR THE GYROTRONS AT TJ-II

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Two 500 kW gyrotrons are used in ECRH experiment at TJ-II device. High voltage low output capacity power supply based on fast switching high frequency technology feeds the tubes. The limit of 1 Joule in dissipated energy should not ever be overhead when breakdown occurs in any gyrotron. Exceeding of this value leads to the lifetime reducing or total damage of the device.

Analyze of destructive energy different components and design concept for the gyrotron protection has been done. Energy dissipation calculation shows the necessity of additional protection even in case the switching-off time of the power supply is about 5...10 μs. Thus, crowbar protection is highly recommended even in this case. State of the art in high voltage switches that can be used for this system is done. Triggered vacuum switches are chosen like most suitable devices. Protection systems using off two and three RVU-31 vacuum switches in series are designed. It allows of 10 kA short circuit current and of 80 kV (130 kV) in hold-off voltage. Switching-on time is less than 1μs, total time delay is less than 3...5 μs. The system consists of main switch, triggering unit and arcing sensor to detect the current or current derivation excess. Results of design and HV tests are presented.

Simple and cost effective decision for the load protection can be effectively used in a fast operating HV power supplies when long distance cable transmission line should be installed.

C-1

DESIGN AND R&D FOR AN ECRH POWER SUPPLY AND POWER MODULATION SYSTEM ON JET

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Introduction

An ECRH (electron-cyclotron resonance heating) system has been designed for JET in the frame-work of the JET-Enhanced Performance project (JET-EP) under the European Fusion Development Agreement (EFDA). Due to financial constraints it has recently been decided not to implement this project. Nevertheless the design work conducted from April 2000–January 2002 shows a number of features, which can be relevant in preparation of future ECRH systems such as the ITER one. The system was foreseen to comprise 6 gyrotrons, 1 MW each, in order to deliver 5 MW of mm-wave power into the tokamak plasma.

System description

The main components of the ECRH system are:
Gyrotrons with depressed collector, each delivering 1 MW of mm-wave power, 10 s pulse duration at a frequency of 113 GHz.
Main power supplies 60 kV / 100 A thyristor controlled with solid-state crowbars.
Fast IGBT switches, connected in series with the gyrotrons, to disconnect the gyrotron from the power supply within 2 µs. to limit the energy deposit into an internal arc.
To modulate the full power of the gyrotron, the IGBT switch has to be capable to switch on and off with repetition rate of a few kHz. A frequency of 10 kHz. at a voltage of 70 kV and a current of 35 A was already demonstrated at JAERI.
Fast solid state high voltage amplifier with a voltage swing of 30kV, to enable a stabilized accelerating voltage at the body of the gyrotron and for fast RF power modulations up to 10 kHz, is being designed and tested at the TREK company in the USA.

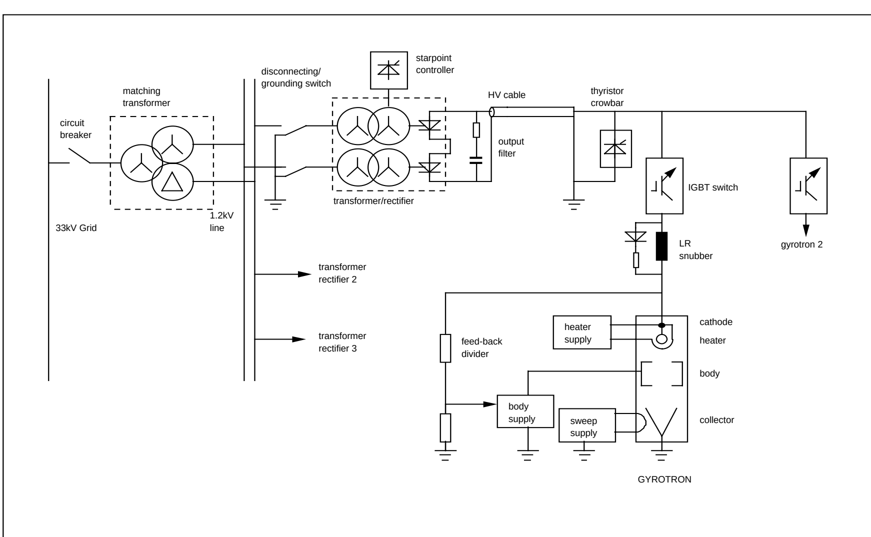

Fig. 1. The high-voltage power supply system.

The paper will describe the high voltage system, especially the fast series switch, the solid state crowbar and the solid state body supply.

C-2

DEVELOPMENT OF EC H&CD LAUNCHER COMPONENTS FOR FUSION DEVICE

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RF beam steering is a critical function of electron cyclotron heating and current drive (EC H&CD) system for advanced scenario of fusion plasma. In ITER, a front steering type launcher is designed for the injection from an equatorial port and a remote steering launcher is considered for the injection from an upper as the reference design.

Front Steering Type Launcher

Equatorial EC H&CD launcher with the front steering concept for ITER, which consists of the movable mirrors placed in front and the dog-legged waveguides, has been designed. For the purpose to confirm the reliability of the concept, the launcher mock-up, based on the front steering design, have fabricated and the availability of the flexible pipe and the steering mechanism has been studied. It has a steering mirror with a flexible cooling pipe, a drive shaft and a driver (air cylinder). The stress measurements of the pipes, made of stainless steel, were carried out under the ITER operational condition. Maximum stress of 60 MPa on the pipes, much less than the allowable stress, was obtained and consistent with the calculation.

Neutron irradiation effects of bearings for a movable mirror were investigated for the reliable launcher design. Stainless steel with high content manganese and tungsten carbide (WC) alloy were selected as the bearing material. The bearings were irradiated at JMTR (JAERI Material Test Reactor) and the post-irradiation tests were carried out. Neutron fluence of the bearings was 10^{24}n/m^2, which are the same as the annual flunece in ITER. No degradation for the rotational movements was obtained although the torque increased to double. It was verified that the bearings were available under the assumed irradiation condition in ITER.

Remote Steering Type Launcher

A remote steering launcher has been considered as an alternative in ITER. The launcher mock-up with this concept was fabricated to study the capability of rf transmission and radiation at high power and long pulse. Transmission and radiation of 0.5 MW-3.0 sec and 0.2 MW-10.0 sec at 170GHz was performed over a steering angle range of 0–10°. Transmission loss of less than 5% in the launcher waveguide was measured, calorimetrically. No arcing and no damage to the waveguide were observed. The new launcher which has the capability of wider range steering of rf beam and the corrugated square waveguide with four miter bends that form the dog-legged structure has been designed and fabricated for high power experiments, based on the test results.

C-3

TOWARDS A 2 MW, CW, 170 GHz COAXIAL CAVITY GYROTRON FOR ITER

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The development work on coaxial cavity gyrotrons at the Forschungszentrum (FZK) Karlsruhe has been performed as an ITER task. In agreement with the final goal of the task the feasibility of manufacturing of a 2 MW, CW coaxial gyrotron operated at 170 GHz as could be used for electron cyclotron heating and current drive in the ITER tokamak has been demonstrated and all information necessary for a technical design and industrial manufacturing has been obtained.

The experimental investigations have been performed on a coaxial gyrotron operated in the $TE_{31,17}$ mode at 165 GHz and designed for a nominal RF output power of 1.5 MW. The gyrotron is of modular type and its technical performance limits the operation to short pulses. With this gyrotron a maximum RF output power of 2.2 MW has been reached in short pulse (typically 1 ms) operation. At the nominal RF output power of 1.5 MW an efficiency of 30 % has been obtained which has been enhanced to 48 % in operation with a single-stage depressed collector. All problems specific to the coaxial arrangement have been investigated.

- The stability and the losses of the coaxial insert has been measured and the influence of misalignment of the insert on the losses has been investigated.
- The amount of microwave radiation captured inside the tube has been found to be (9±1) % of the RF output power. According to the measurements the captured stray radiation is distributed more or less uniformly inside the mirror box.
- In single-pulse operation the pulse length has been extended up to 17 ms. A sudden increase of the current I_{ins} to the coaxial insert has been observed which limited the achievable pulse length. It has been found that the increase of I_{ins} is caused by a kind of Penning discharge in the technical part of the electron gun. The occurrence of such a discharge can be avoided by a proper modification of the gun geometry.
- Parasitic low frequency oscillations have been successfully suppressed and stable operation has been achieved over a wide range of parameters.

By applying a rapid variable bias voltage at the coaxial insert fast (~0.1 ms) frequency tuning has been demonstrated. In particular, step frequency tuning by ± 2.2 GHz due to switching from the nominal mode at 165 GHz to its azimuthal neighbors has been done and continuous tuning by up to 70 MHz within the bandwidth of the $TE_{31,17}$ mode has been performed.

Based on the experimental results a draft design of a 170 GHz coaxial gyrotron for 2 MW, CW output power has been done. The usability of the components for CW operation and their compatibility with technical restrictions has been proven. A suitable cavity mode has been selected and a collector has been designed. In addition, an integrated design of the tube has been performed.

C-4

THE MAIN MICROWAVE COMPONENTS OF THE LHCD SYSTEM FOR ITER

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This paper outlines the preliminary radiofrequency analysis of the prominent microwave components of the LHH&CD system for ITER; its general overview and inclusive analysis will be given in a companion paper in this Conference [1].

The system is conceived to inject into the plasma an overall RF power of 20 MW, at a frequency of 5 GHz, through a single launcher based on 4 Passive Active Multijunction (PAM) modules. The launcher, with a built-in 270° phase pitch between active waveguides, has a $N_{\parallel\ peack} = 2$, with the possibility to vary the $N_{\parallel}$ in the range 1.9÷2.1 when the phase between modules in the same horizontal row is varied between ±90°. Its maximum directivity is about 70%. The RF power density in the active waveguides at the mouth is 33 MW/m^2.

The RF power produced by 24 klystrons (1MW CW), arranged in 6 identical subsystems, is carried to the launcher over a distance of about 60 meters. The 6 Main Transmission Lines (MTL) use circular oversized waveguides (mode TE_{01}) to reduce the transmission losses. One of the most critical components is the Combining Network (CN) that couples the four rectangular waveguides of each LH subsystem to a single MTL; the same component can behave as a Splitting Network (SN) to re–distribute the RF power of each MTL again into four rectangular waveguides. Each of them, through a TE_{10} to TE_{30} mode converter followed by an H–plane power splitter (4.77 dB), feeds the 24 active waveguides (58 × 9,25 mm^2), distributed over 3 rows, of a PAM unit. The RF power distribution in the 8 active waveguides of each row is achieved through conventional E–plane bi–junctions. A PAM module includes 12 PAM units arranged in 4 horizontal rows; in each row 25 passive waveguides (58 × 7.5 mm^2, depth.= 15 mm) are inserted at the mouth into the massive (13.5 mm thick) walls between active waveguides.

A potential configuration of a generic row of a PAM unit is also discussed in the paper.

This preliminary analysis shows that the system is technically feasible even if for some components alternative solutions must be explored.

[1] Overview and preliminary analysis of the ITER-FEAT LH system.
Ph. Bibet, F. Mirizzi et al., this Conference.

C-5

DEVELOPMENT OF LARGE RF DRIVEN NEGATIVE ION SOURCES FOR NEUTRAL BEAM INJECTION

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The goal of this development project is to demonstrate a H^- current density of 30 mA/cm^2 over an extraction area of 370 cm^2 for long pulse operation. This is a critical first step towards a full size source for the NBI system of ITER.

The present source design is based on the concept of small RF driven cylindrical sources, mounted onto the back plate of a 30 l magnetically confined expansion volume. For most of the past experiments a source with only one such driver has been used, recently tests of a source with two smaller drivers driven in series have begun. The objective is to demonstrate the modularity of this concept, which enables the extension to arbitrary source size. The influence of strength and shape of the magnetic filter field on the H^- current are being studied by using permanent magnets or external and internal electromagnetic coils respectively. There is a considerable range of the field from the electron suppression magnets into the source, causing close to the plasma grid surface a complicated superposition with the magnetic filter field.

As an alternative to inductive coupling of the RF power into the driver a RF driven hollow cathode discharge (HC) is being investigated. Homogenous plasma density profile could be generated by the arrangement of numerous small HCs in the back plate of the source. Another potential advantage of HCs is the enhancement of the density of vibrational excited H_2 molecules by surface conversion of molecule ions and recombination of atoms respectively. In order to investigate the contribution of these processes to the H^- production in the RF source, tests have been carried out using additional plates close to the plasma grid in order to enlarge the surface.

Experiments with different plasma grid materials have been continued. Using an in-situ work function probe of the plasma grid surface conditions can be observed. This is of great interest, in particular, during Caesium evaporation.

A number of new diagnostics have been implemented in the test stand: Plasma parameters are studied by laser detachment and by a spectrometer. For the beam diagnostic a calorimeter is used with a new infrared based beam profile measurement.

C-6

POWER MODULATION CAPABILITIES OF THE 140 GHz / 1 MW GYROTRON FOR THE STELLARATOR WENDELSTEIN 7-X

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In present tokamaks, and in particular in future larger devices as ITER, the control of Neo-classical Tearing Modes (NTM) is essential for achieving high performance in terms of the beta limit. A commonly used scheme for NTM stabilization consists in driving a helical current at the resonance surface of interest with Electron-Cyclotron-Current-Drive (ECCD). Depending on the ratio between the magnetic island size and the RF beam width, complete stabilization of the NTM will only be achieved with deep RF power modulation in phase with the mode and, for ITER, recent studies indicate that a modulation depth as high as 70% at frequencies ranging between 1 and 10 kHz would be needed.

In the frame of the European development program of high power sources for ECRH applications between FZK Karlsruhe, IPP Garching/Greifswald, EPFL Lausanne, IPF Stuttgart and Thales Electron Devices, the modulation capabilities of the 140 GHz/1MW gyrotron developed for the Wendelstein 7-X stellarator at Greifswald, Germany, have been experimentally investigated. In RF pulse lengths of 1 s, RF power modulation depths higher than 70% have been obtained either by modulating the main voltage (MVM) or the depressed collector voltage (CVM). Frequency modulation as high as 50 kHz have been obtained with MVM and 1 kHz with CVM, in both cases the limitation was given by the corresponding power supply and not by the gyrotron itself. Detailed analysis of the collector loading with respect to the modulation scheme will be presented and the intrinsic gyrotron limitations for long pulse operation with deep modulation will be discussed.

C-7

DESIGN, TESTS AND RESULTS OF A SECOND HARMONIC FILTER FOR THE ICRH GENERATOR OF JET

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Due to its operation in class B or C, the last stage of an ICRH power amplifier can excite harmonic signals of the operating frequency one in its load constituted by the transmission line and the antenna system, depending on the response of this load to the harmonic signals. These signals can, by interference with the fundamental frequency, produce locally higher voltage and limit therefore the power capability of the ICRH system. On JET, the ratio of the amplitude of the harmonic signals to the fundamental has been measured and can reach up to 10%. A filter designed for the ICRH system of JET will reduce significantly these harmonic signals. Removing the effect of these 10% of harmonics could increase the maximum level of power by 20%!

A resistive harmonic filter for JET has been developed and already partly successfully tested at TEXTOR (50Ω characteristic impedance, 14 kVp at 2 MW). This filter is operating on the JET transmission line of 30Ω characteristic impedance and in a frequency range extending from 23 MHz to 57 MHz (*). It has to produce negligible reflection at the fundamental frequency (VSWR < 1.1 on a matched load) and has to absorb more than 64% of the second harmonic signal (46 to 114 MHz) with total reflection at any phase.

The filter has been tested at low power in the full operating frequency range showing that its electrical characteristics correspond to the modelling and fulfil the requested specifications. Tests at high power have already been successfully done on TEXTOR at 2 MW level for pulses exceeding 10 s and will be repeated on JET in April. The twin filter sections will be mounted on the output of the JET generator (A1 and A2). The paper will also present the results on plasma.

(*cfr. F.Durodié et al., Second harmonic protection filter circuits for the JET ICRF plant, proceedings of 21th SOFT, Madrid, 2000, p. 557)

C-8

DEVELOPMENT OF A LOAD-INSENSITIVE ICRH ANTENNA SYSTEM ON TEXTOR

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Due to the rearrangement of the diagnostic positions resulting from the DED installation on TEXTOR a new antenna system has been designed to be compatible with the inlet of the diagnostic beam between its two radiating strips. This antenna has been designed to be able to test the "conjugated T "mode of operation that is foreseen for the new JET-EP antenna and which is characterised by its insensitivity to the variations of the antenna loading resistance. This could help to solve the problem of generator tripping occurring in the ICRH heating of Elmy H-mode plasmas. The antenna is constituted by a pair of radiating strip conductors, which are grounded at one side and connected at the other side to a vacuum tuneable capacitor, which is also grounded. A tap at an appropriate position feeds each radiating strip conductor. The two feeding lines are connected to the same generator by a T section. The "conjugated T " mode of operation is obtained by appropriately de-tuning the strip conductors by means of the vacuum capacitors and of quarter-wavelength transformer on the feeding lines. Further optimisation is obtained by using a variable phase shifter in each feeding line. There is no more need of the stub tuner and the low remaining VSWR at the transmitter can be compensated with the real time automatic tuning system. With this optimisation a VSWR at the generator lower than 1.1 can be maintained for an antenna lineic loading resistance varying between 2.5 and 9.5 ohm/m (or < 1.5 between 1.6 and 16 ohm/m).

The price to pay for this large resilience to loading variations is a dependence of the phase difference between the strip conductors on the loading and some unbalance between the radiated power by the two strip conductors.

C-9

THE ECW INSTALLATION AT THE TEXTOR TOKAMAK

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Previous Electron Cyclotron Wave experiments on TEXTOR were performed using a 110 GHz, 350 kW, 200 ms gyrotron. The installation is being extended with a new gyrotron at 140 GHz. The gyrotron has been designed to produce more than 800 kW of RF power in a Gausian beam for more than 3 seconds at 1 % duty cycle. This performance has already been demonstrated. The output power can be modulated between 20% to 100% at a maximum frequency of 10 kHz.

In order to accommodate the new gyrotron a major upgrade of the ECW installation was required. It should be possible to operate both the existing 110 GHz gyrotron and the 140 GHz gyrotron although not simultaneously. To achieve this the High Voltage Power Supply and Modulator/Regulator were modified and a new water cooling circuit was built. The control system was redesigned such that it allows safe manual commissioning of gyrotrons and subsystems while at the same time the operator can instantly change to fully automated operation by means of a PLC from the TEXTOR Control Room. Some additional protection features were also required.

The launcher is steerable in 2 directions and designed to operate in three different modes i) deposition of power at a fixed location in the plasma, ii) swept operation, and iii) feedback of the launcher system depending on plasma parameters. The launcher motors are controlled by the same PLC as used for the automated control but can be triggered by the central timing system to allow the fast response times required in the last two modes of operation. For the long pulses protection against deposition on the surface of the mirrors had to be included and materials changed.

First experimental results of the new installation are expected in September 2002, shortly after the installation of the Dynamic Ergodic Divertor.

C-10

DESIGN AND CONSTRUCTION OF THE JET-EP ICRF HIGH POWER PROTOTYPE ANTENNA

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The JET – Enhanced Performance (JET- EP) antenna is being designed to couple ion cyclotron range of frequencies (ICRF) power at high density (8 MW / m^2, 7 MW total) into an ELMy H-mode plasma with a characteristic resistive loading > 2 Ω / m. Features enabling these goals to be met include use of multiple, electrically short, low-inductance current straps in a 4 poloidal by 2 toroidal array, and a "conjugate-tee" matching circuit that maintains a nearly constant input impedance with large increases in plasma resistive loading. The design also utilizes internal matching capacitors to reduce the maximum voltage and limit the region in which high voltages and currents are present. The capacitors can be replaced in the event of failure without the need for access from inside the torus. A frequency range of 30–55 MHz is achieved using an adjustable matching transformer incorporating a novel 9.4 Ω characteristic impedance section in the vacuum transmission line (VTL).

A collaboration between Oak Ridge National Laboratory (ORNL), Princeton Plasma Physics Laboratory (PPPL), and the European Fusion Development Agreement - Joint European Torus (EFDA-JET) is now fabricating a prototype of this antenna. The two-strap High Power Prototype (HPP) closely approximates in geometry and materials the left upper quadrant of the actual antenna. High power tests will be performed in vacuum with the HPP installed in the Radio Frequency Test Facility (RFTF) at ORNL. These include long pulse (20 s) tests at the design current (920 A rms) and voltage (35 kV peak) limits of the internal matching capacitors, with shorter pulses (10 s) at currents up to 1250 A rms. A variety of diagnostics will be used, including arrays of current and voltage probes, directional couplers, an infrared camera imaging the front of the antenna, infrared thermocouples or fluorescent temperature monitors to measure the capacitor braze joint temperatures, and vacuum gauges to measure pressures at various locations in the VTL.

The mechanical and load-tolerant electrical design of the HPP will be discussed and compared to that of the actual antenna. Further details on the diagnostics and the test plan will also be given. Finally, the status of the fabrication and assembly effort will be reviewed.

C-11

DEVELOPMENT OF ICRF COMPONENTS FOR KSTAR

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The ICRF system for the KSTAR tokamak [1, 2] is being developed to support long-pulse, high-β, advanced tokamak fusion physics experiments. With the frequency range of 25 to 60 MHz, the ICRF system provides heating and centrally peaked/off-axis current drive for various operating scenarios over a range of magnetic fields. The ICRF system will deliver 12 MW of rf power to the plasma for 300 second through two antennas located in adjacent ports. The maximum voltage in the system is set to be less than 35 kV.

For high power, long pulse operation, relevant ICRF technologies have been developed in the area of the antenna, the vacuum feedthrough and the matching devices. A high power density (~ 10 MW/m^2) prototype ICRF antenna has been developed. For 300 sec operation, the antenna has many cooling channels inside the current strap, Faraday shield, cavity wall and vacuum transmission line to remove the dissipated rf power loss and incoming plasma heat loads. Mechanical and high voltage tests were performed with the antenna installed in the rf test chamber. During the rf pulse, the peak voltage, forward/reflected powers, temperature on the antenna, and gas pressure are measured. The peak voltages of 38 kV for 10 second and 33 kV for 60 second (without cooling) were found at 30 MHz, 30 kW of rf power. Higher power and lomger pulse tests are underway. A vacuum feedthrough of 1 MW rf power was developed, which has two alumina (Al_2O_3, 97%) ceramic cylinders and O-ring seal instead of a brazed seal for good mechanical and thermal strength which is important in long pulse or steady state operation. For cooling of the ceramics, dry air is injected into the ceramic surface through two outer nozzles. Independent cooling water channels are installed to cool the inner conductor of the feedthrough. Rf high voltage test shows that stable operation is possible with the peak voltage of 30 kV for 300 second without any severe damage. Matching devices, liquid stub tuner and liquid phase shifter, were fabricated and tested for long pulse, high power operation. The performance of transmission line components show that they have a high stand-off voltage (>40 kV) and they are reliable rf components for high power transmission.

The results of the engineering development will make ICRF system a key component for the long pulse, advanced tokamak operation of the KSTAR tokamak.

G. S. Lee et al. The design of the KSTAR tokamak Fusion Engineering and Design **46** (1999) 405–411.
G. S. Lee et al. "The KSTAR Project: Advanced Steady-State Superconducting Tokamak Experiment", Nuclear Fusion **40** (2000) 575–582.

C-12

RECENT DEVELOPMENTS ON THE 110 GHz ELECTRON CYCLOTRON HEATING INSTALLATION ON THE DIII–D TOKAMAK*

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The 110 GHz electron cyclotron system on the DIII–D tokamak is receiving significant improvements to its capabilities. Chief among these is the addition of the fifth and sixth 1 MW class gyrotrons, increasing the power available for auxiliary heating and current drive by nearly 60%. A new fully articulating dual launcher has been installed for high speed scans. A new feedback system linking the DIII–D plasma control system (PCS) with the ECH cathode voltage waveform generators permits real-time control of plasma properties such as electron temperature.

The commissioning of the fifth and sixth gyrotrons will complete the upgrade to 6 gyrotrons begun four years ago. Three of these tubes are from Communications and Power Industries (CPI). They use CVD diamond rf output windows to obtain high power with long pulse capability. The rf beams from these tubes are nearly Gaussian, and require fewer correction optics to couple to the $HE_{1,1}$ mode in the circular waveguide.

The new dual launcher allows for two axis independent steering for each waveguide. The mirrors can currently be moved at up to 100 degrees per second. The system is being further developed to allow for scanning during shots.

The feedback system linking the DIII–D PCS with the ECH cathode voltage waveform generators takes advantage of the external amplitude modulation input on the generators. The signal from the PCS is conditioned so that it will modulate the ECH preprogrammed waveform within acceptable gyrotron operating limits. The PCS can use a variety of plasma monitors to generate its signal, such as electron cyclotron emission or Mirnov probes. The feedback system has been demonstrated to control the core electron temperature development during the early phase of a plasma discharge.

The new capabilities should further enhance the role of ECH in tokamak plasma control and lead to a new set of experiments.

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C-13

OVERVIEW OF THE ITER-FEAT LH SYSTEM

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LH is considered to be used in ITER FEAT steady-state scenario in order to drive the current in the plasma outer part, region where its current drive efficiency is the highest compared with other CD system. It can also be used to stabilise NTM mode and to control the current profile.

For this purpose, the design of a LH system able to inject 20 MW at 5 GHz CW, radiated at a N// peak value of 2, has been done. It is composed of:

- a transmitter with 24, 1 MW CW, sources, fed in group of 4 by HV DC power supply.
- a transmission line, 60 m long. Its main components are: a recombiner network used to combine the power of 4 tubes and to transform the propagating mode from TE10 rectangular mode to TE01 circular mode, an oversized circular straight part, bends and mode filters.
- a launcher based on Passive Active Multijunction (PAM) concept. It is made of 2 main parts: the Cryostat Part (CP) and the Vacuum Part (VP). On the flange of the CP there are 24 RF windows. Each of them is linked to one 3 dB hybrid coupler which then feeds the VP through two RF windows installed on the vacuum flange. Each RF window feeds one TE10 to TE30 mode converter connected to a 4.77 dB, H plane coupler, followed by multijunction made of 8 active waveguides. The VP is inserted in a casing similar to other RF system

The following preliminary studies have been done:

- the coupling and the power directivity of the launcher have been analysed with SWAN code as a function of the electron density,
- acceleration of electrons in the antenna near field has been checked,
- thermal analysis have been done with Castem 2000 code taking into account the neutron flux, the plasma radiated flux and the RF losses,
- mechanical stress due to disruption has been analysed,
- the level of activation in the different part of the antenna has been estimated,
- RF computations have been performed [1].

After a general overview of the whole LH system, the results of the previous studies will be given. The general conclusion is that a reliable LH system in the ITER FEAT environment is totally feasible.

[1] The main microwave components of the LHH&CD system for ITER.
F. Mirizzi, Ph. Bibet, S. Kuzikov#, this Conference.

C-14

VERY LONG PULSE OPERATION OF THE TORE SUPRA ECRH SYSTEM

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An ECRH (Electron Cyclotron Resonance Heating) system is under development at CEA (Commissariat à l'Energie Atomique) Cadarache for experiments on the Tore Supra Tokamak; it is presently capable of coupling 800 kW for 5 s to the plasma and will be upgraded by increasing the system performance to 3 MW for 5 s and to 2.4 MW for up to 600 s.

The main components of the generator (see [1] for the description of the whole system) are six gyrotrons TH 1506B developed in collaboration between European research laboratories and Thales Electron Devices (TED). The main features of this gyrotron are: frequency 118 GHz, output power 400 kW for pulse length up to 600 s and 500 kW for 5 s. The short pulse specification forms part of the factory acceptance test, while the long pulse capability is only tested at CEA Cadarache after delivery. Two gyrotrons have already been manufactured and tested on dummy loads and on plasma, individually and together. The first series tube has achieved a 110 s pulse on a water load, but with a limited output power of 300 kW. This limitation has been observed through a strong degassing within the tube due to the overheating of the internal mirror box. The overheating was shown to be due to spurious oscillations within the injector of the gyrotron. New studies to prevent these oscillations and to improve the performance of the tube, by modifying the shape of the injector and the cooling system of the mirror tank, are in progress at TED. This work will lead to the manufacture of a new gyrotron by spring 2003. The two existing gyrotrons were connected to Tore Supra for a short experimental campaign at the end of 2001 and further plasma experiments will be performed in 2002. The two gyrotrons were used independently and together for pulse lengths up to 2 s, injecting up to 800 kW under various toroidal and poloidal angles, various polarizations of the wave and for the first time, with a modulation frequency of the RF signal up to 25 Hz to allow heat transport and deposition profiles analysis.

The different results both on dummy loads and on plasma are described in the paper, as well as some problems encountered mainly during the tests on plasma, on the regulation of the beam current. This regulation will be improved with a new heating power supply. In the mean time a conditioning procedure of the gyrotrons carried out on the Tokamak, without plasma, will allow a better setting of the present supply parameters for the following plasma shot.

[1] The 118 GHz ECRH experiment on Tore Supra, C. Darbos et al. Fusion Engineering and Design 56-57 (2001), pp 605–609.

C-15

TORE SUPRA ICRH ANTENNA PROTOTYPE FOR NEXT STEP DEVICES

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The present Tore Supra (TS) ICRH antennas are recognised to couple efficiently the power into the plasma. Routinely, a power level of 3.6 and 3.8MW is launched into the plasma during about tens of seconds. Even, the full nominal power supplied by the amplifier, 4MW per antenna, has been coupled to the plasma corresponding to a power density record of 16 MW/m2. At this level, the antenna mouth is positioned 2cm behind the plasma boundary and the plasma loading has been found between 2 and 6Ω/m. Using the principle of the Resonant Double Loop (RDL) antenna, the RF circuit consists of a cavity, a Faraday screen, a current strap and 2 adjustable capacitors. This circuit has localised high RF voltage and high RF current in a small volume thus reducing the power losses to a minimum, compared with the others antenna RF circuits.

This technology tested in Tore Supra, for 10 years, can easily be applicable to the next step devices working at similar plasma Scrape Off Layer (SOL) parameters. Unfortunately, in the next step devices, such as ITER-FEAT, in some mode of operation the SOL parameters may significantly and rapidly vary. Besides, the distance between the antenna mouth and the plasma boundary will be larger (near 12 cm) compared to the TS configuration.

An electrical model of the present TS antenna circuit shows a limitation in case of a high and fast variation of the plasma load resistance. For that, a new arrangement in the short-circuit position and in the vacuum transmission line connection of the RDL circuit has been studied leading to a RF circuit with a significant insensitivity to variation of plasma load. Based on this new circuit version, a prototype antenna is being designed. In order to save money and gain time, several modifications on one of the present antennas will be carried out to build this prototype antenna. Therefore, its installation inside TS has been planned for the 2003 campaign.

After a brief comparison between the new circuit version and the conventional one, the paper will focus at the same time on the design and manufacturing description of this ICRH antenna prototype, the principle of which being suitable for the next step devices.

C-16

PLANS FOR A NEW ECRH SYSTEM AT ASDEX UPGRADE

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Construction of a new ECRH system for ASDEX Upgrade with a power of 4 MW and a pulse duration of 10 sec with linearly polarized Gaussian output has started last year. Four gyrotrons with single stage depressed collector will generate the power. A particular feature will be the possibility to operate them at different frequencies. The first gyrotron can work at 105 GHz and at 140 GHz, making use of the resonances of the diamond vacuum window at these frequencies. A second step-tuneable gyrotron is designed to work at several frequencies within the same interval. A diamond output window mounted at the Brewster angle allows broadband transmission. Two gyrotrons will be fed from one thyristor controlled power supply 70 kV / 80 A. However, each gyrotron will have its own series tetrode modulator with an output of 60 kV / 40 A. The body will be driven from a switching power supply 45 kV / 0.3 A with 5 kHz modulation capability. The whole system is planned to be operated remotely.

The transmission line, similar to the one which is used now in the first ASDEX Upgrade ECRH system, will have a quasioptical section next to the gyrotrons and a 70 m long corrugated waveguide line with 87 mm i.d. at normal air pressure. The corrugation profile was optimised for the broadband transmission. The quasioptical section contains phase correcting mirrors, a pair of broadband polarisation mirrors and a switchable mirror to direct the beam into a dummy load or a calorimeter load. The dummy load is located central in the system with a rotatable mirror which can be directed towards each one of the 4 gyrotrons. At the torus side of the transmission line we will install short pulse loads (≈ 0.1 sec) which allows testing of the transmission line every day before the experiments start.

For the torus window we will also use a diamond window resonant at the two frequencies. However, for the step-tuneable transmission line we think of using a tuneable double disk diamond window, instead of a Brewster angle window, to allow transmission of an arbitrary polarisation without the need for polarising mirrors inside the torus.

The launchers will allow to scan the whole poloidal crossection at toroidal angles from -25^o to $+25^o$ at moderate speed between the pulse. Over a limited range the poloidal scan is planned to be fast (10^o in 100 msec) with the intention of a feedback controlled power deposition. The launching mirror will be made of graphite with a conductive copper coating.

C-17

HIGH FREQUENCY/HIGH VOLTAGE SOLID STATE BODY POWER SUPPLIES FOR CPD GYROTRONS

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The ITER EC Heating and Current Drive scenarios demand the modulation of the RF output power. For the ITER Gyrotrons, of Collector Potential Depressed (CPD) type, this modulation require the accurate control of the acceleration voltage between body and cathode.

These body power supplies or acceleration power supplies, APS, have been traditionally based on vacuum tube amplifiers. In view of the continuing developments in power electronic devices, there is a wide spread interest in developing new systems based on solid state devices. APS based on solid state technology are in fact expected to be less expensive and with lower maintenance and service requirements.

As a part of a development programme possibly leading on the short-time at the manufacture and testing of a prototype, the paper reports on the conceptual study and design of solid state high voltage power supplies with the required dynamic and control characteristics. Different possible solutions and topology are analysed and compared to identify the most convenient scheme both under the technical and economical viewpoints. In particular switched mode amplifiers and linear amplifiers are considered.

The target specifications for the APS are:

- body to cathode (acceleration) voltage:	up to 90 kV
- DC Accuracy	± 0.5%
- frequency modulation range:	DC to 5 kHz
- THD (sinusoidal reference at 5 kHz)	5%
- phase shift (at 5 kHz)	typically 10 degrees
- switch off time:	< 10 μs
- max energy deposited in case of arc:	< 10 J

The load is essentially capacitive with a value of up to a few nanoFarads.

The most promising systems are described in more detail, including analyses in normal and fault conditions. Future activities on the field are also outlined.

C-18

MATCHED CALORIMETRIC LOADS FOR HIGH POWER MILLIMETER-WAVE GYROTRONS

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High power gyrotron testing is a necessary step in the development of mm-wave systems for fusion research. For this purpose a compact matched load is being developed, with low reflection and fast calorimetric measurement of the millimeter-wave power as the main goals. The results of the tests (at 0.5 Mw and 0.5s pulse length) of loads installed on the ECRH plant of the FTU Tokamak in Frascati were confirmed at higher power and longer pulse on the AUG ECRH plant. New tests and improvements on material, geometry and cooling aim at a higher power capability and at longer pulse duration.

The load tested so far, made essentially of a sphere in copper with the internal surface covered by a plasma sprayed mixture of ceramics (Al_2O_3,TiO_2), provides a low reflectivity, in the order of 5%, exploiting the multiple reflections of the radiation on the inner, partially reflective walls. When used repeatedly at full gyrotron pulse, it shows signs of localised damage of the absorbing layer, with damage patterns at different scales. Patterns of damage, recognised as interference of multiple reflections of radiation inside the load, and accumulation close to the entrance port, are both explained running a detailed numerical model. The occurring degradation of the absorbing material is clearly due to the repeated exposures to the high energy flux. Samples of degraded and non-degraded coating have been analysed with various techniques: ESEM (Scanning Electron Microscopy), micro-analysis with X-ray EDS, Differential Scanning Calorimeter (DSC), X-ray diffraction analysis (XRD).

Some techniques indicate that a phase transition is occurring in Al_2O_3, involving changes of the substrate's physical characteristics. Moreover, optical microscopy has revealed surface damage due to electric arcs, which may start a cascade process resulting in a growth of the damaged area. Fast measurements of absorber temperature during and after the pulse, performed with an infrared detector directed into the load, allow estimates of the peak temperature of the coating in the real working conditions.

Effects of repeated power deposition also on the TiO_2 component are discussed, and new candidate materials have been tested for reflectivity at 140 GHz.

Any improvement of the present design in power and energy deposition capability follows from the comprehension of the degrading phenomena and on appropriate steps to mitigate the effects. The use of different plasma-sprayed materials as mm-wave absorbers, a more homogeneous deposition and an improved cooling geometry are considered, to reach the goal of the 1MW-CW power capability.

C-19

STATUS OF THE 1MW, 140 GHz, CW GYROTRON FOR WENDELSTEIN 7-X

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In the framework of a collaboration between FZK-Karlsruhe, CRPP-Lausanne and TED-Vélizy, with contributions from IPF-Stuttgart, CEA-Cadarache and IPP Garching/Greifswald, a 1 MW, 140 GHz, CW gyrotron for the 10 MW ECRH system of the new stellarator experiment Wendelstein 7-X at IPP Greifswald (Germany) is under development.

The tube operates in the $TE_{28,8}$ cavity mode and provides a linearly polarised, fundamental Gaussian output beam. It is composed of a diode-type electron gun, an improved beam tunnel, a high-mode-purity low-ohmic-loss cavity with rounded transitions, an optimised non-linear uptaper, a highly efficient internal quasi-optical mode converter employing an improved launcher together with one quasi-elliptical and two beam shaping reflectors, a large single-stage depressed collector with beam shaping and sweeping magnets, and a horizontal RF output through a large-aperture, water edge-cooled, single-disk CVD-diamond window.

A first tube has been tested at FZK. In short pulse operation and at a beam current of 40 A, an output power of 1.15 MW was achieved with an accelerating voltage of 84 kV and a depression voltage of 25 kV (49% efficiency). RF mode purity measurements performed using an IR camera and a thin dielectric target located at different positions across the RF beam showed a very good agreement with the theoretical predictions. After long pulse conditioning of the gyrotron and auxiliaries, the following record performances were finally achieved: 1 MW – 10s (efficiency of 50 %), 0.9 MW – 45 s (efficiency of 39 %), 0.74 MW – 100 s (efficiency of 32%), 0.64 MW – 140 s and 0.47 MW – 180 s. Power modulation experiments performed on a few seconds pulses with modulation frequencies up to 50 kHz showed the gyrotron capability to operate from 0.1 to 1 MW.

The tube has then been sent back to TED for completion of an detailed expertise prior to the fabrication of an improved prototype which will have been tested by end of August 2002.

C-20

THE ITER-LIKE ICRF LAUNCHER PROJECT FOR JET

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One of the outstanding challenges to qualify ICRF as a mature plasma heating system for ITER is a convincing demonstration of achieving the required power density in relevant antenna coupling conditions: a low antenna to plasma coupling together with fast antenna impedance transients due to ELMs.

The aim of the ITER-like ICRF antenna project for JET is to validate a design able to couple 8 MW/m^2 to an ELMy plasma across a frequency band of 30 to 55 MHz while at the same time testing the conjugate-T matching concept as a so-called passive load tolerant matching technique. This requires a minimum antenna coupling between 2 to 4 Ohm/m.

The proposed launcher integrates four matched loops each consisting of two short antenna straps with low inductance. Each of these pairs of straps are fed from a 2MW JET amplifier through capacitors implementing the conjugate-T matching circuit. The overall radiating surface of the whole launcher is about 0.9 m^2 and aims at achieving 7.2 MW, taking into account about 10% losses in the transmission system.

The design is nearing completion in April 2002 and will be tendered for shortly there after. It is expected that the launcher will be available for installation on the JET torus by mid 2004 permitting an experimental campaign beginning at the start of 2005.

US teams of ORNL and PPPL are preparing a High Power Prototype (HPP) of one of the four loops. The results from the HPP experiments expected by October 2002 will permit to carry out small corrections to the design prior to the start of the manufacturing of the key components making up the launcher.

The paper will highlight and explain the main design features of this new ICRF system for JET.

C-21

RADIOFREQUENCY MATCHING STUDIES FOR THE JET ITER-LIKE ICRF SYSTEM

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Beside the unique characteristics of the launcher [1], the transmisssion and matching system of the JET ITER-like ICRF antenna includes a number of specific design features, to be tested on the JET tokamak for the first time:

1. The "Conjugate T" circuit [2], providing internal matching of the launcher over its operating frequency range (30 to 55MHz) by feeding pairs of radiating straps in parallel through adjustable capacitors.
2. A very low reference characteristic impedance, as low as Z0 = 3Ω, for matching the antenna input.
3. An original 30 to 55MHz impedance transformer between these 3Ω and the 30Ω of the JET ICRF transmission lines. Its design, strongly constrained by the practical layout of the system, includes a $\lambda_0/4$ low impedance (9.4Ω in-vessel vacuum transmission line, a $\lambda_0/2$ 30Ω line section including the vacuum window, and a fixed low impedance (12Ω) $\lambda_0/4$ stub which combines RF and ancillary functions. (λ_0=1.765m is the mid-band wavelength.) Perfect adaptation of the 3Ω reference is obtained at midband and band edges. A final adjustable stage directly follows, which consists of a standard line stretcher and a stub, allowing perfect adaptation at other frequencies and compensation of constructional inaccuracies in the fixed elements.

These three ingredients contribute to the exceptional tolerance expected from the system to the large antenna load increases induced by Edge Localized Modes (ELMs).

The paper will discuss the design choices leading to the final matching layout, and present its simulated performance: maximum excursions of power reflection at the generators during typical ELM events (which include both active and reactive load variations), peak voltages and currents along the system, sensitivity to matching and component errors, and proposed automatic capacitor matching algorithm.

[1] F. Durodié et al., this Conference.
[2] G. Bosia, High power density Ion Cyclotron Antennas for application to ITER. To be published in Fusion Science and Technology.

C-22

DESIGN STUDY OF A TEST STAND FOR ITER GYROTRON

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In the frame of the development of the ITER electron cyclotron wave (ECW) system, the development of a 170 GHz, 2 MW CW coaxial gyrotron with depressed collector is foreseen in the European Technology Workplan, during the 6th Framework Program (2003–2006). Such development relies on the availability of a test stand capable of delivering the electrical energy and cooling capacity. This test stand will certainly be used, in a later stage, for the component test of the ITER ECW system.

The gyrotron main electrical specifications are:

Nominal RF Power	2 MW
Nominal pulse length:	CW
Depressed collector:	Yes
RF output efficiency:	≥45% with depressed collector
Main power supply (Collector to cathode): **Voltage** **Current**	 < 60kV/ 90A
Body power supply: **Voltage** **Current**	 30 kV 0.2A(1)

[1]This value refers to the steady state, the peak value will directly depend on the stray capacitance of the connections and of the gyrotron itself.

The design of the electrical system fulfilling the gyrotron requirements will be presented. As compared to the ITER reference design, the one of the test stand will put emphasis on the requirement of flexibility, which is necessary during the development of the gyrotron. Additional equipments, such as the body power supply or the series solid-state switch, will also be presented.

The cooling system will be an important part of the design study. Indeed, as mentioned before the approximate efficiency of such depressed collector gyrotron is about 50%. This implies that more than 4 MW power will be continuously dissipated and evacuated by the cooling equipment. The specifications of the cooling system must also comply with ITER reference design values.

Finally, considerations on the use of such test facility for the commissioning of the ITER launcher prototype will be developed.

C-23

A NEW BOX SCRAPER FOR THE JET FACILITY'S UPGRADED NEUTRAL BEAM INJECTION SYSTEM

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The downstream exit aperture of the JET Neutral Beam Injector Box defines the edge of the profile of each beam as it enters the drift-duct region. This aperture is fitted with a *Box Scraper* assembly comprising high heat-flux hypervapotron elements mounted in a picture-frame arrangement in order to intercept the peripheral part of the beam profile, representing up to 20% of the transmitted power. In order to cope with the enhanced power resulting from upgrading the Neutral Beam Injection Systems at the JET Facility, in which the deuterium neutral beam power from a single box has been increased from 7.5MW to 15MW, it has been necessary to re-design the Box Scraper assembly completely. The design has utilised the original JET hypervapotron concept for the heat flux components, but the internal geometry has been further optimised in order to achieve an increase of peak power density to 13 MWm^{-2} at the Hypervapotron element's surface. Comparisons between the old and new designs with regard to geometry, front surface operating temperature and cooling water velocity are presented. Design constraints of the Box Scraper assembly due to the limitations imposed by the available space envelope within the Injector system are discussed. The adopted design has an optimised arrangement of a pair of hypervapotron elements side by side inclined at different angles of attack to match the gradient of the power density at the beam edge, giving a design figure of 400kW power handling per beam. This satisfies the further power increase. The paper also describes the first operational experience with the new Box Scraper on the JET beamlines and results of the power handling tests carried out on the new Box Scraper prototype assembly at JET Neutral Beam Test Bed operated with new high power 130kV/60A Positive Ion Neutral Injectors. Finally, the problems of installing such an assembly into a tritiated neutral beam system and the methodology adopted are reviewed.

This work has been conducted under the European Fusion Development Agreement and is partly funded by Euratom and the UK Department of Trade and Industry.

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C-24

DESIGN OF AN ECRH LAUNCHER FOR JET

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An ECRH (electron-cyclotron resonance heating) system has been designed for JET in the frame-work of the JET-Enhanced Performance project (JET-EP) under the European Fusion Development Agreement (EFDA). Due to financial constraints it has recently been decided not to implement this project. Nevertheless the design work conducted from April 2000 January 2002 shows a number of features which can be relevant in preparation of future ECRH systems such as the ITER one. The purpose of this abstract is to present an overview of the development and design status of the ECRH launcher. The launcher system enables eight mm-wave beams to be directed from the feeding waveguides through the main horizontal port to the plasma. Water-cooled steerable mirrors will provide the beam deposition into the plasma. The steerable mirrors are adjustable in both poloidal and toroidal direction, which provides the system the required launching range. The paper will contain detailed information on technical solutions developed for the -ECRH launcher. The paper also includes subjects as launcher installation, tritium handling and the launcher test bed. The design status is a description of the situation early 2002, the moment that the project was cancelled. To develop a reliable launcher system, which has to operate in the extreme JET in-vessel environment, tests were required. In close cooperation with the JET drawing office and the ICRH project team, plans were made to modify the ICRH test bed in order to make it applicable for ECRH launcher tests. The test plan was mainly focused on life-cycle tests and low power mm-wave tests. A shock test was planned to simulate vessel acceleration during disruptions. Technical formulas and specific design information concerning materials were used in spreadsheets to establish the best geometry and material for several parts of the launcher steering mechanism. The assembly was made in Catia 4.22 design software. A test was done on the flexible, double bellows, stainless steel water hose, which is meant to provide watercooling on the steerable mirrors. This first test was meant to determine the behaviour of the hose under a pressure of 20 bar. The test and the results will be presented in the paper.

C-25

ICRF HEATING FOR WENDELSTEIN 7-X

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The stellarator Wendelstein 7-X which is being built by the Max-Planck-Institut für Plasmaphysik in Greifswald will be equipped with ECRH, NBI and ICRF heating. ECRH is planned for 10 MW continuous wave (cw) operation and will be the principal heating method at the standard magnetic fields of 1.25T and 2.5T. In a first stage, NBI heating will operate at a power level of 5 MW for 15 sec pulses thus facilitating high beta experiments and ICRF heating is planned to operate at a power level of 4 MW cw facilitating both long pulse experiments at magnetic fields other than required for ECRH operation and high beta experiments.

The ICRF heating system is designed for a frequency range of 25 to 75 MHz making different heating scenarios possible for a wide range of magnetic fields and varying hydrogen concentrations, e.g. He3 and H minority heating, mode conversion heating of H/D mixtures and second harmonic H heating. For magnetic field values between 0.8 T and 3 T and hydrogen concentration between 0 and 100% slab geometry full wave calculations predict single pass absorption values above 0.7 of at least of above heating scenarios. First calculations with a three-dimensional full wave ICRF heating stellarator code developed by V. Vdovin for the Max-Planck Institute also predict good absorption for the different heating schemes on Wendelstein 7-X.

In a first stage two antennas will be arranged at the low field side near the symmetry plane with bean shaped plasma and tokamak like magnetic field distribution. The fully water cooled two-strap antennas will operate at the same temperature (up to 150°C) as the wall protection (liner) and will be radially movable to adjust to different plasma shapes. The antennas are of the conventional push-pull type arranged in a resonant transmission line loop. The antennas are planned to be powered by two generators with an output of 2 MW cw each which cannot be reliably provided by the present tube and generator technology. Therefore the development of a common European generator type for these parameters has been initiated by the ITER, Tore Supra and W7-X teams meeting the requirements of all three experiments.

The matching of the antenna impedance to the generator impedance will be done by a feedback controlled fast ferrite transmission line system. Thus antenna load variations on time scales slower than 0.1 msec can be compensated. The generators will be passively isolated from faster antenna load variations by a system of 3dB couplers in a setup similar to the one used on ASDEX Upgrade. Different frequencies are possible by the use of transmission line stretchers and can be changed within several minutes.

CW operation requires water cooling for almost all components of the system, calling also for further development and for test facilities for coaxial line components, e.g. shiftable contacts, switches, ceramic spacers, feedthroughs etc. Therefore a coaxial test resonator is being built allowing high voltage and high current cw tests in Garching in spite of the only 50 kW cw power available from the ASDEX Upgrade generators.

C-26

ELECTROMECHANICAL ANALYSES OF THE JET ICRH ITER-LIKE ANTENNA

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A high power and high power density Ion Cyclotron Resonance Frequency (ICRF) antenna is being designed for the Joint European Torus (JET). The aim is to couple 8 MW of RF power (30–57 MHz) to a plasma for 10 s and reaching a power density of 8 MW/m^2 radiated power at the level of the antenna. This experiment is considered to be part of a demonstration for the feasibility of heating plasmas with ICRF in the next step fusion device such as the International Tokamak Experimental Reactor (ITER).

3D finite element models of the antenna and neighbouring components (poloidar limiters, the main horizontal port and vessel double wall) have been developed.

The flow pattern of halo currents and the distribution of induced currents during disruptions have been determined with electromagnetic static and transient analyses, imposing loads (rate and amplitude of the poloidal field change and halo current sources) consistent with those observed in the last years of operation of JET. These currents have subsequently been used to estimate the electomechanical loads (Lorentz forces) due to the interaction with the externally imposed magnetic field. Stress, displacements, forces and torques have been calculated in all parts of the antenna subassembly. Usually the same mesh has been used for both the electromagnetic and the structural analysis.

The results of the analyses validated the selection of resistances between the antenna housing and the in-vessel components, aiming to avoid high current densities in all the antenna components. The mechanical analyses showed that stresses in all parts of the structure are under the materials yield strength. The computed reactions calculation served as an input to design the mechanical structure supporting the antenna and its components.

This work was performed under the European Fusion Development Agreement.

C-27

CIRCUIT DESIGN AND SIMULATION OF A HV-SUPPLY CONTROLLING THE POWER OF 140 GHz 1MW GYROTRONS FOR ECRH ON W7-X

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ECRH on the Stellarator W7-X under construction needs the generation of 10 MW cw power at 140 GHz millimeter-wave frequency. The generator system consists of ten 1 MW cw gyrotrons. These gyrotrons under development are equipped with a voltage depressed collector for electron beam energy recovery which results in power efficiencies around 50%.

The electron beam of the gyrotron is accelerated between cathode and resonator body by a voltage source of about 80 kV and decelerated by an opposite electrical field between resonator and collector depressed by about 30 kV. The microwave output power depends sensitively on the acceleration voltage. Therefore the quality of high power voltage supply for 50 kV and 40 A beam current has no influence on the stability of the gyrotron output power as long as a highly stable low power source for the 80 kV accelerating voltage is used.

The high power HV-supply for beam current used on W7-X is of the PSM type (THALES) consisting of 84 switchable DC-sources connected in series. Programming of the output voltage is made by switching on and off individual sources at a specific time sequence. The voltage noise generated by this switch mode operation is suppressed by a low-pass filter. The elements of this filter are subjects for stored energy which should be taken in consideration in case of voltage breakdown by arcing in the gyrotron. The specified maximum of released energy in an arc demands for an additional tube protection circuit. This circuit is designed as a fast crowbar by means of a thyratron installed at the HV feed nearby to the gyrotron. Simulation of the whole circuit including the PSM-supply with PSPICE is a tool for optimization of the decoupling elements for the design of the tube protection circuit.

The low power HV-supply for the accelerating voltage of the gyrotron electron beam is realized by a HV servo amplifier in vacuum tube technology. This amplifier is fed by a 50 kV, 400mA power supply in switch mode technology. The output of the amplifier is connected between tube collector at ground potential and the resonator body at around 30 kV potential with respect to ground. The control of the accelerating voltage between cathode and body at a stability level of 0.2% is made by sensing of this voltage difference for the feed back loop of the servo amplifier. The resonator body collects only a very low fraction of the beam electrons such that the output load of the servo amplifier is mainly capacitive. The amplifier is designed to drive a capacitive load of 1 nF at a slewrate of 600 V/µs. This will allow full modulation of gyrotron output power with a frequency of up to 20 kHz. Modelling of the amplifier circuit and simulation of operation are presented.

C-28

THE HIGH VOLTAGE POWER SUPPLY FOR THE ALCATOR C-MOD LOWER HYBRID HEATING SYSTEM

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Alcator C-MOD is a high-field, high-density, diverted, compact tokamak, which, in its present form uses inductive current drive and is heated with 5 MW of ICRF auxiliary power. C-Mod is in the process of being upgraded with a 4.6 GHz Lower Hybrid heating and current drive system. The purpose of the experiment is to develop and explore the potential of "Advanced Tokamak Regimes", i.e., regimes with high bootstrap fraction (~ 70%), high β_n (~3) and high confinement (H_H ~ 1–2) under quasi-steady-state conditions.

The amplifier uses twelve klystrons each rated 250 kW with the possibility to add another four klystrons in a later stage. All klystrons are powered in parallel from a single power supply rated 50 kV / 208 A. The power supply is designed using solid-state switches and pulse step modulator (PSM) technology. The power supply incorporates fast on and off switching capabilities which eliminates the need for an additional modulator or crowbar. Due to space restrictions, the complete power supply is installed outdoors in a container. The required space inside the building is kept to an absolute minimum.

Currently, the power supply is installed and under testing. First high-voltage tests on the klystrons are planned in 2003.

The paper gives an overview of the complete RF system and a detailed description of the power supply. It describes in detail the voltage control and regulation for the system. The results of the power supply commissioning and the tests on the klystrons will also be presented.

C-29

EFFECT OF PRESSURE ON THE CONDUCTIVITY OF NBI INSULATOR GASES UNDER IRRADIATION

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Neutral beam injectors will require the use of some type of gas contained under pressure within an earthed pressure vessel to insulate the high voltage transmission line, ion source, and accelerator tube. This insulating gas will be in a radiation field of the order of 1 Gy/s due to the plasma and the NBI accelerator itself. The radiation will cause ionization in the gas and hence an increase in the electrical conductivity. As this is a source of power loss due to the corresponding leakage current which in addition will produce heating and possibly breakdown, the radiation effect must be quantified and taken into account in the engineering design of the NBI system. Initial experimental results for the radiation induced leakage currents in dry air and SF_6 at atmospheric pressure indicated that large power losses (≈1 MW) would occur in the ITER NBI system. Due to the experimental difficulty in obtaining data for larger volumes and voltages, and in particular for gas pressures > 1 bar, a model was developed which makes predictions for the gas conductivity in terms of electric field, ionizing dose rate, and gas pressure. The model was found to predict the observed experimental data to such a degree as to give confidence in the extrapolated predictions.

In the work to be presented, experiments have been carried out in order to study the effect of pressure on the radiation induced electrical conductivity (RIC) and comparison with the model has been made.The experiments have been carried out in the beam line of a 2 MeV Van de Graaff electron accelerator, with the gases being irradiated either with Bremsstrahlung produced by stopping the electron beam in a gold target, or directly with 1.8 MeV electrons. In this way radiation levels from 0.02 to 2 Gy/s have been covered. The experimental set-up permitted an electric field to be applied to the irradiated volume of gas and the electric current flowing through the ionized gas to be measured. Two gases SF_6 and dry air have been irradiated. During irradiation the gas pressure was changed and the RIC measured for different applied voltages.

The theory predicts a complex behaviour of the RIC with gas pressure. For low voltages the conductivity decreases as the pressure increases due to a decrease of the electron-ion pair lifetime. However for larger voltages the situation reverses. The experimental results follow this theoretical behaviour with pressure very well, allowing extrapolation of experimental data to larger volumes and voltages, and in particular for high gas pressures.

C-30

INLINE MAGNETIC RESIDUAL ION DUMP FOR THE ITER NEUTRAL BEAM SYSTEM

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The presently foreseen residual ion dump for the ITER neutral beam system is based on electrostatic deflection of the ions sideward to inline dump plates. These plates are alternatively biased with about 25 kV and form four narrow channels (1800 mm long and 90 mm wide at the entrance), corresponding to the grounded grid geometry. This concept has the advantage of a simple and compact design and it generates no additional magnetic stray field. However, such a concept was never be used in any working neutral beam system. All systems use a magnetic deflection system with remote ion dumps in order to avoid creating secondary electrons and beam blocking due a possible high pressure in the dump channels. Furthermore, lifetime considerations[1] showed that power sweeping is necessary in order to reduce the power load to 6 MW/m^2.

In a previous paper[2], we presented a design for a magnetic removal system with remote ion dumps. The ions are deflected by a horizontal magnetic field up- and downwards. The design of the ion dumps was very carefully adapted to the grid geometry. Due to the limited space and the geometric restrictions of the ITER beamline design, only 90^o deflection was possible – the preferred 180^o deflection requires beamline dimensions at least three times larger than the beam dimensions; hence the power distribution onto the target plates was dominated by focussing effects. Furthermore, these effects depend very critically on beam alignment, divergence and steering. In the present design upgrade, the vertical steering angle of the ITER source is variable in order to change the power deposition in the plasma. Hence, vertical deflection is not possible. The same geometric restrictions and focussing effects are valid for a sideward deflection to remote ion dumps.

Our final concept, which is presented in this paper, consists of still sideward magnetic deflection, but to inline vertical dump plates, as in the electrostatic case. The two plates are roughly 1500 mm high and 2200 mm long and presents a 500 mm wide opening to the beam. The ions hit the target plates only from one side with oblique incidence ($<14^o$). First calculations show, that the power load can be kept below 10 MW/m^2 for a 5 mrad divergent beam; in the case of 3 mrad divergence, there are a few hot spots up to 20 MW/m^2. The target plates can be fed with cooling water from the backside (in contrast to the 'electrostatic' plates, where the feeding has to be done from top), increasing the maximum acceptable power load. Sweeping is not necessary. Also, the open structure has the advantage of an increased pumping behaviour as well of a higher beam transmission[3]. Furthermore, there is only a weak dependence of the power loads to the detailed magnet design and source geometry; hence, our design offers a large operating window for beam alignment, divergence and steering.

However, the present design needs more space – about 400 mm – as the electrostatic case. This additional space could be provided by smaller gaps between neutralizer and the magnet as well as between the target plates and the calorimeter; this may be possible due to the better pumping behaviour of our open design.

[1] A. Panasenkov, ITER Review Meeting on Neutral Beams, NAKA, May 2001.
[2] P. Franzen, et al. Fusion Engineering and Design, Vol. 56–57, 2001, p. 511.
[3] H. P. L. de Esch, CEA Cadarache, private communication.

C-31

EXPERIMENTAL STUDIES OF THE JET NBI NEUTRALISER PLASMA

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Neutralisation efficiency is an important issue in the design of neutral beam injection (NBI) systems for heating and fuelling fusion plasmas. The JET Neutral Beam Injectors system do not reach the efficiency which is expected on the basis of current understanding of the neutralisation process. If this neutralisation efficiency deficit could be eliminated by re-design of the neutralisers, additional neutral beam heating power would become available.

The interaction of the high energy H+ or D+ ions and the gas in the neutraliser cell results in the partial neutralisation of the beam and also in the formation of a low temperature plasma inside the neutraliser. The interaction between this plasma, the neutral gas and the walls can give rise to various phenomena capable of depleting the neutral gas molecular density hence reducing the effective neutralisation target. For example, wall pumping, gas heating and modification of the flow regime are all possible effects.

This paper presents the results of a number of experiments that were carried out in order to measure the plasma and gas parameters inside the neutraliser. The plasma parameters were obtained using a combination of planar Langmuir probes and a novel electrical diagnostic called a ‘plasma eater’. The gas pressure was measured using hot cathode ionisation gauges and capacitance manometers. An attempt has been made to measure the neutral hydrogen gas temperature using a spectroscopic technique that requires the measurement of the intensities of the spectral lines of the hydrogen Fulcher-α system*. The experiments were carried out at the JET Neutral Beam Test Bed which was modified by inserting a flange containing eight ports in place of the isolation gate valve between the first and second stage neutraliser. The beam was produced from the recently upgraded JET Positive Ion Neutral Injector (PINI) which is capable of delivering a 130 kV/60 A deuterium beam. The measurements were made over a range of beam energies from 40 keV to 90 keV and the results include power, pressure and energy scans in both hydrogen and helium. In addition, time resolved measurements of the plasma and gas parameters are also presented. The results compare reasonable well with a 1-dimensional model based on plasma diffusion.

*T. Gans et al., Plasma Sources Sci. Technol. 10 (2001) 17–23.

C-32

ACCELERATOR R&D FOR JT-60U AND ITER NB SYSTEMS

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The neutral beam (NB) injection has been one of the most promising methods for plasma heating and current drive (H&CD) in tokamak fusion devices. For JT-60U and ITER, required beam performances are 500 keV, 10 MW (from 2 ion sources) for the pulse length of 10 s, and 1 MeV, 16.5 MW and 3,600 s, respectively, to achieve sufficient beam penetration in the large and dense plasmas for the H&CD. JAERI has developed high energy electrostatic accelerators for the NB system in JT-60U and ITER.

The JT-60U negative-ion based NB injector was constructed and is operational since 1996. The longest pulse length achieved in JT-60U negative-ion based NB injector is 5.35 s at 360 keV for D^0 injection of 1.9 MW (from one ion source). One of the issues limiting the NB injection pulse length [1] was excess heat load and resulting temperature rise of the NB port limiter on JT-60U. Recent observation of the beam footprint revealed deflection of beamlets at the joint of acceleration grid segments, which form each grid with 5 segments. The cause of deflection was identified due to distorted electric field by a step of 5 mm deep at the bottom of the extractor. By filling the step with metal bars, the deflection was corrected, and the heat load on the limiter was decreased to less than a half of previous value. As the result, a continuous injection of H^0 beam was succeeded for 10 s with the NB power of 2.6 MW using one ion source at 355 keV.

After long pulse conditioning of the JT-60U accelerator, many traces of discharge were found on the inner (vacuum side) surface of the insulator ring made of FRP (fiber reinforced epoxy, dielectric material). The traces started at triple junction (interface point of insulator, metal flange and vacuum). Electrostatic analyses of the accelerator interior structure showed that the electrostatic field concentrated at the triple junction, as high as 3 kV/mm. To lower this stress, metal structure called “stress ring” was designed on a basis of the analyses. The analysis shows reduction of the stress at the triple junction to ~ 1 kV/mm. The initial test of the JT-60U accelerator without beam acceleration showed fast increase of the voltage hold off up to 460 kV with the stress ring. At present the voltage is limited to avoid possible trouble on the power supply since it is the highest voltage ever achieved. A similar stress ring was also designed and installed in 1 MeV accelerator used in the ITER NB R&D. The test of the accelerator in 1 MeV test facility showed stable increase of the voltage hold off with the stress ring.

[1] M. Kuriyama et al. Proc. 20th Symp. Fusion Tech., Marseille, 7–11 Sep. (1998) 391–394.

C-33

PERFORMANCE OF THE HL-1M NEUTRAL BEAM INJECTION SYSTEM

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To heat plasma in the HL-1M Tokamak, a neutral beam injection system, HL-1M NBI has been built. Its design parameters are 0.8MW neutral beam power, 45kev ion energy, 0.2s duration. The arc discharge volt-ampere characteristic at this ion source is divided into three regions: the first positive resistance region in which the arc discharge is operated inefficiently, negative resistance region and the second positive resistance region in which arc discharge is operated high efficiently. The first injection experiment at the HL-1M was begun at 0.4MW and ion temperature rise in HL-1M from 400ev to 800ev was obtained in 1998. At that time the arc discharge of the ion source was only operated at the first positive resistance region, so the extraction current is less than 20A. Since then, the ion source system has been improved, including reducing the transient region in the arc discharge by improving the arc power supply and adjusting the arc discharge condition, reducing the particle deposited on the grid through finding optimum extraction condition, improving the control system and so on. The ion current of up to 50A at energy 25kev with 0.2 s duration was achieved in Nov. 2000. The ion temperature of the plasma in Tokamak is up to 1kev when the ion source arc discharge is operated at second positive region, but a plenty of impurity is produced and the discharge of the Tokamak is interrupted during neutral beam injection. In this paper the HL-1M NBI performance characteristic will be described. Parts of plasma heating experimental result by neutral beam will be shown in this paper.

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DESIGN AND OVERVIEW OF FABRICATION TESTS FOR THE 1 MV BUSHING FOR ITER NB SYSTEM

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The ITER NB System consists of two injectors each delivering 16.7 MW of D0 beam to the plasma at 1 MeV. Each injector has a single negative ion source connected to a 1 MeV electrostatic accelerator. The accelerated negative ions pass through a neutraliser then an electrostatic deflection system, where residual ions, exiting the neutraliser, are dumped. The neutral part of the beam continues to the torus through a duct connecting to the torus vessel. The 1 MV bushing is an insulated cylindrical, feed-through (about 2 m diameter and 2.5 m height) which forms the physical boundary between the gas-insulated transmission line and the torus primary vacuum. This bushing is part of the primary safety barrier and of the primary vacuum confinement, yet it allows all the electric, gas and coolant services to reach the ion source and the electrostatic accelerator. The design of this component has to fulfil a set of requirements including 1 MV class insulation in vacuum for a large number of services, vacuum compatibility, operation under neutrons and gamma radiation, safety requirements for pressure containment; moreover, the components has to remain within maximum size limitation driven by a reasonable layout.

This has been accomplished with an innovative design. The HV bushing consists of 6 flanges separated by cylindrical insulators and contained inside a vessel filled with pressurised insulation gas (sulphur hexaflouride, SF6); between two flanges, insulation is provided by an inner alumina ring and an outer fibre-reinforced plastic ring. The inter-space between the outer insulator ring (FRP) and the inner insulator ring (alumina) is pressurised with dry nitrogen as insulating gas. This arrangement allows to have a guard volume that prevents any leakage of SF6 (potentially damaging for tritium catalysts) inside the primary vacuum, and provides a decoupling where the largest share of the mechanical loads are taken from the robust and conventional fibre-reinforced plastic ring, relieving the more delicate alumina insulator.

Mechanical analyses of the component have been performed showing that satisfactory safety margins can be achieved. The selection of materials is discussed with consideration of the electrical and nuclear requirements.

The manufacturing of the large diameter (about 1.6 m diameter) alumina insulator stays within the limit of existing technology and the brazing used to seal the alumina rings to the flanges is an established process, though at smaller dimensions. To validate the actual design a reduced scale prototype including all the relevant features of the insulator has been successfully manufactured and tested, passing all the mechanical, vacuum and high voltage tests.

C-35

EFFECT OF ARGON SEEDING ON THE NEGATIVE ION YIELD OF THE KAMABOKO III ION SOURCE

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It has been previously reported that the addition of argon to a hydrogen plasma in an RF driven ion source can substantially increase (up to a factor 3) the extracted and accelerated negative ion (H-) current [1]. Realizing such an increase in the filamented arc discharge negative ion sources used for neutral beam injection systems would have significant benefits. Unfortunately the reported studies of argon addition to filamented sources have not shown a similar gain, but so far these have been carried out with arc powers and plasma densities far from those typical of the plasma in the negative ion sources used on neutral beam injectors [2, 3]. The KAMABOKO III ion source operates at the pressure and plasma density close to those anticipated in the ion source proposed for the ITER neutral beam injectors.

Measurements have been made of the plasma density, electron temperature and the negative ion yield as a function of the argon seeding rate. The plasma parameters are determined with a fast spatially scanning Langmuir probe system. The effect on the H- yield is determined from the effect on the current extracted and accelerated from the source. Data will be presented for source filling pressures between 0.1 and 0.5 Pa of hydrogen, additions of argon from 0 to 30 %, and a discharge powers of 38 kW. Some increase in the extracted H- yield is measured for small percentage additions of argon, (0–20 %), but only at the highest H2 pressure used, 0.5 Pa.

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[2] N. Nishiura et al, Rev Sci Instrum 73 (2), 2002
[3] N. Curran et al, Plasma Scources, Sci. Technol. 9 2000

C-36

130kV 130A HIGH VOLTAGE SWITCHING MODE POWER SUPPLY FOR NEUTRAL BEAM PLASMA HEATING. DESIGN AND ASSEMBLY ISSUES

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The company JEMA has designed and manufactured two High Voltage Switching Mode Power Supplies (HVSMPS), rated at 130kVdc and 130A, which will feed the grids of two PINI loads, each one to be installed at the Joint European Torus (JET facility in Culham).

The main requirements are a pulse mode operation (maximum ratio, 20sec On, 580sec Off), a relative accuracy of ±1300V, a switch-off time of 7μs, a low energy stored at the high voltage output side, the possibility of performing up to 255 re-applications of the high voltage during a 20 second pulse and the possibility of reversing the polarity of the supply.

The solution designed by JEMA includes 2 matching transformers which adapt the 36kV of the grid to the required 670V at the secondary side. Additionally, such transformers provide a 30 degrees phase shift which is required by a 30000A 12 pulse thyristor rectifier. The obtained and stabilised 650 Volts feed 120 IGBT inverters, which operate at 2778Hz with modulated square waveform.

Each inverter feeds a High Insulation High Frequency Transformer. The 120 transformers corresponding to one power supply are arranged in 3 oil filled tanks and provide the main insulation from the low voltage to the high voltage side. The square waveform obtained at the secondary of each transformer is rectified by means of a diode bridge. The connection in series of the 120 diode bridges provides the required 130kVdc at the output.

In order to protect the load, a redundant solid state crowbar has been designed. Such short circuiting device is composed of 26 Light Triggered Thyristors (LTTs), connected in series.

The scope of the delivery includes a high voltage tri-axial cable of 200 metres, and two current limiting reactors, which dump the energy transfer to the load in case of breakdown. Additionally, the system includes high voltage and high insulation measurement devices, safety interlocks and a water cooling plant.

Electrical simulations have been carried out in order to ensure that the system complies with the requirements of high accuracy and adequate protection of the load. The critical design of the High Voltage-High Frequency Transformers has also required electrostatic simulations of the electric field distribution.

This article describes the most relevant design and manufacture issues.

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NEW POWER SUPPLIES FOR JET NBI TESTS AND PRELIMINARY RESULTS

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With regard to the European Fusion Development Agreement (EFDA), the European Atomic Energy Community has place an order for the delivery of two power supplies for the enhancement of the JET Neutral Beam system. Each of them, with 130 kV and 130 A during 20 seconds, will feed two injectors. The power supplies, whose design is presented in a separated paper, are planned to go in service in autumn 2002. But before delivery to the JET site, both units will be completely assembled in a new test hall located beside the Jema factory, and fully tested on a dummy load.

The present paper will go through the type tests of the power supply main elements and report in detail about the tests of the complete assembly. The peculiar choices which make the interest of this new kind of technology, similar to the ECRH power supply installed in 1990 in the CIEMAT – Euratom Association in Madrid, will be analysed and their performances will be presented.

Based on the series connection of 120 fast rectifiers each providing 1100 V and 130 A, their associated insulation transformer designed for an insulation voltage of 260 kV DC and the IGBT inverters working at 3 kHz which are connected to transformers primaries, the power supply is very promising with regard to the low residual energy which can flow to the load in case of short-circuit. The adopted pulse width modulation technique together with the phase shift between each one of the 120 inverters, allows for a rapid and precise output voltage regulation. Very fast cut-off delay together with a crowbar based on light triggered thyristors, will allow for a rapid discharge of the energy stored in the 200 m long DC line which connects the power supply to the load.

The tests procedures which apply to the insulation transformers and associated rectifiers will be described and the results presented. The output voltage waveforms in steady state, during load short-circuits, fast cut-off, and voltage reapplication will presented and analysed.

The intensive test programme which has to be carried out with a complete installation identical to the final one, where the only difference will be the replacement of the Neutral Beam Injector by a resistive dummy load, should allow for a fast commissioning on site and high security level at the operation start of the injectors.

C-38

THE PAM LAUNCHER FOR FTU: RESULTS OF THE PRELIMINARY TESTS

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In the frame of a collaboration between ENEA–Frascati and CEA–Cadarache, a Passive Active Multijunction (PAM) antenna has been designed and built to validate on FTU the concept of this launcher.

12 PAM modules arranged in 3 vertical rows compose the launcher. Each module has 2 active and 2 passive waveguides at the launcher mouth. The cross section of a waveguide is 28 × 5 mm^2 and the vertical wall thickness is 0.8 mm; the depth of the passive waveguides is 0.25 λ_g. The phase difference between active waveguides is set to 270°.

With these parameters, and 180° between modules on the same horizontal row, the $N_{\|}$ peak of the structure is 2.42 according to the analysis made by a coupling code. The $N_{\|}$ of the main wave spectrum can be varied between 2 and 2.82 when the phase between modules is varied in the range ±90° around 180 degrees. The power directivity is close to 70% for electron density at grill mouth near cut-off density.

The complete characterisation of the launcher on the test bench, both at signal level and at high power, is now starting. This characterisation must be completed by the end of summer to allow the assembly of the PAM on a FTU coupling structure (three traditional grills assembled in a vertical row) in substitution of its upper grill. This coupling structure will be installed on FTU during the next winter shut down. First experimental results are expected during the 2003 spring campaign.

C-39

LATEST RESULTS FROM THE CADARACHE 1 MV SINGAP EXPERIMENT

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The European concept, the SINgle GAP, SINGle Aperture (SINGAP) accelerator has been developed as a simplified alternative to the Multi-Aperture, Multi-Grid (MAMuG) accelerator of the ITER Neutral Beam reference design. SINGAP accelerates a pre-accelerated beam with an energy E of tens of keV to E ~1 MeV in a single step, whereas MAMuG performs this using intermediate grids. The objective of the present experiments is to demonstrate reliable acceleration of a D^- beam to 1 MeV with parameters relevant to the ITER Neutral Beam Injector requirements (350 mm main acceleration gap and D^- current density j_{D^-} =200 A/m^2).

During previous studies it has been demonstrated that the SINGAP concept works and good quality 860 keV H^- beams (43 mA, 1 s, j_{H^-} =40 A/m^2) and 630 keV D^- beams (106 mA, 1s, j_{D^-} =50 A/m^2) have been produced. In the recent experimental campaign a refurbished 1 MV epoxy insulator / bushing has allowed high energy D^- beams to be produced. Refurbishment of the 1 MV bushing became necessary after the top 2 of the 9 epoxy rings making up the bushing had been perforated and carbonised due to high voltage discharges.

The best results achieved during the recent campaign are 911 keV D^- beams (new world record) with a current of 33 mA and a duration of 1 s and 600 keV D^- beams with a current of 100 mA, 1s, and a current density of 70 A/m^2.

The prototype accelerator so far used has had an acceleration gap of 625 mm, which was larger than the one foreseen for ITER (350 mm). Also, the pre-accelerator is far from optimised. The main accelerator gap is now shortened in order to have the same length as the ITER version and voltage holding tests have been performed with and without beam.

The paper will describe the actual accelerator layout, the results from the recent experimental campaign and the latest results from the experiments with the modified acceleration gap. The planned development to reach the required D^- current density (200 A/m^2) and the proposed new (optimised) pre-accelerator beam optics required for producing “ITER-like” beam optics will also be described.

C-40

GRIDDED TUBES IN FUSION APPLICATIONS

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THALES ELECTRON DEVICES (TED) has been for many years involved in fusion additional heating and CD schemes. In the ICRF range, TED has developed and supplied gridded tubes and associated RF circuits, in short and multisecond pulses. The most recent development has been an RF amplifier stage, continuously tunable in the 27 MHz to 55 MHz range, and delivering 250 kW cw on a mismatched load with a VSWR = 1.7 : 1 (any phase). The tetrode used is a TH 519. TED has also developed the high power diacrode® TH 628 (1 MW cw at 200 MHz on a matched load). The diacrode particular design allows a reduction by 4 of the RF losses compared to those in a conventional tetrode. This diacrode TH 628 is a proof of feasibility for an upgraded model aimed to the future ICRF needs, with a target performance of 3 MW cw on a matched load at 70 MHz (or 2 MW cw with VSWR = 2 : 1).

The paper shall describe the present status of these products and discuss the potential for the future.

C-41

A NOVEL APPROACH OF HIGH VOLTAGE FILTER DESIGN FOR SMOOTHING THE OUTPUT VOLTAGE OF HIGH VOLTAGE HIGH POWER PULSE STEP MODULATORS

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At the Max-Planck-Institut für Plasmaphysik in Greifswald, Germany, there are planned several plasma heating systems with high output power. There is on the one hand an ion cyclotron resonance heating (ICRH) with a pulsed output power of 2 MW per RF amplifier, a neutral beam injection (NBI) with a pulsed output power of up to 3 MW per accelerator and an electron cyclotron resonance heating system (ECRH) with totally 10 MW output cw with 10 tubes. For the supply of the different systems universal high voltage power sources with totally 26 MW electrical power in cw operation and 52 MW in pulsed operation are under construction.

The output voltage can be adjusted up to 130 kV. The system has a low amount of stored energy. For this reason a commonly used crowbar system is not necessary in most of the considered applications. The power supplies have the capability of modulation up to 500 Hz square wave with a modulation degree of one and several kHz with lower degrees. To meet all requirements for all the heating systems, the supply can be configured concerning polarity, maximum voltage and current, modulation parameters, series operation, pulse length supervision etc. The supply comes in four units where the first unit with 6.5 MW cw is in operation since end of 2001.

Especially for the ECRH heating system there are high requirements on the stability and the quality of the DC output voltage. Due to the capability of modulation and the requirement on low amount of stored energy, the output filter of this switched mode power supply has to be designed very carefully. For this, it is not enough taking principle effects like noise due to pulse width modulation into consideration but also noise which comes from charging and discharging of hundreds of parasitic capacitors in the system.

A good approach to meet this problem is to model the system and analyzing it with the simulation tool SPICE. A comparision of the simulated results with measurements, which were done at the first high voltage power supply situated in Greifswald, shows good agreement. In addition the simulation results gives rules how to build a high voltage filter with a low amount of stored energy and a good supression of the unwanted noise. With the new filter design it is possible to build up switched mode supplies for gyrotrons and other vacuum electronic systems.

C-42

A DOUBLE DISK WINDOW FOR THE JET EP ECRH SYSTEM

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The work of double disk window design for the JET EP ECRH project has been done at FZK Karlsruhe. The window unit has to provide a reliable barrier between the torus and the outer system for tritium containment. An additional requirement is that the microwave transmission of the double disk arrangement has to be suitable for two frequencies: 113.3 GHz and 170.0 GHz. Based on a new development of brazing techniques the design of the window unit has been modified. Detailed calculations of the microwave transmission characteristic of a double disk window have been performed. In particular the influence of mechanical tolerances on the transmission characteristic have been investigated in order to be able to specify the mechanical dimensions of the CVD-diamond disks and the window unit.

The calculations have been performed for several distances d_{gap} between the disks. The first result was that an uncertainty of ± 10 micron in the thickness of each disk of the pair used in a window unit cannot be tolerated since it may cause an unacceptable reduction of the transmission characteristic (power reflection > 1%). In order to obtain sufficiently low power reflection (< 1%) the pair of CVD-diamond disks have to be within the following geometrical specifications: mean thickness of both disk used in one window unit: $d_{disk0} = (1.111 \pm 0.010)$ mm; thickness of each disk: $d_{disk} = d_{disk0} \pm 0.005$ mm. The optimum gap distance d_{gap} between the disks depends slightly on the mean thickness d_{disk0} of both disks and has to be determined for each window unit individually taking the mean value d_{disk0} of the selected disk pair. Depending on $d_{disk0,}$ the optimum gap distance is between $d_{gap} \cong 3.4$ mm and 3.2 mm for 113.3 GHz. Unfortunately, the optimum gap distance d_{gap} is slightly different for 170.0 GHz, namely between $d_{gap} \cong 3.2$ and 3.0 mm. Another possible gap distance suitable for 170 GHz is e.g. between $d_{gap} \cong 4.1$ and 3.9 mm or between $d_{gap} \cong 5.8$ and 5.6 mm. The mechanical design of the window unit has to allow the modification of the distance between the disks in order to fulfill the requirements of operation at both frequencies. With the assumed thickness tolerance of ±5 micron the reflectivity of the double disk window unit is in the worst case -25 dB at 113.3 GHz and about -22 dB at 170 GHz. In both cases the bandwidth is as large as about 2 GHz.

C-43

MIRROR DEVELOPMENT FOR THE 140 GHz ECRH SYSTEM OF THE STELLARATOR W7-X

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The stellarator W7-X which is currently under construction in Greifswald, Germany, will be equipped with a powerful ECRH system, working at 140 GHz. The ECRH system is designed to operate in cw regime. The microwave power will be generated by 10 gyrotrons delivering 1 MW each and will be transmitted from the gyrotron hall to the machine via a fully optical system. The mirrors (more than 160) are water cooled and can be adjusted by step motors remotely. The transmission line consists of a single beam waveguide part (SBWG) and a multi beam waveguide part (MBWG). In the SBWG part (length appr. 10–15 m), close to the gyrotrons, the mirrors are related to the individual tubes, two confocal MBWG lines (length appr. 40 m) support 5 beams simultaneously each. At the stellarator the beams are separated again and launched by individual antennas to the plasma.

To reach high transmission efficiency, the design of the mirrors must guarantee a mechanically stable surface under the heat load imposed by the ohmic loss of the millimetre waves. The design consists of a 60–70 mm thick honeycomb structure from stainless steel and a thin (2 mm) sheath of electro-formed copper on the mirror surface. Optimized cooling channels are milled directly below the copper in the stainless steel structure and form one or several spirals going from the centre to the edge of the mirror. According to thermo-mechanical calculations, a very low thermal deformation is obtained with this design, even during the transient phase in the first minute after power switch-on. This behaviour has been confirmed experimentally in a test set-up with a MBWG type mirror. It turned out, that deformation under application of a heat load equivalent to the absorbed power from a 1 MW mm-wave beam is in the tolerable range.

Water cooled mirrors of the SBWG type have been used successfully during conditioning of a prototype gyrotron with a mm-wave power of 740 kW and a pulse length of up to 100s . To test and optimise in-situ beam and power measurements of the CW ECRH system on W7-X these mirrors are equipped with different diagnostics tools. We use a grating coupler, a waveguide coupler integrated into the surface of a mirror and sensors for calorimetric measurement of millimetre wave absorption.

Measurements and results on the deformation of mirrors for cw high power application in the mm-wave range will be presented and problems of the actual design will be discussed.

This work has been performed in the frame of the project PMW, ECRH for W7-X hosted at FZK Karlsruhe (collaboration between FZK Karlsruhe, IPP Garching and Greifswald, and IPF Stuttgart).

C-44

DESIGN AND R&D OF AN ECRH SYSTEM ON JET

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An ECRH (electron cyclotron resonance heating) system has been designed for JET in the framework of the JET Enhanced-Performance project (JET-EP) under the European Fusion Development Agreement (EFDA). Due to financial constraints it has recently been decided not to implement this project. Nevertheless, the design work conducted from April 2000 to January 2002 shows a number of features which can be relevant in preparation of future ECRH systems such as the ITER one.

The ECRH system was foreseen to comprise 6 gyrotrons, 1 MW each, in order to deliver 5 MW into the plasma. The main aim was to enable the control of neo-classical tearing modes (NTM). A frequency of 113.3 GHz was selected to enable a wide range of operating toroidal magnetic fields from 3 to 4 T. Furthermore, operation at second harmonic at lower fields was foreseen and the design was compatible with a possible upgrade to 170 GHz (the frequency presently foreseen for ITER), using the same systems including the double-disk diamond windows at the torus side. The main elements of the ECRH project were:

- 6 gyrotrons with a single-stage depressed collector, each 1 MW, 10 s, with a diamond window and linearly-polarised Gaussian output mode
- Evacuated corrugated HE_{11} waveguides to transport the mm-wave power to the double-disk diamond windows on the tokamak side
- A plug-in launcher, steerable in both toroidal and poloidal angle, that is able to handle 8 separate mm-wave beams. 4 steerable launching mirrors were foreseen to handle 2 mm-wave beams each. Water cooling of all the mirrors was a particularly ITER relevant feature. An existing test facility was being modified to include life-cycle tests. (A separate contribution "Design of an ECRH launcher for JET" is foreseen by B.S.Q. Elzendoorn)
- A power-supply and modulation system, including series IGBT switches, to enable independent control of each gyrotron. An all-solid-state body power supply was foreseen to stabilise the gyrotron output power and enable fast modulations up to 10 kHz. (A separate contribution "Design and R&D for an ECRH power supply and power modulation system on JET" is foreseen by A. B. Sterk).

C-45

DESIGN AND ANALYSES OF THE MAIN OPTICAL COMPONENTS OF THE ELECTRON-CYCLOTRON LAUNCHER FOR THE ITER

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According to the ITER task agreement the main optical elements of ECH&CD system were designed and analyzed. The following points were taken into account: vacuum environment; cooling; magnetic field; assembly. Thermal and stress analyses were conducted of the different designs of steerable mirrors reflecting one RF beam and eight RF beams; fixed mirror of the miterbend, diamond windows. Three variants of the electrical driving mechanism for steerable mirror were designed.

The results of thermal and stress analyses of the different design options of the steerable mirror reflecting one RF beam showed that maximum temperatures of the mirror structural materials do not exceed allowable maximum. Thermal stresses in the mirror structure materials are acceptable with the exception of the case of one cooling tube. Boiling will occur only in the case of one cooling tube; critical heat flux is not achieved in any case. Temperature rise of water between inlet and outlet cross-section consists 25^{o}C. Chosen materials (Cu-Cr-Zr alloy and SS-316) could be recommended as the base materials for the steerable mirror.

Thermal and thermal-stress analyses of the different design options of the steerable mirror reflecting 8 RF beams were carried out. Variants of copper alloy and molybdenum mirrors were considered. Temperature rise of the water coolant in the case of water flow velocity 2m/s; inner diameter of the cooling tubes 8 mm; maximum heat flux 3.2 MW/m^2 consists 60^{o}C. In this case nucleate boiling is possible. It seems reasonable to increase water velocity up to 3 m/s. Copper alloy mirror is preferable than molybdenum one due to lower value of maximum temperatures and stresses.

Thermal and thermal-stress analyses of the fixed mirror of the miterbend were carried out. Maximum temperatures in structure, equivalent stresses and displacement of the surface don't exceed allowable maximum.

The results of thermal and stress analyses of the different design options of the diamond window show that maximum temperature in the structure is not more than 164^{o}C in all cases. The value of stresses is relatively low and does not exceed allowable maximum too.

Electrical driving mechanism for steerable mirror was designed. Magnetic shield of the electrical motor was calculated. In order to provide electrical motor work in the external magnetic field B = 0.5 T it is necessary to use ferromagnetic shield with thickness $\delta \geq 8$ cm.

C-46

X-RAY PHOTOELECTRON SPECTROSCOPY STUDY ON CHANGE OF CHEMICAL STATE OF DIAMOND WINDOW IMPLANTED IONS

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An ECH/ECCD (Electron Cyclotron Heating and Current Drive) system is an important tool of fusion reactors. The ECH/ECCD system with high power rf power source has been developed in JAERI (Japan Atomic Energy Research Institute) for ITER (International Thermonuclear Experimental Reactor). A vacuum window is one of the most important components in the ECH/ECCD system development. Diamond is the leading candidate for rf windows. A window in a transmission line, the so-called a torus window, belongs to the primary boundary of the tritium and the vacuum confinement. The diamond using the torus window would receive PSI (Plasma Surface Interaction). The understanding of PSI effect for the diamond is very important to realize the ECH/ECCD system with diamond window and evaluate the aptitude for the window from viewpoint of fusion safety.

Chemical behaviors of implanted tritium into PFMs (Plasma Facing Materials) dominate PSI such as existing states, desorption behaviors of high-energy implanted tritium, and erosion processes of PFMs. In the present study, the change of chemical state of diamond implanted deuterium and/or the existing state of deuterium were investigated using XPS (X-ray Photoelectron Spectroscopy) technique.

A sample used in the present study was a polycrystalline diamond provided by Shannon Co. This sample was developed for radio frequency applications. The sample is 10.06 mm in diameter and 0.210 mm in thickness. As received sample was measured by XPS and then C1s XPS peak appeared at 288.06 eV. This peak was typical C1s signal of diamond. The sample was heated to 1373 K and holding for 3 min. After heating to degas, the diamond structure was changed from sp^3 state to coexisting state of sp^2 and sp^3, which was suggested by XPS measurement. More heating at 773 K for 10 min, in the region of XPS mean range, the diamond structure was changed to incomplete graphite structure. The C1s XPS peak was changed dramatically by only heating.

After heating, XPS measurement using Ar^+ implantation, which is an acceleration test for that using He^+ implantation, was carried out. Ar^+ implantation was carried out under implantation condition of 1.0 and 2.0 keV, 2.1×10^{17} Ar^+ m^{-2} s^{-1} (1.0 keV), 7.4×10^{17} Ar^+ m^{-2} s^{-1} (2.0 keV), and 2×2 mm^{-2}. The change of chemical state for diamond was suggested by the change of XPS spectra. C1s XPS spectra resulted in FWHM increasing to lower binding energy side (graphite XPS peak side) and peak shift toward graphite XPS side. In more implantation, however, the C1s XPS peak was shifted to higher binding energy side (diamond XPS peak side). The complex behavior of C1s XPS peak shift was observed and suggesting the dramatically change of chemical state for diamond. The results of XPS measurements using hydrogen implantation is very interesting for reason why the hydrogen provides with much chemical affinity to carbon.

C-47

THE INTERNAL VACUUM TRANSMISSION LINES OF THE ITER-LIKE ICRH ANTENNA PROJECT FOR JET

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A new ITER-like ICRH antenna is being developed to deliver over 7 MW at high power density (8 MW/m^2) to the JET plasmas [1].

Four Internal Vacuum Transmission Lines (VTL) are to be installed inside the antenna support box and feed RF power to the antenna straps through in-vessel matching capacitors mounted on the end of the internal-VTLs. It is foreseen to install a series of RF capacitive probes along the VTL in order to measure the VSWR (Voltage Standing Wave Ratio) on the VTL.

Key functional elements of the system are located inside the body of the Internal VTL conductor:

- The actuators are required to provide adequate force to overcome the spring constant of the bellows mounted in the variable capacitors and give linear travel with positional feedback instrumentation.

- The cooling system is designed to limit the temperature excursion of the capacitors. It must provide turbulent flow within the capacitor bellows convolutions to prevent air pockets forming, whilst simultaneously respecting the design pressure of the capacitor which is 3 bar absolute.

The various RF requirements, such as actuator accuracy (especially at high frequency) together with vacuum/tritium boundary issues, effects of disruption loading, and assembly considerations, make the Internal VTL design a challenging piece of work.

The paper will discuss the mechanical, hydraulic and electrical design of the new Internal Vacuum Transmission Line.

[1] F. Durodie et al., this Conference.

C-48

GAMMA AND PROTON IRRADIATION EFFECTS ON OPTICAL TRANSMISSION MATERIALS FOR PLASMA DIAGNOSTICS IN NUCLEAR FUSION REACTORS

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Ceramic materials are anticipated to play very important roles in developing nuclear fusion reactors, where they will be used under heavy irradiation environments (neutrons, gamma-rays, protons, helium and other ions) for substantial periods for the first time. We seriously need to understand the effects induced not only by neutrons and alpha particles but also by gamma and proton irradiation. Among the important applications of fiber optics envisioned for the near future are light pipes for monitoring Tokamak fusion reactor plasma conditions. We started a general program on gamma and proton induced degradation in optical transmission materials, including windows and optical fibres, mainly used for plasma diagnostics studies. Our project focuses on the comparison of the ionization and displacement induced damage in SiO_2 and Al_2O_3 based materials and the influence on the UV and visible optical transmission properties. The main characteristics of Bucharest irradiation facilities are:

- IRASM IRRADIATOR: 200 kCi ^{60}Co "swimming pool" irradiation source, minimum 2 kGy/h with an isotropic dose rate up to 1 Gy/s, ethanol chlorine benzene dosimetry, ionization chamber dosimetry, ESR dosimetry
- HVEC 8 MV FN TANDEM: protons up to 16 MeV and 200 nA, alpha particles up to 21 MeV and 50 nA, O, C, N ions approximately 100 MeV up to 50 nA.

Until now, we realized a gamma irradiation chamber, acting underwater, allowing various temperatures (20–300° C) on the sample, and a high energy proton irradiation chamber, acting in high vacuum, allowing various temperatures on the sample in the presence of irradiation proton beam heating (1 mA x 1 MeV = 1 W). As irradiations on different samples, we performed until now gamma irradiations between 2 and 40 Mrad (20 – 400 kGy) on some semiconductor lasers (visible range) and two alumina pellets (Merck). As post-irradiation studies we can mention optical absorption measurements in the range of UV-VIS range of gamma irradiated alumina pellets, performed at room temperature by means of a Cary 5 (Varian) double beam spectrophotometer. The spectral bandwidth was kept to 1 nm. Some preliminary results are shown where the diminution of the transmission caused by the gamma irradiation is clearly displayed. Concerning the 15 MeV proton irradiation, the main problem is the induced radioactivity in the sample. As a preliminary test two alumina pellets (Merck) were 5×10^{15} and 10^{16} proton implanted at 15 MeV; the residual radioactivity measured after 48 hours was approximately 30 μGy at the surface. We have waited for six weeks for a diminution to 3–5 μGy to start the post-irradiation experiments. Absorption-absorptivity, reflectivity and transmission spectra are presented.

E-1

RADIATION INDUCED ELECTROMOTIVE FORCE (RIEMF) AND VOLTAGE DRIFT IN MAGNETIC COIL

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A radiation induced electromotive force (RIEMF) is a phenomenon, where an electric power is generated between two materials which are electrically separated by an electric insulator under irradiation. The phenomenon itself has been realized for a long time and many related researches have been carried out up to now. In general, the RIEMF is understood to be a current driven electric source and its voltage is determined by a current generated by the radiation field and by an impedance of insulator separating two materials. A typical example of the RIEMF is a current and voltage generated between a center lead and a sheath of mineral insulating cables (MI-cables). The RIEMF is now considered to be one of main sources of electrical noises for diagnostic components in fusion devices with burning plasma.

Magnetic coils are expected to play crucial roles in plasma diagnostics. In the case of International Thermonuclear Experimental Reactor (ITER), magnetic coils should work reliably for an expected plasma operation period, namely extending to be nearly 1000s, under intense radiation environment. In ITER-EDA (Engineering Design Activity), a magnetic coil made of MI-cable is selected as a primary candidate for a magnetic field diagnostics. It is robust and its radiation resistance is good in general, namely its structural and electromagnetic wholesomeness will withstand fast (E > 1 MeV) neutron fluence up to 10^{25}n/m^2. Also, its electromagnetic properties were well evaluated quantitatively under the ITER-relevant irradiation conditions. The RIEMF generated in MI-cables themselves and in coils made of MI-cables were quantitatively measured under gamma-ray irradiation in cobalt-60 cell and under JMTR irradiation. Electrical voltage of 1–10V was observed between a center lead and a sheath. The voltage had very weak dependence on intensity of irradiation field but had temperature dependence. In the meantime, electrical current generated between a center lead and a sheath had nearly linear dependence on the intensity of irradiation. These results agreed with accumulated data on the RIEMF. The experimental results and their analysis showed that the RIEMF generated between a center lead and a sheath would not directly contribute to the drift voltage apprehended in the diagnostics. However, there are some possibility that the RIEMF may become a signal disturbance to diagnostics. The paper will describe recent results on measurements of the RIEMF in ITER-relevant magnetic coils.

E-2

STUDY OF THE RADIATION INDUCED ELECTRO-MOTIVE FORCE EFFECT ON MINERAL INSULATED CABLES FOR MAGNETIC DIAGNOSTICS IN ITER

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To reconstruct the shape and position of the plasma boundary of ITER, in-vessel magnetic coils and loops will be employed. These coils, made of mineral insulated (MI) cables, will most likely be affected by radiation. Whereas radiation levels on ITER will be similar to machines that operate in Deuterium/Tritium (JET, TFTR), the pulse duration will be roughly 100 times longer (longer integration time for the coil-induced voltages). In addition, the requirement for ITER on accuracy will be two times higher. Any spurious steady voltage appearing across the coil terminals will thus be 200 times more significant than on most other devices [1]. A good understanding of the physical processes giving rise to radiation induced currents and voltages in MI cables is thus required to allow to minimize the impact on magnetic coil measurements or on other tokamak diagnostics employing MI signal cables.

In this study, an extensive literature review of the Radiation Induced Electromotive Force (RIEMF) effect is presented and a detailed Monte Carlo model has been developed to calculate the radiation induced currents in cables in a prescribed neutron and gamma field. Using this model, the impact of cable geometry, gamma spectrum, core and insulator material as well as the surrounding material on the RIEMF induced currents has been studied in detail. The model has been validated using data from Self Powered Neutron Detectors and shows excellent agreement [2] between calculated and measured values. On the basis of the calculations, a set of cables was selected for test irradiation in the BRIGITTE gamma ray facility. The measured RIEMF induced currents in these cables are compared to the theoretical calculations.

Provided the neutron and gamma spectra in ITER are sufficiently known at the location of the magnetic probes, the code can be used to design optimal cables with minimal radiation-induced currents and concomitant induced voltages. For a typical copper-core MI cable (outer diameter 2 mm) envisaged for use in ITER magnetic diagnostics with a total length of 10 m, the model predicts a delayed neutron induced current of about 1 μA in addition to a gamma induced current of about 1 μA (using rough estimated values for gamma and neutron fluxes). Based on this study, some recommendations on MI cables and the electronic detection system for magnetic coils for ITER are given.

[1] G. Vayakis, "Assessment of the probable effect of RIEMF on magnetic measurements for ITER", 14th Diagnostics Expert Group, Jülich, March, 2001.

[2] L. Vermeeren, "Absolute on-line in-pile measurements of neutron and gamma fluxes using self-powered detectors: Monte Carlo sensitivity calculations", OECD Halden Reactor Project, HPR-356, 2001. Presented at the Enlarged Halden Programme Group Meeting in Lillehammer, Norway, 11–16 March, 2001.

E-3

NEUTRONIC ANALYSIS OF DIAGNOSTIC SYSTEMS IN PORTS OF ITER

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This presentation gives an assessment of the neutronic effects on ITER of in-port diagnostic systems elements and the integrated port environment. Optimized designs of several key diagnostic systems and integrated diagnostic are considered in detailed neutronic calculations. Resulting values are given of nuclear heating and radiation damage in materials for special diagnostic systems; absorbed gamma-doses in imbedded diagnostic components, connectors and cables; the effectiveness of the proposed shielding of the integrated diagnostic ports with respect to heat load at coils & activation at the port plug seal. *The design of complicated diagnostic in-port systems is described.

The key diagnostic systems include: the Neutron Camera through the Upper vacuum vessel; the Motional Stark Effect (MSE) system, Radial Neutron Camera, Thomson Scattering system (LIDAR), Polarimetry and Wide Angle Viewing systems in equatorial ports; Edge Thomson Scattering system in upper ports; Remote Handling In-Vessel Viewing system in divertor port. The nuclear responses in specific optical elements such as first mirrors, shutters, windows etc are established.

The design of integrated port structures are developed and optimized, in particular for the Viewing system’s Head Mirror Assembly, the labyrinth in equatorial port with LIDAR, the Optical/wave guide labyrinth in the equatorial port with MSE common apertures, additional shielding in port interspaces. Also given is the effect of combining several diagnostics in one port and the effect of an uncooled port wall.

The activation of the cryostat and other systems nearby and the dose rate is considered at regions of diagnostic apertures and equipment.

The results presented show that designs evolved for these diagnostic systems meet the specified requirements of nuclear effects on the ITER machine and on diagnostic components.

E-4

STEADY-STATE DATA ACQUISITION METHODS FOR LHD DIAGNOSTICS

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In case of the recent quasi-steady-state fusion devices, the non-stop real-time operation becomes indispensable to the data acquisition system, which was not significant in conventional short-pulse experiments. The newest non-tokamak fusion devices applying the superconducting magnets, such as LHD (Large Helical Device) and Wenderstein 7-X, usually plan to hold a quasi-steady-state experiment with over ten minutes plasma duration. In such cases, the data acquisition system also has to run in real-time so that it can display the transient behaviours in accordance with the plasma discharge going on.

The LHD diagnostics have over 30 kinds of plasma measurement devices and their total number of CAMAC modules and channels are about 300 and 2000, respectively. In the 2001 campaign, their acquisition data makes up to 620 MB/shot in 150 shot/day usual operation. Under those situations, the effective utilization method of existing digitizer resources, such as CAMAC and VMEbus modules, also becomes an important subject. It also sustains this subject that in short duration CAMAC modules still have enough capabilities to digitise any transient plasma phenomena.

In this study, 3 minutes cyclic operation scheme of the CAMAC-based data acquisition system has been developed toward the LHD long-pulse experiment, which succeeded to demonstrate the synchronized digitizer operations repeatedly. As there are too many signal channels in LHD to enable the 2-way alternating and break-less operation, additional diagnostics timing modules (VME) and wired-logic handling PLCs have been installed to generate/distribute the sub-structured control sequences within the master one. Sub-sequences will be practically recognized as a series of the short-pulse operation sequences so that all the acquisition instruments will run synchronously on them.

To complement the intermittent operation of the CAMAC digitizers in the long-pulse experiment, R&D for the wide-bandwidth real-time digitizer frontend (DFE) system has been proceeded simultaneously. For that purpose, the CompactPCI standards can smoothly replace the CAMAC digitizers because of its popularity and low price by the PCI compliance. As a preliminary result, the prototype system has achieved its continuous data acquisition and transfer performance up to 80MB/s in one DFE. Such fast steaming transfer of the massively-sized LHD physics data has proved its instrumental wide possibility.

The CompactPCI standard has an affinity with the PCI bus so that it will be also quite applicable for the PC-cluster or PC-based distributed system, especially in construction of a new data acquisition system. Its complementarity with the conventional CAMAC and VMEbus based system has been successfully verified in LHD.

E-5

THE INSTRUMENTATION OF THE ITER PORT-LIMITER

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The ITER port-limiter with the alignment system are intended to work at high temperatures (up to 750°C), under action of cyclic both EM and thermal loads. The neutron flux exceeds 10^{22} 1/sm^2 on the outer surface. The force-flow water is applied to cool the port-limiter down. The total flow rate achieves to 60 kg/s, and pressure achieves to 4.3 Mpa, that requests of effective methods to both leak tightness, and design strength monitoring.

An accuracy of the port-limiter alignment is requested to be not worst than part of millimeter and part of angle degree, that demands an special method of positioning. The common diagnostic of both thermal and stress condition of the port-limiter is requested to estimate the useful operating life and currently status.

Both methods and means are described in paper to measure the temperature of the limiter parts and alignment system, strain, acoustic emission, vibration. The methods and equipment are proposed to measure the heat load on the port-limiter, to monitor of the possible cracks appearance and evolution, to estimate the heat load distribution on the first wall of limiter, to observe the Halo currents passing across the alignment beam.

The original method is described how to define the accurate position of the port-limiter basing on the measurement of determined toroidal magnetic field by the sensors, fixed on the back side of the port-limiter. The overall view and test results of the prototype of positioning system are given.

The schemes of the primary transducers on the back side of limiter and alignment beam are proposed and discussed. The properties of the transducers with the improved radiation durability. The schemes of measurement are presented and considered detailed.

E-6

RADIATION DEFECTS IN ANTIFERROELECTRIC THIN FILMS

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Irradiation effects on highly oriented antiferroelectric (AF) $PbZrO_3$ (PZ) films with a thickness of approximately 1 μm, are investigated in view of their possible application as a temperature sensitive element in a new bolometer system for fusion devices like ITER (International Thermonuclear Experimental Reactor). The advantage of this system is a lower sensitivity to electromagnetic noise compared to the present bolometer system. Furthermore, easier remote handling is offered, because only one transmission line is needed for several channels.

The films were deposited by a sol-gel technique on a TiO_2/Pt/TiO_2/SiO_2/Si substrate. Lead oxide was used as a precursor to enhance the film properties, and gold for the top contacts. We also present measurements on a 650 nm thick PZ film, deposited by PLD on a Si/SiO_2/Ti/Pt substrate with Pt top contacts. The dielectric constant of the films was measured in a frequency range from 1 to 250 kHz during a stepwise cooling down mode (~2°Cmin^{-1}) from 400°C to room temperature before and after irradiation to a fast neutron fluence of $2*10^{22}$ m^{-2} (E > 0.1 MeV). Furthermore, the hysteresis loop was investigated at 20 Hz at room temperature. After irradiation, the films were annealed in several steps up to ~400°C to remove the radiation induced defects.

We presume that the oxygen atom is the most sensitive element for neutron irradiation in perovskite like materials. The results are discussed in terms of two kinds of radiation effects, i.e. structural defects (oxygen vacancies) and radiation-induced charges, trapped at defect complexes. We find that the AF films show a significantly different behavior from ferroelectric (FE) films. After irradiation the PZ films show increased dielectric constants, but a decrease of T_c. This behavior could be related to the fact, that the net polarisation of the antiparallel ordered array of local dipoles is approximately zero in contrast to the polarisation in FE domains. The measurements also show that the radiation-induced charges play a significant role.

E-7

ELECTRICAL PROPERTIES OF MINERAL-INSULATED CABLE UNDER FUSION NEUTRON IRRADIATION

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In fusion reactors, mineral-insulated (MI) cables are expected to be helpful for plasma diagnostic system, reactor control system and others because of their good properties of high electrical insulation, heat resistance and mechanical strength. However, in severe radiation environment of high dose rate, the insulation of an MI cable degrades because electric charge is induced in the insulator by radiations. Sufficient irradiation data on electrical properties of MI cables are required for the evaluation of the reliability of the system with MI cables for fusion reactors. In the present study, fusion neutron irradiation experiments on MI cables were performed for the examination of the degradation of the cables.

Effects of fusion neutron irradiation on MI cables were examined by use of Fusion Neutron Source (FNS) at JAERI. The MI cable for irradiation had MgO insulator of 1.5 mm in thickness (cable diameter; 4.8 mm) and its length was 2.0 m. For the bias voltage of +200V between the central wire and the sheath, the increase in the leakage current (i.e., neutron-induced current) was ~ 5 pA during DT neutron irradiation of 2.9 x 10^7 n/cm^{2}s. Also, a large transient current was observed at the start of the neutron irradiation. On the whole, the neutron-induced current of the cable was approximately proportional to the neutron flux and the bias voltage, though a very small current was observed for the condition of the bias voltage of 0 V. The induced current by DT neutrons was ~ 3 times lager than DD neutrons. Behaviour of radiation-induced charge in the insulator was also discussed together with results of ^{60}Co gamma-ray and pulsed X-ray irradiation experiments on the same MI cable.

The voltage and flux dependence of the radiation-induced current of the MI cable suggests that the degradation of the insulation property is dominated by the production rate and drift of the charge in the insulator. The evaluation of the transient current at the change of neutron flux and the small induced current for the bias voltage of 0 V needs the precise treatment for the transport of electrons in the insulator. For the analysis of the mechanism of the electrical degradation of the MI cable, we give discussions with numerical calculations based on the Poisson's equation for the charge carrier transport in the insulator.

E-8

DIELECTRIC AND MECHANICAL PROPERTIES OF NEUTRON IRRADIATED KU1 AND KS4V GLASS

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KU1 and KS4V quartz glass provided by the Russian Federation within the ITER task sharing agreement has been shown to be highly radiation resistant with respect to its optical properties for use in both diagnostic and remote handling applications. The use of fused silica windows for broadband transmission in Electron Cyclotron Wave diagnostics imply their applicability over a wider spectral range reaching from DC/RF applications to optical systems. Material performance analysis requires mechanical strength studies together with dielectric property studies. Whereas mechanical strength in fusion ceramics like alumina and SiC was investigated in experiments reaching dpa levels, neutron irradiation data for fused silica have only recently come up with first results for KU1 studied at the 10^{-4} dpa level which is considered as an ITER relevant level for window materials.

The present paper extends the experimental data base to the KS4V material grade which is designed to extend the optical transmission for laser diagnostics further into the UV. The dielectric studies show that with this grade the mm-wave absorption is reduced by a factor of 3–4 relative to KU1 and thus it is comparable to the case of typical EC window grades such as Infrasil.

Dielectric and mechanical test specimens from both grades as well as from Infrasil as a reference material were neutron irradiated to 10^{21} n/m^2 and to 10^{22} n/m^2 (E > 0.1 MeV) at reactor pool temperature (T ≈ 320 K). Thus the effect of structural damage is focused to the step from 10^{-4} to 10^{-3} dpa.

First results from ongoing post-irradiation studies show the onset of a radiation-induced change in dielectric properties in this range. This is first put to evidence by slight increases in the permittivity and dielectric loss of KU1 and Infrasil measured at 90–100 GHz. Whereas for specimens irradiated to 10^{21} n/m^2, values identical to the unirradiated material are found, the permittivity increases after irradiation to 10^{22} n/m^2 from 3.81 to 3.83 and the dielectric loss tangent, tan□□□by about 10%. A similar onset of radiation-induced dielectric property changes was previously reported for polycrystalline and single crystal Al_2O_3 as well as for CVD diamond and was attributed there to the onset of point defect clustering.

Similar comparison will be given for the results from mechanical strength testing based on the ball-on-ring method which is still under analysis at present. In particular, it will be discussed to what extent the predominant role of the surface flaws which was put to evidence for unirradiated specimens and specimens irradiated to 10^{21} n/m^2 is still valid when the structural damage reaches to the 10^{-3} dpa level.

E-9

NEUTRON FIELD IN THE WENDELSTEIN 7-X HALL

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The (d,d)-reactions between the deuterons will produce neutrons with an average energy of 2.46 MeV which are shielded by the concrete wall of 180 cm thickness in case of Wendelstein 7-X. The knowledge of the neutron field inside the hall is of special interest for the various diagnostic equipment including neutron diagnostics. A detailed analysis of the neutron field has been carried out where the MCNP-Code [1] together with the nuclear cross section library ENDF/B-VI is used.

For the present computation, the cell model of experimental device and hall as prepared by Junker et al. [2] is used. The torus (i.e. modular field coils, auxiliary field coils, cryostat etc.) is modeled as circular torus with circular cross section, i.e. the only cell bounding surfaces are planes and cylinders. For the present purpose this is a fairly good approximation of the complicated twisted modular coils. The various cells are filled with material according to the specifications of the coils, supporting structure, heat shield, isolation etc. (i.e. Ni-Ti superconductor, copper, special steel, fiber glass epoxy etc.). The divertor structure, graphite tiles and openings for diagnostic and heating facilities are included in that cell model. The hall is filled by air; a deuterium plasma with 10^{14} particles per cm^3 is assumed (temperature 4 keV); the cylindrical wall is 180 cm thick (floor 180 cm, ceiling 120 cm) and consists of special boronized concrete (700 ppm boron). The ring source for the neutrons is located at R = 550 cm, z = 0 (cylindrical coordinates with origin at R = 0, z = 0) and produces $Q = 10^{16}$ neutrons per second with a Gaussian energy distribution (full half-width 0.2 MeV). Typically 20 to 60 energy groups are taken into account.

The total neutron flux and the energy spectrum is computed at various radial and vertical positions by using a ring detector, a point detector and control spheres (small radius 50 cm). The modification of the neutron spectrum by the presence of concrete wall (acting also as reflecting material; concrete walls with and without boron has been taken into account) and the other materials will be reported. There is only a weak dependence of the neutron flux on the radial position. For comparison results will be shown where the wall is modeled as hollow cube with the same wall thickness.

[1] J.F. Briesmeister (ed.), MCNP – A General Monte Carlo N-Particle Transport Code, Version 4B (LA-12625-M, Manual 1997).
[2] J. Junker, A. Weller, Neutrons at W7-X, Report IPP 2/341 (Oct. 1998).

E-10

QUARTZ MICROBALANCE: A REAL-TIME DIAGNOSTIC TO MEASURE DEPOSITION IN JET

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At present material erosion and tritium accumulation in fusion devices are key problems in fusion research. They will strongly influence the lifetime of ITER and the need for tritium removal. Thus measured data are vitally important to improve the database for modelling erosion particle transport and deposition layer formation.

At the end of the cleanup after the JET tritium experiment DTE1, 6g tritium of 36g injected were still remaining in the vessel /1/. It was found to a large extent as co-deposit in amorphous carbon layers grown on the inner louvers of the divertor. Those layers peel off due to internal stress after growing to a critical thickness. Flakes of the order of about 40 μm thickness were found.

To further understand co-deposition processes, the development of a new time resolved (at least shot by shot) diagnostic to measure in situ and the growth or removal of those carbon layers was approved and carried out.

The measuring principle of this new JET diagnostic named "Quartz Microbalance (QMB)" is based on the effect that the eigenfrequency of a quartz crystal integrated in a resonator circuit depends very sensitively on its mass. Therefore the calibrated change of frequency of the resonator circuit provides direct measurement of mass of the layer deposited on top of the quartz crystal. These changes may be due to particle deposition or removal of the layer and can be measured with a sensitivity of the order of 20×10^{-3} μg/ cm^2 Hz.

The challenge of the project was to develop a complete new system since available commercial systems did not meet the following JET constraints: • JET in vessel wiring (6 m) is unsuited to transmit high frequency ➔ electronics needed to be installed inside the vessel to reduce frequency for transmission (6 MHz to 6 kHz); • no cooling is accepted inside the vacuum vessel ➔ electronics must be high temperature compatible ($\leq 200^oC$.); • remote installation was required ➔ system needs to be designed to fit onto a divertor carrier; • items placed inside the vessel needed to be vacuum compatible with respect to outgassing. ➔ electronic on ceramic board; • robustness of the QMB unit with respect to electromagnetic and mechanical impact during the plasma discharge ➔ bonding technique for gold wires to quartz crystal.

After testing its reliability in the TEXTOR tokamak at Jülich (Germany) a QMB-unit was installed in front of the louvers of the JET inner divertor during the 2001 shutdown. Mechanical and electrical design features of the QMB and a protective actively controlled shutter will be presented. One task of the shutter (time resolution ~ 0.1 sec) is to protect quartz crystal and electronics against unacceptable power flux from the divertor and the other task is to pick up only selected phases of plasma discharges for exposure. Calibration measurements for the temperature and mass thickness dependence of the quartz crystal will be presented as well as the first measurements obtained during the Spring 2002 experimental Campaign.

[1] P. Coad et al., J. Nucl. Materials 290–293 (2001) 224.

E-11

RESULTS OF IRRADIATION TESTS OF KU-1 AND KS-4V SILICA GLASSES AS ITER CANDIDATE WINDOW MATERIALS

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To choose the suitable window material for the ITER optical diagnostics and to define the influence of the gamma and neutron irradiation on its properties the investigations in different countries were performed. Radiation-induced absorption and luminescence of sapphire, spinel, cerium and silica glasses with different hydroxyl content were studied. Results of irradiation tests have shown that KU-1 type of silica glass with largest content of hydroxyl (OH $\leq$ 1000 ppm) is the best candidate for material of ITER primary windows. This silica glass retains practically the initial optical properties after irradiation in nuclear reactor up to neutron fluence $F_{>0,1} = 6\times10^{19}$ n/cm^2 and attended gamma dose $D_\gamma \cong 2.5$ GGy in wide spectral region of 0.35 – 2.5 μm. However, the transparency degradation is very intensive in UV spectral region and permissible absorption dose is only about 10 kGy for the region of 240–350 nm. Therefore, more detailed *in situ* investigation of the KU-1 light transmission in UV region under neutron and gamma radiation and a search of new perspective window materials are urgent issues.

First investigations have shown that purified silica glass KS-4V with low hydroxyl content (OH < 0.1 ppm) is a perspective candidate for window material in UV spectral region. Comparative irradiation tests of KU-1 and KS-4V silica glasses in Co^{60} γ-source and in nuclear reactor have been undertaken. Radiation-induced absorption was measured after gamma irradiation of samples from 1 kGy up to 3000 kGy in the spectral region 0.19–1.0 μm at the temperature about 35°C. Maximum induced absorption is observed in the ultraviolet spectral region at λ = 215 nm. Values of induced absorption at this λ are varied from 2–4 times after irradiation up to D_γ = 2,9 MGy for KS-4V samples prepared from different ingots and the minimal induced absorption is about 0.17 cm^{-1}. Induced absorption of KU-1 silica glass at the same dose is about 3.5 cm^{-1} at λ = 215 nm. The short-life component of induced absorption was observed directly after irradiation up to $D_\gamma \sim 10$ kGy for KS-4V samples from some ingots. The influence of gamma–neutron ITER relevant irradiation on transparency degradation of KU-1 and KS-4V silica glasses has been modelled in nuclear reactor. Optical absorption spectral distribution after irradiation in nuclear reactor up to fluence about 10^{17} n/cm^2 were measured. The effect of additional γ-irradiation and irradiation temperature on radiation induced absorption were investigated. Transparency restoration of irradiated samples under thermal annealing was also study.

It may be concluded that after gamma irradiation the KS-4V silica glass has by order of magnitude lower induced absorption in UV range as compared with KU-1. However there is no considerable difference in transparency of KU-1 and KS-4V silica glasses after (n–γ)-irradiation up to $F_{>0,1} \sim 10^{17}$ n/ cm^2 (attended $D_\gamma \cong 3$ MGy) in nuclear reactor although induced absorption of KS-4V is some lower. Further *in situ* tests of transparency degradation under neutron and gamma irradiation of KU-1 and KS-4V silica glasses in UV range are necessary for ITER database preparing and final choice of window material at UV region.

E-12

FREQUENCY LASER DAMAGE OF Mo MIRRORS

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The first mirror (FM) is one of most critical elements of ITER's diagnostics. It must survive in an extreme environments and maintain a good optical performance. There are two main reasons of first mirror degradation in a fusion reactor: deposition of contaminants and sputtering by CXA. Besides the special precaution has to be attended to the FMs of laser diagnostics. Namely, because of high repetition frequency of laser shots the thin surface layer of mirror is subjected to short-term thermal impacts with a resulting effect very much similar to a fatigue deformation. Measurements of degradation of FM prototypes under pulsed radiation of YAG laser were made in Kurchatov Institute at the maximum number of repeated laser shots reached 40 thousands what just corresponds to one plasma pulse duration of ITER FEAT.

The laser tests were performed for First Mirror prototypes manufactured from mono and polycrystal Mo and with reflective Mo coatings on Mo substrates. The frequency-operated pulsed YAG laser (λ = 1.06 μm, τ = 12 ns, energy - 30 mJ, F = 12.5 Hz) was used in the experiments. The output beam of the laser operating in the TEM_{oo} mode has a Gaussian profile. Due to movable focusing lens the diameter of laser spot and, in that way, the laser energy density on the mirror surface can be changed. Data on laser damage thresholds were obtained for single laser shot and under multipulse laser influence.

At first, the single shot damage thresholds were measured and they were in the range from 2 J/cm^2 for polycrystal Mo to 3.5 J/cm^2 for Mo monocrystal mirrors. After that a long-term effect of frequency-operated YAG laser on mirror reflectivity was studied. At the latter tests an average laser energy density was equal to 1.5 J/cm^2 that about 2 times lower than the single shot damage threshold for monocrystal Mo. Nevertheless the degradation of mirror optical quality started after ~2 - 10 $\times$ 10^3 laser shots for different types of mirror. Monocrystal and Mo/Mo mirror were much more stable than polycrystal. In fact, monocrystal mirror did not degrade after than 10000 laser shots with 1.5 J/cm^2. The next step of this experimental test – to approach to the 1 million shot boundary at ITER relevant laser energy density and thus to shorten the range for approximation of obtained data to the laser shot number exceeding 100 million shots which will be necessary for the total time of LIDAR ITER FEAT operation.

E-13

DESIGN OF A NEW ELECTROMAGNETIC DIAGNOSTIC FOR RFX

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The design of a modified toroidal assembly for RFX, with a thinner shell and a system of coils for local field control, requires also the design of a new set of electromagnetic probes outside the vessel. Furthermore, typology and layout of the probes have been revised on the basis of the analysis carried out in previous experiments, and taking into account the foreseen scenarios of operation.

Magnetic field configuration in RFX is characterised by fast variations of all the three field components during the pulse, with relevant non axis-symmetries along the torus. Typical spectra exhibit modes up to n = 15 in toroidal direction and mainly m = 0 and m = 1 in poloidal direction. As a consequence, probe signals have a large dynamic (more than 60 dB), and extended frequency spectrum (several tens of kHz). Moreover, a large number of probes is required to correctly identify the complex spatial structure of the plasma column. An accurate calibration and a very low angular tolerance in probe mounting is required. To avoid shielding effects, probes must be installed inside the stabilising shell, which will be placed very close to the vessel; a very small room (less than 6 mm) is allowed for the probes. A further design specification for the sensors is due to the maximum operation temperature of the vacuum vessel (200°).

The probes used for the measurement of the global plasma parameters are rogowski coils, toroidal and poloidal loop flux coils; probe to pick up partial poloidal voltage are foreseen to allow the study of halo currents poloidal and toroidal spectra. All these probes are mounted on the external surface of the vacuum vessel or into existing grooves, and a very clever positioning has been necessary to minimise mechanical interference with the new shell supports. A set of 192 saddle loops, which cover the whole torus, will provide the feedback signals for the local coils control and the measurement of the radial field.

For the local measurement of toroidal and poloidal fields, a two axes probe has been developed, to be bolted on the internal surface of the new shell. Probe materials and construction techniques have been optimised in order to minimise coupling between the two coils and the dependence of the measurement from the temperature. Poloidal and toroidal distribution of the probes have been chosen carefully to avoid aliasing effects, taking into account not only plasma mode analysis, but also the periodicity of conductive structures in the torus. A number of 48 probes along toroidal direction and 4 along poloidal direction is foreseen, together with thickened sections in correspondence of important diagnostics or the shell gaps for a total of 234 two axes local probes. Extensive tests have been carried out on probe prototypes to characterise their electrical and thermal behaviour.

In the paper, the design of the new set of probes is presented, and the results of the tests performed on prototypes are reported.

E-14

INVESTIGATIONS ON THE DEGRADATION OF VISIBLE LASER DIODES UNDER GAMMA-RAY IRRADIATION

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The possible use of semiconductor optoelectronic devices (laser diodes, LEDs, various photodetectors) as component parts of remote sensing and diagnostics systems, in different robotic setups in nuclear equipment or for optical fiber-based communication led to the intensive investigations of these components as they operate in radiation environment. We carried out some research to evaluate the Gamma-Ray induced degradation of several semiconductor laser diodes, emitting at three visible wavelengths (635 nm, 650 nm, 670 nm), in order to make a preliminary evaluation of their possible use in fusion installations. The irradiation were done at the National Institute for Physics and Nuclear Engineering, Department of Applied Nuclear Physics, in Bucharest.

Low power (up to 7 mW) laser diodes were irradiated, and we investigated different parameters such as:

A. electrical characteristics: the voltage to current curve V(I); the thermal resistance (ΔT vs. heat generation rate); the threshold current temperature dependence $I_{thr}(T)$;
B. optoelectronic characteristics: the wavelength variation with the driving current $\lambda(I)$; the optical power vs. the driving current P_o (I); the optical power vs. the case temperature $P_o(T)$; the monitoring photodiode responsivity $I_{PD}(P_o)$;
C. optical characteristics: the temporal modification of the wavelength $\lambda(t)$; the wavelength dependency on the temperature $\lambda(T)$; the temporal stability of the optical power P_o (t); the beam quality (transversal mode structure, point stability, shape); the irradiation induced modification of the laser diode emitted spectrum.

Within this frame, accurate evaluation of laser beam parameters such as power, energy, wavelength, transversal/longitudinal mode structure is mandatory. To support this demand, we developed several set-ups for laser beam characterization under PC control, based on: laser power and energy meters, a laser beam analyzer, a wavelength meter, a laser diode driver/temperature controller, and a miniature, optical fiber-based spectrometer. Virtual instruments to control the above mentioned instrumentation were built up using the graphical programming environment LabVIEW, provided by National Instruments. Our main achievements on this subject are shortly described in the paper. The investigations underlined the degradation upon irradiation of the laser diodes external quantum efficiency, the increase of its resistance, an increase of its wavelength, as well as the change in the emitted spectrum. According to our knowledge, these results are among the few reports on the radiation induced changes in such semiconductor lasers.

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E-15

RADIATION RESISTANT ALTERNATIVE SUBSTRATES FOR ITER BOLOMETERS

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Present day JET type bolometers employ a thin mica substrate as the support for a gold meander electrical resistance bridge network. The use of mica is dictated by the commercial availability of very thin (= 25 μm) sheets with surface roughness in the range of 50 nm. These bolometers are being considered for use in ITER. However the radiation limit for such bolometers is expected to be about 10^{-2} dpa, due to swelling of the substrate, and also possible transmutation effects in the gold itself. In order to find an alternative to mica, the technical feasibility of producing sufficiently thin substrates from more radiation resistant materials with adequate thermal and dielectric properties has been addressed. Aluminium oxide, silicon dioxide, aluminium nitride, silicon nitride and diamond have been examined and found to be appropriate, with similar properties to those of mica.

As a first step, commercially available sheets of the different materials were thinned, by diamond grinding, down to a thickness of about 100 μm. Both sheet surfaces were then polished to a mirror finish, with a mean surface roughness better than 20 nm being obtained for the AlN, alumina, Si_3N_4 and silicon dioxide. The diamond samples were employed in the as-received condition, with a polished face of roughness better than 50 nm, and the as-growth surface with a roughness greater than 500 nm. In this way sheets of the different materials with sizes of about 15x15 mm^2 and thickness about 50 μm were obtained. Thin (about 200 nm) gold "U" shaped 1 mm wide tracks were then sputtered on to the prepared substrates. Thermal annealing at 473 K was used to enhance the gold track adhesion and stability.

These alternative bolometer prototypes were firstly characterized for their I–V behaviour. The sample track resistances, measured as a function of temperature in vacuum (~ 1 mbar) by a four contact method, showed the expected ohmic behaviour. Resistance values at 293 K were between about 5 and 20 ohms. This large variation is due to slight differences in the gold track thickness and different substrate surface roughness. However the extrapolated resistance values for real bolometer gold meanders indicates that the surfaces obtained are adequate.

The radiation resistance of the prototype bolometers is now being characterised. As a first step the behaviour as a function of temperature during gamma irradiation (^{60}Co) will be examined. A special irradiation chamber for the CIEMAT "Nayade" gamma pool facility has been constructed. This will permit the bolometer prototypes to be irradiated at about 10 Gy/s under controlled temperatures from 303 to 523 K, and the behaviour of the gold resistance track monitored. Following this first step, neutron irradiation tests must be performed.

E-16

ITER WIDE ANGLE VIEWING SYSTEM HEAD MIRRORS ASSEMBLIES (DESIGN & ANALYSIS RESULTS)

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The critical first component of the ITER wide angle viewing system [1] are the so-called head mirror assemblies. The main features of these assemblies are as following:

- an aspherical concave mirror viewing through a minute aperture in a solid flat mirror, thus offering significant protection of the mirror from erosion and deposition by the plasma;
- the primary incident beam may or may not be coincident with the main axis of the optical system;
- located close to the ITER first wall to acquire the required view of plasma and first wall.
- views in several directions from one blanket aperture location and both a mono-pupil and a multi-pupil module assembly is required.
- water cooled because of powerful plasma radiation and nuclear heating giving a problem of thermally induced the assemblies structural distortion.

A description of the design variants of the assemblies, including both mono- and multi-pupil solutions, is given in the report. Each assembly is designed as a rigid monoblock. Once set up the assembly does not require any adjusting elements for the mirrors. Problems of structural materials choice, as well as some technological items are considered. Results of the calculations of volumetric heating rate, radiation damage generation rate as well as analysis of the thermal and stress-strain status and the vibration properties are given.

A comprehensive thermo-mechanical analysis of the performance of the head mirror assemblies has been performed. The assembly strength with respect to thermal stresses, mirror alignment, validity of choice of structural materials, vibration effects etc. have been assessed. No fundamental problems have been revealed. The thermal distortions derived from the analysis are small and do not degrade the optical image.

The analysis indicates where local effects should be analyzed more comprehensively and a discussion of design improvements is given.

[1] J. P. Coad et al. Plasma Viewing in JET using Endoscopes and a Detailed Design for ITER. Diagnosatics for Experimental Thermonuclear Reactors (2) Plenum Press 1998.

E-17

STUDY OF RIEMF EFFECT UNDER DIFFERENT IRRADIATION CONDITIONS

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Introduction
The magnetic sensors (coils and loops), made of mineral insulated (MI) cables, are intended for ITER. In MI-cables the radiation initiates electric charge separation, emergence of the radiation induced electromotive force (RIEMF) and leakage current between cable strand and sheath. This effect may impact the magnetic measurements and other tokamak diagnostics using MI-cables.

Work description
The goals of this work were to determine the physical processes responsible for RIEMF effect in MI-cables and to develop methods for its minimization. In this study the set of MI-cables (with different size and strand materials) were irrudiated on the pulse reactor BARS-6 with dose rates from 10^3 Gy/s up to 10^5 Gy/s in pulse maximum. The dose rate dependences for several types of cables were obtained at different temperatures. The analysis of the obtained data and the transient electric processes under pulse action permits us to propose the model of radiation-induced separation of electric charge in MI-cables between the insulated central cable strand and the grounded metal sheath.

Conclusion
Based on this study, some recommendations concerning MI-cable for ITER magnetic sensors are given.

E-18

JAVA-BASED GAS INLET CONTROL SYSTEM FOR THE STELLARATOR TJ-II

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This paper describes the Gas Inlet Control System for the TJ-II Stellarator, which is currently in operation in Madrid (Spain). This system is an essential control element of the TJ-II, which has been designed, built and commissioned to provide the appropriate values of plasma density during the whole evolution of the experimental pulses. The gas is puffed using a set of 4 piezoelectric valves, which are driven by 4 pre-programmed voltage signals. Mainly aperture time and the desired amount of gas inside the vacuum vessel, define the signals profile. This amount depends on a complex function of the voltage applied to the valves and the accumulated effect of consecutive inlets during a shot. Four arbitrary waveform generators, composed by a set of VME modules, generate the output signals. The gas inlet control system synchronises the waveform generation with the TJ-II operation during the pulse sequence.

In order to do all this in a friendly environment and with access from different hardware platforms, a graphical user interface has been developed using the JAVA programming language; thus, all the parameters in the system can be fully configured using Internet applet viewers. This graphical application interfaces with the control software running on OS9 real time operating system. Specific sockets and a web server are used on the VME crate side. Remote access to the system is allowed from any computers in the TJ-II control Intranet.

All the software and hardware have been tested satisfactory during typical plasma shots. A time constant of 1–2 ms for the injection of hydrogen into the plasma has been obtained.

E-19

A RATING SYSTEM FOR POST PULSE DATA VALIDATION

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The aim of an automatic data validation system in a fusion experiment is to account – after every shot – for any occurrence of faulty sensors and unreliable measurements, thus preventing the proliferation of poor pulse data. In the past years a prototype has been successfully developed at FTU on a small set of density measurements.

The results have shown that the model can be further extended to plant and diagnostic data, and that the same system can be used to assign to raw data a quality factor, to be stored in the archive and to be used in the post-shot elaboration phase as a selection criterion. In this way, a data validation system can also provide data analysts with an useful tool to be used as a key – together with other significant parameters, like plasma current, or magnetic field – to search the archive for quality data.

This paper will describe how, using soft computing techniques, both these functions have been implemented on FTU, providing the users with a simple interface for fault detection developed in an open source environment (PHP-MySQL), to be finalized into the realization of an overall rating system for FTU data.

E-20

HARDWARE AND SOFTWARE UPGRADES TO DIII–D MAIN COMPUTER CONTROL SYSTEMS*

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The complexities of monitoring and controlling the various DIII–D tokamak systems have always required the aid of high-speed computer resources. Because of recent improvements in computing technology, the existing resources have become antiquated and cannot meet increasing demands. Also expected maintenance costs make it more feasible to replace the older hardware rather than continue their upkeep. State-of-the-art machines not only out perform the present systems, they can be purchased easily off-the-shelf from major distributors with standardized components. These newer systems also take advantage of more established and robust operating systems, i.e. Linux. With computer hardware upgrades in mind, a unique opportunity has presented itself: a major redesign of the high-level control software which manipulates the major subsystems of the DIII–D tokamak.

As expected, upgrading the corresponding computer software has become the more time consuming and expensive part of this upgrade. During this redesign, the main issues focused on making the most of existing in-house codes, speed with which the new system could be brought on-line, the ability to add new features/enhancements, ease of integration with all DIII–D systems, future portability and upgrades. The resulting blueprint has addressed all these issues and may well be used by others with similar constraints. As in the past, consideration was given first to third party commercial software. However, all the evaluated packages were removed from consideration after they failed to meet certain criteria. In all the potential upgrade systems, a key part of the DIII–D control software was the interface to CAMAC. A custom Linux software driver, mated with the Kinetic Systems 2115 serial highway driver, was created to meet this need. Other components of a control software packaged fell into place soon there after; I/O handling, real-time data storage, graphical user interfaces (GUI), procedural codes. Long term data storage is accomplished via a commercial relational database management system (RDMS), Microsoft's SQLserver. Graphical displays are created with the aid of Borland's Kylix development tools. The procedural codes were adapted from the previous system of rules and actions and then merged with the new I/O handling. This resulting system is modular and easily customizable to fit the needs of the target system.

The initial phases of this upgrade encompass the central tokamak control, neutral beam, and main data acquisition hardware/software systems. Other data acquisition systems and diagnostics are being considered.

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E-21

FLUORIDE GLASSES AS MATERIALS FOR RADIATION OPTICS

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Introduction

The principal advantage of fluoride glasses as compared to usual optical materials is their initial transparency in the wide range from 0.2 to 7.0 μm and stable transparency in the range from 0.4–6.0 μm after irradiation. The peculiarities of the fluoride glasses are the self-recovery of transparency and the recovery of transparency under the weak illumination after irradiation. Therefore, the fluoride glasses can be promising materials for a new direction of development of radiation resistant optical materials, namely, the materials with ability for self-recovery under irradiation.

Work description

The features of the production of the glasses based on zirconium, hafnium and aluminum fluorides are reviewed focusing to relationship with their nanostructure and optical properties. It has been shown that the shift of the fundamental absorption edge and the decrease of its slope depend on the cluster size d (2–8 nm), that is the main characteristic of a glass structure. The shift of the fundamental absorption edge is proportional to d^4. Gamma irradiated fluoride glasses recover their transparency due to illumination with the intensity from 1 to 3 mW/cm^2 and the wavelength from 400 to 1200 nm at room temperature. Significant increase of the recovery rate with increasing the wavelength of illumination from 400 to 600 nm is observed. Further increase of the wavelength of the illumination results in sharp decrease of the recovery rate. The recovery time depends on both illumination conditions and the cluster size d.

Conclusion

The recovery of the fluoride glasses during the illumination with the wavelength corresponding to the transparency range of the glasses is important for application of the fluoride glasses as radiation resistant optical elements.

E-22

OPTICAL NON-LINEARITY OF SILICA GLASS FIBERS UNDER INTENSE PULSED REACTOR IRRADIATION

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Introduction

Research and development of optical fibers exhibiting sufficient radiation stability and acceptable level of optical loss are actual due to prospective application of the optical fibers for diagnostics of plasma in fusion reactors at the dose rates of 10^4–10^5 Gy/s.

Work description

The light emission intensity and transient optical absorption at the wavelength of 400–750 nm in KU-1 silica core (OH content 1000 ppm) fiber waveguides under irradiation at BARS-6 pulsed fission reactor (pulse duration 80 μs, dose per pulse < $5.5 \cdot 10^{12}$ n/cm^2 (9 Gy), dose rate < $7 \cdot 10^{12}$ n/cm^{2}s ($1.1 \cdot 10^5$ Gy/s) have been measured. The intensity of radiation-induced light emission has been found to depend on intensity of probing light. Lower intensity of the light emission has been observed for higher intensity of probing light. The light emission quenching (down to complete suppression of light emission) occurs at the wavelengths shorter and longer than the wavelength of the probing light. Spectral and temporal characteristics of the radiation-induced light emission intensity and transient optical absorption are presented. The fast component of light emission observed during pulses is followed by a weaker tail of emission with the characteristic time ~150 ± 50 μs. Intensity of the fast component of light emission decreases (~5 times) on increasing wavelength in the range of 400 to 750 nm and obeys sublinear dose rate dependence with the exponent of ~0.7. The exponent and characteristic time do not depend on the wavelength.

Conclusion

Possible effects of electric charge partitioning and transfer, Cherenkov and thermal radiation on performance of optical waveguides are considered. Non-uniform ionization in the bulk of waveguides during reactor irradiation is assumed to be a possible cause of the observed quenching, sublinear dose rate dependence and constant characteristic time of light emission. Transient optical absorption may be explained by optical inhomogeneity (due to non-uniform ionization) of fibers during reactor irradiation. A model taking into account dependence of absorption coefficient on intensity of light in waveguides during irradiation has been introduced. Calculations conducted for various light intensity and dependencies of absorption coefficient on light intensity give both suppression and enhancement of light in fibers during irradiation. A possibility to control optical properties of waveguides during irradiation are discussed

E-23

CONTROL AND DATA ACQUISITION SYSTEM FOR MULTI-BARREL PELLET INJECTOR

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A control and data acquisition system has been developed for the MAST multi-barrel pellet injector. The main features of this system are pellet preparation with recording of relevant parameters, accurate pellet firing with easily configurable and recorded delays, and registering of fast signals associated with pellet injection - pellet speed, mass, and other signals.

The system is implemented on one PC running Windows NT using 4 embedded PCI cards. This configuration provides a reliable and cost effective solution. The Windows environment is convenient for fast program development.

The system software consists of two separate programs interacting by message interface. The pellet preparation program uses 32 channel analogue input and 64 channel digital input/output cards. The program is realised as a state machine. The number and duration of states are fully configurable, as well as program's action in every state. PC multimedia timers provide accurate timing. The program displays and records all parameters of pellet preparation in real time in physical values (pressures, temperatures). The program contains a pellet mimic, which shows the current state of the injector's devices and the pellet preparation sequence.

A second program is used to fire pellets and to record associated fast signals. The program uses one PCI card, which includes 4 channels, 20 MHz ADC and 24 digital I/O channels. The PC timer controls timing of the pellet firing. The firing error is less than 10 microseconds, which is adequate for this application. The program displays acquired data and calculates each pellet’s speed and mass; it uses effective data reduction – the acquisition lasts usually about one second, and only hundreds microsecond of pellet's signals are detected automatically and saved.

There are three modes of the system operation – manual control for injector testing, standalone, and the mode controlled by the MAST tokamak. All data are saved in the MAST format and transferred to a common archive.

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E-24

GAS JET ACTIVATION METHOD FOR A FUSION POWER MEASUREMENT ON ITER

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The major advantage of the neutron-activation method lies in the fact that neutron-activation measurements are by in no way affected by electromagnetic and radiation noise so the data obtained regarding neutron fields is accurate. Traditional variant relies on irradiation of specially selected activation foils placed at certain positions near the first wall. But this method provides no temporal resolution. Another activation system intended for ITER is based on measuring ^{16}N isotope activity. First experiments applying FSN neutron generator demonstrated time resolution of 50 ms at ~10 m/s water velocity and 9.4 m distance between irradiation and registration zones (J. Kaneko et all, 2001). Such resolution is not sufficient for measuring neutron field fluctuations (fusion power). Water flow velocity cannot be increased due to numerous technical problems caused by power consumption, hydrodynamic, cavitations, high pressure, mechanical strength etc.

We suggest a method for measuring fusion neutron flux based on applying gas jet activation. The general analysis of conditions suggests that higher temporal resolution requires larger velocity of activated matter. However at larger velocity the flow becomes turbulent. We are interested in Taylor's longitudinal diffusion rather than in transverse diffusion ($D_{\|} > D_{\perp}$) because the average cross-sectional concentration profile of forming radioactive matter is determined by both turbulent diffusion and transverse heterogeneity of flow. Gaseous matter is compressible and thus V changes along its flow axis. In order to determine the temporal resolution and sensitivity for the method of gas jet activation while jet is traveling in a pipe we have created a numerical code. The code describes the dynamics of the induced activity variation being subject to incident neutron flux exposure. We consider the activation of a subsonic gas jet. In this case the gas flow must be arranged so that the maximum velocity equal to sonic speed is realized after the registration zone has been passed. The distance H from the inlet to activation zone and from activation zone to registration zone makes 15 meters. In order to reach the sonic speed in a tube of 1 cm diameter after the detection zone has been passed gas must attain the speed of ≈ 72m/s. In addition gas velocity is ≈ 95 m/s within the activation zone. Thus the gas velocity along the path between activation and registration zones becomes ~3.6 times faster. Numerical calculation show, that temporal resolution does not exceed 10 ms, when activation and registration zones proportions are 0.5 m and 1.5 m respectively.

Result of numerical simulation has demonstrated that the temporal resolution of the method for gas jet activation being proposed is determined by gas velocity within the activation zone, length and diameter of the pipe, dimensions of activation and registration zones and has no dependence upon gas pressure. It should provide fusion power measurement for ITER conditions in the range from 2 to 500 MW with temporal resolution of 10 ms and 10% accuracy.

E-25

UPGRADE OF THE DIAGNOSTIC NEUTRAL BEAM INJECTOR FOR THE TCV TOKAMAK

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The diagnostic neutral beam injector (DNBI), manufactured by the Budker Institute of Nuclear Physics (BINP) Novosibirsk, was commissioned at the TCV Tokamak in 2000. The DNBI operational energy range was 20–50 keV, with an equivalent beam current of the full energy component of 0.5 A at 50 keV within a 10 cm diameter at the beam focus. The neutral particle beam fraction in full, 1/2 and 1/3 energy fractions was 42:37:21 % respectively delivered to the plasma. The beam is injected at a toroidal angle of 11.25^0 in the horizontal mid-plane. The plasma in DNBI source is produced in the RF driven plasma generator. Ions are extracted from the cylindrical RF plasma box and accelerated by a four grid ion optical system and the accelerated ions are neutralised by charge exchange in the gas flow from the plasma source. The neutral beam passes through a vacuum tank and into the TCV beam line. The vacuum tank houses a bending magnet that separates the residual ions from the neutral beam particles and an insertable segmented calorimeter which is used for measurements of the beam profile.

The main task of the DNBI is to provide together with charge exchange recombination spectroscopy (CXRS) for local measurements of plasma ion temperature, velocity and impurity density. Other applications, such as Motional Stark Effect (MSE) and Active Charge eXchange Neutral Particle Analysis diagnostics are under consideration for TCV. The first CXRS measurements of visible carbon CVI ($n = 8 \to 7$) transition at $\lambda = 5291$ Å, which were published in 2000, showed that the active CX to passive CX intensity was 25% for an average plasma density (n_e) of $1–2\times10^{19}$ m^{-3} and only ~5% at 5×10^{19} m^{-3} (ohmic TCV plasmas). In order to increase the active/passive and signal/noise ratios both the optical system and the DNBI were upgraded. The optimisation of the optical system was completed in 2001 permitting plasma measurements with n_e at 5×10^{19} m$^{-3.}$

A series of DNBI technical upgrades was undertaken to obtain a increase of the neutral beam current density in observation region which is required to extend the operational range of the CXRS diagnostic. The aim of DNBI upgrade, presented in this paper, is to increase the full energy beam current density in plasma by a factor of 2. An ion source was installed on DNBI with corresponding changes in power supplies. The diameter of RF cylindrical plasma box was increased from 10 to 12 cm, the extraction area was increased from 72 to 92 mm and the number of cylindrical apertures in the grids was increased from 163 to 241. The full energy (52 keV) beam fraction was increased from 0.5 to >1 A and the total extracted ion current – from 1.75 to 3.4 A. This paper also describes data from an optical system that measures the doppler-shifted H^{α} emission for the beam fraction measurements and the optimisation of the source RF plasma discharge to increase the full energy fraction and beam focal quality delivered to the plasma.

E-26

PRESENT AND PERSPECTIVE ROLES OF SOFT X-RAY TOMOGRAPHY IN TOKAMAK PLASMA POSITION MEASUREMENTS

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In ITER, the equilibrium position and shape control requires precise diagnostic information to meet the stringent positional accuracy. The long pulses and the conducting machine shells provide challenges at the very high frequencies used in the control system. Considerable effort has already been put into examining the problem of integrator drift for magnetic measurements of the plasma position and shape for pulses durations over 1000 seconds. In the present ITER plasma equilibrium control algorithm a two-component controller is used that separates the positional stabilisation (higher frequencies) from drift errors (lower frequencies). It is interesting to investigate whether a three-component controller where non-magnetic diagnostics could be used for the validation of the lowest frequencies. We have investigated this possibility using the TCV soft X-ray tomography system to determine its suitability for real-time position control.

The TCV tokamak is equipped with 10 pin-hole soft X-ray (SXR) cameras each with 20 channel photodiodes filtered by 47μm Beryllium foils. Tomographic inversion is used to determine the SXR emissivity using the minimum Fisher method on a rectangular grid [1]. The code package has been optimized with the aim of obtaining calculate-efficient and reliable reconstruction for present post-shot and perspective real-time control. The major speed-up was obtained by implementing a single reconstruction matrix with time averaged smoothing and pre-calculated smoothing would be required for real-time control. Test runs on phantom functions were used to verify the viability of this technique, which resulted in a near two order of magnitude reconstruction speed up over the more elaborate step-by-step approach. By defining the plasma position as the centre of gravity of the emissivity core, we were able to compute a SXR-position for all past TCV plasma discharges, for comparison with magnetic position data. Statistical regression was applied in matching the two sets of data. Minor systematic errors including a plasma shape dependence and a systematic drift of the magnetic axis were exposed. The standard deviation of the residual mismatch indicates that the common resolution limit of magnetic and SXR position measurements at TCV is 2 mm and 2.7 mm in the radial and vertical directions, respectively.

These results show that SXR on TCV can be used to measure the plasma position with a resolution comparable to the magnetic measurements. The algebraic simplicity and possibility of using parallel processing techniques (DSP) suggests real-time SXR tomography for use in real-time plasma position and shape control obviating the problems associated with purely magnetic feedback systems. These concepts will be discussed in more detail in the poster.

[1] M. Anton et al., Plasma Phys. Control. Fusion **38** (1996) 1849–1878.

E-27

DESIGN REQUIREMENTS FOR TORE SUPRA AND ITER DIAGNOSTIC WINDOWS

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Progress, both in technology and in physics, allowed the definition of more ambitious objectives concerning performances of in-vessel components. Indeed, the upgrade of Tore Supra (CIEL Project) foresees high power and high radiating plasma scenario during very long pulse operation (yielding radiated heat flux up to 0.3 MW/m^2 during up to 1000s). This long plasma operation times impose the improvement of diagnostic systems and the design of thermally resistant in-vessel components with reliable window assemblies.

The specific requirements and the methodology adopted to design for Tore Supra/CIEL (TS) high thermal loaded diagnostic windows are briefly summarized here:

- Technological constraints led to typical material choices based mainly on optical properties for different wavelength ranges.
- The shape, dimension and position of windows have to fulfill optical requirements in terms of visual solid angle or path lines across the plasma volume.

Results of thermal and mechanical finite element calculations (CASTEM 2000 code) of diagnostic windows under normal and off-normal thermal loads have been performed for TS and ITER using different materials and assembling methods. A non-linear transient elastic stress analysis associated with fatigue performance, has been systematically carried out taking into account recommended safety factors on ultimate thermo-mechanical material properties. This allowed the establishment of window operation requirements (look-up tables) in terms of material choice, geometry, assembling method and fatigue probability values, as a function of heat load deposition. This procedure has also been applied to an ITER sapphire-window assembly under accidental pressure and baking case. However, it should be remind, that window stresses depend also on bonding joint properties, associated creep effects and the assemblies are not stress-free due to entire fabrication procedure. Therefore, appropriate qualification tests are needed such as heat flux tests in IRIFA (Infrared Red Irradiation test Facility). This installation allows, if required, testing of window assemblies in vacuum under heat loads (0–1MW/m^2) and specified temperature gradients.

The test results obtained on a TS endoscope's head and on a crystal quartz-window assembly are reported and compared with finite element calculations, while ITER sapphire-window assemblies have been qualified following ITER test specifications for normal duty events and accumulated life.

E-30

REFURBISHMENT OF THE JET HALO CURRENT DIAGNOSTICS

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The refurbishment of the JET halo current diagnostics was much needed as most of the original probes have failed over the years. Originally there were two toroidal field pick-up coils at the top and the bottom of the vessel in two locations set diametrically 180^o apart. When the divertor structure was installed two additional toroidal field pick-up coils were incorporated. Now only one toroidal field pick-up coil is still working at the top and at the bottom. The refurbishment is aimed primarily to allow the monitoring of disruptions in future JET experimental campaigns. However, the new halo current diagnostics will also allow a better characterisation of halo current at JET. In fact, the new top set is favourably placed at the upper inner saddle coils and will be exposed for the whole duration of upward events. This new set could provide data on the peak poloidal halo current from three toroidal locations, while in the past at most two were available. In addition, an estimate of the minimum halo current density could be obtained from the Rogowski coils, these are employed to measure directly intercepted halo for the first time in JET. This set of diagnostics will be complemented by a further enhancement to be installed in 2004, as an improved understanding of halo current would help in the design of new plasma facing components, and is also important for ITER.

In order to minimise the impact on the in-vessel activities, the new pick-up coils and Rogowski coils were installed together with other modular based components during the 2001 shutdown, mainly carried out by Remote Handling (RH). The refurbishment includes two sets of diagnostics: one at the bottom (in the divertor) and one at the top of the vessel. The divertor set consists of two toroidal field pick-up coils, each embedded (hence transparent to RH) in a carrier of the septum replacement plate at octants 1 and 6. The other set is located at the inner top of the vessel, integrated with a new set of discrete wall protection plates. There are three subassemblies, each consisting of a toroidal field pick-up coil and a Rogowski coil embracing the plate support posts. They are located at the back of a protection plate, set diametrically 90^o apart (octants 3, 5 and 7), and can be fully installed or removed by RH (except for the existing connection to the feedthrough). Signals from the three toroidal locations are brought to a single feedthrough (octant 7) by a conduit running in the toroidal direction, which spans 180^o. This conduit has been designed to be RH compatible, but a few structural welds did require in-vessel man activity because they are not accessible otherwise. The toroidal field pick-up coils, relying on a well-established design and interpretation method, are used to estimate the total poloidal halo current; while the Rogowski coils are to determine the minimum halo current density.

The pick-up coils and the Rogowski coils have been installed during the 2001 shutdown. The first experimental results are available now, these will be also presented.

This work was performed under the European Fusion Development Agreement.

E-31

THE DATA ACQUISITION AND INTERLOCK SYSTEM FOR TORE SUPRA INFRARED IMAGING

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The data acquisition for the infrared measurement system on Tore Supra is a key element to insure the overall security of the CIEL project. This acquisition system will allow us to follow the thermal evolution of several components of Tore Supra, in particular the Toroïdal Pumped Limiter (LPT) and the additional heating launchers.

The objectives of the data acquisition system for Infrared Measurements are as follows:

- to implement and interface trip levels and hence insure the security of the in-vessel components,
- to accomplish a feedback control of the plasma and the additional heating systems to prevent formation of dangerous hot-spots,
- real-time recording and analysis of each viewed element, and subsequent further post-pulse analysis in order to understand the physics phenomenon.

When fully installed (a 3 year project), the infrared measurement system will be composed of 12 digital 16-bit infrared cameras which will give a complete view of the limiter and the five additional heating launchers. They cover a 100–1200°C temperature range, and each picture has a definition of 320x240 pixels (76800 measures) with a 20ms (50Hz) time resolution.

In this study, we will define the features being performed by the data acquisition system and the practical engineering implementation. In order to increase the security of the system, it has been decided to separate the two functions, acquisition/local storage and real time processing. Features include :

- transmission of the measurements using TAXI components through optical fibres,
- the numerical data acquisition and local storage using a PC network (one PC per camera) without compression storage onto local disks,
- real time on-board hardware calculation of minimal, maximal and average temperatures within user defined zones. The outputs have both direct security connections with defined trip levels, and shared data streaming (reflective “Scramnet” memory network) for feedback control of the plasma and different additional heating systems.
- Quasi-real time storage of these 3x12 temperature signals in the Tore-Supra data base (available for immediate viewing).
- Centralised storage of the compressed video data with access after the discharge.

One IR camera has been installed in 2001 and the acquisition system has been validated during the last Tore Supra campaign. Three cameras will be installed in 2002 and therefore three units will be operational. Because of its modularity, the full system will be installed as required, following the staged installation of the 12 cameras over the next three years.

E-32

DATA ACQUISITION AND CONTROL SYSTEM FOR A HEAVY WATER DETRITIATION PLANT

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It is well know the importance of detritiation of heavy water from CANDU type reactors, as well as the implication of detritiation in fusion processes and installations.

The nature of the fluids that are processed into detritiation heavy water requires the operation of the plant in maximum-security condition in order to protect the working staff and the environmental. The paper presents how we make the data acquisition and control for the plant. The plant for tritium and deuterium separation is an experimental project for extraction of the tritium from heavy water. This water is received from moderator of nuclear reactor that has primary radionuclid tritium. Radiological risk is because of tritium leakage that can be gas or vapors. The plant was design to operation without working staff in technological space. The purpose of the security and control system is to ensure for population an irradiation risk under the prescription limits and to obtain a lower possible risk.

In control room are digital computers that are used for plant control, alarm annunciation and data display. The digital computers communicate and manage all the other components from system using software applications (LabView platform). During normal operation, the data acquisition is done by a digital computer that provides signals from transmitters (flow, pressure, level, and temperature) and the measured parameters are recorded and displayed in the control room. The digital control computer display the safety parameter and has the following functions: to control automatically the process, to give commands in order to shut down and empty urgently the plant when the preset limits of the parameters are reached; the system gives commands to isolated whole plant and each section individually; it controls the interphony and telephony systems; it indicates the liquid level in tank collector; it commands the ventilation system; it controls the access and physical protection system.

The safety parameter display computer assists control room personnel in evaluating the safety status of the plant by providing a continuous indication of parameters or derived variables which are representative of that status. It is important to consider the experimental nature of the plant, and therefore all data equipment is flexible and easy to be adapted to the technological process. With this type of data acquisition system it can be assured the base for the measurement which is needed in most of tritium experiments. In case of industrial plant, dedicated equipment must be considered.

E-33

LOSS LESS DATA ACQUISITION SYSTEM FOR SST-1

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SST-1 being steady state Tokamak having a discharge duration of 1000 sec. demands a data acquisition system which is capable enough to acquire data loss lessly from slow diagnostics. The needs of loss less continuous acquisition has a significant effect on selection of acquisition hardware. The evolution of Data Acquisition System (DAS) for steady state operation of Tokamaks has been technology driven. The main issues considered in the design are the engineering aspects related with loss less data acquisition for long duration discharges. The paper describes the various needs of SST diagnostics and our selection of most suitable configuration for long duration discharges and discuss measure of performance achieved on various platforms. The loss less acquisition of slow channels for 1000 sec. generates a very large amount of data. Acquisition and handling of such large data poses unique needs on system hardware as well as monitoring during discharge and storing on a permanents hardware adds further constraints. For slow diagnostics the PXI based system with 48 channels was evaluated for continuous, single shot and windowed acquisition. In case of continuous acquisition it acquires data from 48 channel @ 10KHz sampling rate and pushes all the 48 data channels on network as well as writes to the hard disk simultaneously. A client utility also has been developed to view the data from any network node. It generate a data file of about 1Gbyte over a period of 1000 sec. The system was tried out on win98 and win2000 for performance test.

E-34

TELECONFERENCING FOR THE EFDA LABORATORIES

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With the introduction of the shared-facility approach to JET from 2000, tools for the remote participation of scientists in the experiment have become a necessity. Teleconferencing is one of the cornerstones of the Remote Participation infrastructure for JET under EFDA. With the opening up of more experiments towards joint scientific exploitation, and with the increased need for remote collaboration between design engineers, particularly for ITER and JET EP design and construction activities, teleconferencing between all laboratories has become all the more important. Teleconferencing needs also exist in the administrative and management areas, at least at some laboratories, like IPP Garching/Greifswald, where ISDN-based videoconferencing is in use since 1997, and especially within the EFDA bodies.

The 26 Associations of the European Fusion Programme now make significant use of teleconferencing techniques. Many technical and scientific meetings are now regularly organised as distributed meetings with audiences and speakers spread over several locations.
However, some of these Associations have built their teleconferencing infrastructures more or less independently and have thus adopted different technologies. While presentation (electronic slides) sharing has converged on a single approach based on the open-source VNC software, two technically different approaches are being used for video and audio. Several labs in the JET collaboration are using the VRVS infrastructure, originally developed by Caltech for the CERN LHC community, with the open-source Mbone tools VIC and RAT as clients for video and audio. Other labs have opted for an approach based on the H.320 & H.323 umbrella standards. In particular IPP Garching/Greifswald has invested heavily in this approach (after trying the Mbone tools without VRVS infrastructure in 1998/9) in the framework of the DFN Verein in order to provide high quality videoconferencing.

Both systems are based on the same basic protocols and Internet infrastructure, but they are not fully interoperable. Hence, an EFDA working group has been created with the task to explore and compare the two approaches, and to produce recommendations for the further investment in teleconferencing equipment at the EFDA labs.

The paper presents the functional requirements for teleconferencing facilities, taking into account all the different application scenarios that are expected in the European Fusion community. It describes the technical and cost aspects of the solutions that were taken into consideration; it describes the selection process and suggests methods for the convergence of these approaches, taking into account the rapid developments and progress expected in this area.

This work has been performed under the European Fusion Development Agreement

E-35

A "UNIVERSAL TIME" SYSTEM FOR ASDEX UPGRADE

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In many ways the concept of time is essential in fusion experiments. Technical operation requires proper protocolling, recording the absolute times at which actions occur or settings are changed. Physical measurements need a precise time measure in order to compare data acquired by the various diagnostic systems and to analyze plasma evolution. The current ASDEX Upgrade timer system distributes timer events that can be used for triggering but the connected devices have no direct access to the associated time. This makes time-correct processing of sensor signals a difficult task, especially if different sources are involved. To better deal with the measurement of time a new general concept has been developed. The idea is to make time a universal quantity to be measured and attached to sensor signals, command values and events. Based on this idea, an improved time system has been designed and integrated with the new distributed real-time plasma control and data acquisition system.

The universal experiment time is represented as a 64bit number and outlasts experiment lifetime with nanosecond resolution. This is sufficient for all physics applications. Time information is distributed optically and recovered locally via controller boards resident in the plasma control and diagnostic computers. Real-time applications may access the universal time at will, via the system host bus. In addition, functions have been added to link data acquisition and command output to the time information. Host software may specify time lists, resulting in external triggers. External triggers can be used to generate time list information (along with analog data sampled at the trigger point).

A variety of applications could benefit from such a universal time system, for example: intelligent real-time plasma control and data acquisition, steady-state experiment operation, spatially distributed systems, and data archiving.

This paper shows the structure and details the integration of the universal time system and discusses operational aspects.

E-36

AN ANALOG INTEGRATOR FOR THOUSAND SECONDS LONG PULSES IN TORE SUPRA

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Magnetic measurements provide one of the most important diagnostics to control the plasma localisation in tokamaks. The method generally used to perform these measurements in such an environment - high temperature, limited space, high-vacuum, radiation (n,γ) - is the association of a passive pick-up coil and an electronic integrator. However integrators are subject to intrinsic problems, the two most important being the integrator drift, which introduces an absolute error that increases with integration time, and secondly the saturation in case of disruption. For many years Tore Supra has acquired considerable experience in the development of analog integrators, and hence drift and saturation problems have now been minimised for several hundred seconds of pulse duration. In 2000, Tore Supra was stopped to allow the installation of the CIEL project components. We took advantage of this period to improve the performance of the existing integrator in order to satisfy conditions of 1000s or more pulse duration, without modifying the basic principles, which confer reliability and low cost.

In order to have a good plasma position control in Tore Supra (< 1 cm), the total integrator drift must remain less than 10 mV throughout the pulse. The new integrator version is still based on a differential input structure principle with two frontal integrators and on a dynamic auto-compensation feedback for the drift of each integrator, but it has been improved on several levels. Corrections have been made to the thermal drift of the auto compensation system and the drop of integration capacitors, special care has been put into the earth-screen layer design to reduce ground voltages. In addition, new improvements have been implemented to facilitate adjustment and calibration. The main results obtained in laboratory are :

- For a 1000s pulse duration, the new integrators have an average drift of 5 mV (vs. 18 mV for old integrators). The lowest drift encountered is 0.40 mV (vs. 0.91 mV) and no significant differences were found between tests at 25°C and tests at 40°C.
- The slew rate goes up to more than 10V / 50 µs (vs. 10 V / 500 µs) without significant output distortion. This characteristic is essential to obtain a high resolution in case of disruptions.
- The phase between input and output remains in a range of [$\pi/2$ and $\pi/2 + \varepsilon_\phi$], the phase error ε_ϕ being always lower than 3° (0.052 Rad).

Many supplementary tests have been carried out in order to qualify new integrators and during the 2001 campaign about 150 new integrators have successfully worked for several months, providing high quality magnetic measurements.

The main objective of the new integrators is to manage precise and reliable magnetic field measurements for long time pulses of up to 1000s. This objective has been surpassed, and today, we can reasonably hope to reach the ITER objective : drift less than a few mV per several thousand seconds. The paper underlines the main features of the new integrators, tests and obtained results as well as future development perspectives.

E-37

NATIONAL SPHERICAL TORUS EXPERIMENT REAL TIME PLASMA CONTROL DATA ACQUISITION HARDWARE

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Introduction

The National Spherical Torus Experiment (NSTX) is currently providing researchers data on low aspect-ratio toroidal plasmas. NSTX's Plasma Control System adjusts the firing angles of thyristor magnetic coil power supplies, in real time, to control plasma position, shape and density. A Data Acquisition system comprised of off-the-shelf and custom hardware provides the magnetic diagnostics data required in calculating firing angles. The NSTX Real Time Plasma Control System's Data Acquisition Hardware is an evolving system that will continue to evolve even though it is now entering its "final" implementation phase.

Description

The Real Time Plasma Control Data Acquisition Hardware is based on the VERSAmodule Eurocard (VME) bus. Components co-located in the same chassis utilize the high-speed (160 Mbytes/second) Front Panel Data Port (FPDP) parallel data link to transmit data between VME modules. For safety considerations data is transferred out of the NSTX test cell via a serial FPDP fiber optic link. Eight processors located on two real time computers provide 20 giga-FLOPs.

NSTX's investigation of non-inductive plasma start-up utilizes Coaxial Helicity injection (CHI). CHI involves creating a discharge by placing a low kilo volt potential across the Inner and Outer Vacuum Vessel walls that are insulted from each other. Test Cell magnetic diagnostics referenced to the Inner and Outer Vacuum Vessel walls must be isolated from each other. This creates a need for test cell Data Acquisition hardware referenced to two different potentials. There are three sources of data (Test Cell referenced to Inner Vacuum Vessel, Test Cell referenced to Outer Vacuum Vessel and Coil Currents referenced to local ground) that must be input to the real time computers. This creates a need for components with multiple FPDP ports that are not available off-the-shelf. While custom hardware was being designed, a less than ideal Data Acquisition configuration was implemented and is now in place. Currently being tested is an in-house designed 4 to 1 FPDP Multiplexing Module (FIMM) that will enable implementation of a "final" optimal configuration.

Conclusion

While there have been great advances in the availability of off-the-shelf components, there is and will likely remain the need to design and fabricate custom components. By the very nature of applied research there is no such thing as a "final" configuration for control, diagnostic and protection systems. The National Compact Stellarator Experiment (NCSX) is now in conceptual design. It is proposed that NCSX operate concurrently with NSTX utilizing the same Power Conversion and Plasma Control systems. Expansion flexibility, designed into the FIMM and "final" Data Acquisition Hardware configuration, will enable reconfiguration to support both NSTX and NCSX operations.

E-38

MDSPLUS DATA ACQUISITION IN RFX AND ITS INTEGRATION IN LEGACY SYSTEMS

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The current reconstruction of the RFX power supplies requires a re-engineering of the data acquisition system. The previous centralised approach using CAMAC front-end does not represent in fact the optimal choice for several reasons, among which the cost of CAMAC components and the limitations deriving from the exclusive use of the openVMS operating system. On the other side, our experience in the use of the MDSPlus data acquisition system has been quite positive and consequently the use of MDSplus has been retained, based however on a different architectural organisation. This has been possible as the system is available under many software platforms, including Windows and several flavours of UNIX.

The RFX data acquisition architecture defines now a set of compactPCI (CPCI) crates, each hosting a CPU running embedded Linux. Each CPU supervises local data acquisition, possibly performing data pre-elaboration to reduce, for example, the amount of data to be stored. Communication among distributed components is achieved using mdsip, the TCP/IP-based data communication of MDSplus.

This distributed approach allows also the integration of different components, such as legacy CAMAC based data acquisition systems and new diagnostic systems using Windows PC for data acquisition. Java-based graphical user interfaces running on Windows PCs are extensively used for providing waveform display and experiment configuration set-up.

More generally, subsystems using MDSPlus can be integrated also in other data acquisition systems. This has been done, for example, in a spectroscopic diagnostic currently used at FTU. The diagnostic has been developed and tested at RFX and can work as a standalone system, with its own pulse file stored in a Windows PC. The integration of this subsystem in the host data acquisition of FTU requires then the development of a thin interface layer. Such an approach looks promising for the development of diagnostic components as they can be easily moved across experiments, and is foreseen for other diagnostics to be developed at RFX and initially used in other experiments.

E-39

FIELDBUS FOR NEXT GENERATION OF FUSION DEVICES

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Most of devices used in presently operating fusion plants, like PLCs, started in the early '80.s. While Ethernet has rapidly become the standard for computer communications, different choices have been made by every laboratory about fieldbus in plant control. At present, very powerful network devices (as GigaBit Ethernet Switches) allow fast data transfer. Therefore, technological evolution gives the chance to adopt Ethernet for all the communication needs.

This paper will present the experience with Ethernet as fieldbus for monitoring and controlling FTU (Frascati Tokamak Upgrade) subplants. For example, the evolution of antenna temperature and gyrotron vacuum involved in LH (Lower Hybrid) process control during plasma experiments.

Particularly, some methods of getting data from Opto22 I/O modules will be described: reading from browsers; a XML-based technique for presentation into Excel; and a Java/PHP/ MySQL solution to import data into a database; all working in a multiplatform environment. A Java tool, developed as graphical user interface to meet operator requirements, will be also presented.

This work shows how this solution can cope with the needs regarding subplants remote control of next generation of fusion machines.

E-40

MECHANICAL BEHAVIOUR OF THE ITER TOROIDAL FIELD MODEL COIL (TFMC) IN THE SINGLE COIL TEST

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The ITER TFMC, built by the industry consortium AGAN (ACCEL, Alstom, Ansaldo, Noell) now is being tested in the TOSKA test facility of the Forschungszentrum Karlsruhe. The test is performed in two steps, TFMC as a single coil (phase I) and in the background field of the Euratom LCT coil (phase II). Phase I now is finished and the evaluation is in progress.

In order to observe the mechanical behaviour displacement transducers and strain gauges are fixed on the coil case and Inter Coil Structure (ICS). The displacement sensors allow the observation of the global coil extension, the extension of the joint leg and the coil displacement relative to the ICS. Some strain gauge rosettes detect the stress level of the coil. Uni-axial strain gauges give information about the coil case wall deformation in selected cross sections.

The tests are performed by current ramping up to 80 kA with and without steps and with ramp rate as parameter. In addition cycling tests were performed. All tests were done with continuous monitoring of strains and displacements.

The evaluation is concentrated on stress and strain values at maximum current, but of special interest is also the observation of nonlinearities and hysteresis effects of stress and strain versus Lorentz force which, e.g., may occur due to friction effects between winding pack and coil case. These values and curves are compared to predictions gained by FE analysis .

In phase I the mechanical behaviour of the coil submitted to in-plane loading only was tested. With one exception the sensors worked rather well. Due to the symmetric loading the behaviour at symmetric positions could be checked. It turned out that some symmetry disturbances occur in the strain behaviour. However, it still has to be clarified whether this is in the range of the sensor accuracy or due to some real coil geometry effects. The temperature compensation of the sensors proved to be limited. Therefore mainly such tests were evaluated where the temperature of the coil case was at steady state conditions.

The overall coil deformation and the stresses are rather close to the predictions. When operated, the racetrack shaped coil tends to enlarge itself and to become circular, which leads to an increase of its horizontal and a decrease of its vertical diameter. The hoop stress combined with in-plane bending results in a maximum tensile stress in the inner bore of the vertical axial plane. Some significant deviations in local strains seem to indicate deficiencies in the FE model or some local pre-stress due to manufacturing. The paper will give more details of the results and consequences for the phase II measurements.

F-1

ANALYSIS OF THE MEASUREMENT OF THE CURRENT SHARING TEMPERATURE IN THE ITER TF MODEL COIL

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The ITER Toroidal Field Model Coil (TFMC) is a racetrack coil consisting of 5 double pancakes, each with two joints at the inner and outer sides of the coil. The circular cable-in-conduit conductor uses Nb3Sn strands and a thin stainless steel jacket. The TFMC has been tested in the TOSKA facility of the Forschungszentrum Karslruhe (Phase I, single coil test).

The heat generated by the external heaters upstream of the inner joint propagates through the joint and downstream to the high field region of the conductor. To heat the TFMC up to the current sharing temperature (Tcs), the ramp heating was proposed by CRPP in 1999 as a compromise between two heating procedures: the heat slug injection and the steady state heating. In the TFMC there are no sensors inside the coil. Therefore the only possibility to assess the temperature in the coil is to use the measured inlet temperature (Tin) and to estimate the downstream propagation of the heat wave. However, in the run with ramp heating Tin is not a reliable figure to estimate the temperature in the coil at the location where the normal zone develops, because it overshoots at the end of the ramp. Our approach is then split into the following steps: (1) use of predictive simulations to assess the impact of the current sharing power, i.e. to investigate the voltage build up before the quench detection, (2) analysis of the experimental run using the results of the simulations, and (3) comparison of the above results with the expected current sharing temperature resulting from scaling of the strand data.

The analysis of the measurement of the current sharing temperature in the ITER TFMC has shown that: (1) the ramp heating leads to a quench in the winding (and not in the joint) of the TFMC, as predicted by the analysis; (2) a quantitative assessment of the Tcs from the test results of the 60 s ramp run (dT/dt ~ 6 K/min) tends to overestimate the cable performance compared to the strand performance, and this is likely due to heat exchange from the heated pancake into the coil through the radial plates; (3) as far as the cryoplant allows, a very slow ramp would be suitable to assess the conductor limits, avoiding the small transients of the multi step heating.

F-2

THERMAL-HYDRAULIC SIMULATION OF THE ITER NORMAL OPERATION

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A combined thermal-hydraulic simulation of the ITER magnet and cryogenic systems during normal operation is considered. Appropriate numerical models of different complexity have been developed and analysed for the past 4 years for the TF, CS and PF coils and their cryogenic interface using the comprehensive VINCENTA code developed at the Efremov Institute. The main attention of the work is focused on the effect of cryoplant load smoothing for the TF coil cooling performance. The original results are presented.

An active smoothing of the total pulsed heat load absorbed by the heat exchangers of the TF, CS and PF cooling circuits is supposed to be implemented for increasing the cryogenic efficiency of the cryoplant. An appropriate way of smoothing in ITER is a controlled bypassing of the helium flow through the heat exchanger of the TF coil structure cooling circuit. The control algorithm is based on measurable parameters such as a helium temperature and pressure before and after the coil/structure heat exchangers and a mass flow rate through them. For the present thermal-hydraulic analysis an advanced quasi-3D numerical model for the ITER magnet system with the main elements of cryogenic interface has been implemented in the VINCENTA code. The joint thermal-hydraulic model of the ITER magnet cooling system consists of three separate models for the TF, PF and SC coils. Since the PF an CS coil cooling circuits are not controlled, they previously simulated irrespective of the TF coil, but their loads on the cryoplant are taken into account. For a numerical simulation of the TF coil cooling a combined 1-D finite difference model for the transient compressed SHe flows and a 2-D finite difference model for the transient thermal conduction problem have been used. The TF coil is modeled with a set of 32 variable calculation cross sections equispaced in the poloidal direction. The total number of finite elements in all cross-sections is about 320,000. The cooling scheme provides for two separate cooling circuits for the TF winding pack and the TF coil structures. The TF case cooling circuit includes a centrifugal pump; a heat exchanger; control and bypass valves; return and supply cryolines; feeders; 50 cooling pipes to absorb the heat from the case walls; two return pipes with two choke tubes. The TF winding cooling circuit consists of a centrifugal pump; a heat exchanger; cryolines; feeders; 14 conductor channels for seven cable pancakes (individually for the cable annulus and cable central channels). So more than 80 channels are simulated simultaneously besides collectors and other elements.

Complex thermal-hydraulic behavior of the ITER coils has been analyzed under AC losses, nuclear and static heat loads distributed in space and time over the TF coil in accordance with reference scenario I for normal operation. Simulation result database allows building detail diagrams and maps of the transient thermal-hydraulic parameters for all ITER magnet and cooling system components modeled.

For the cooling layout and SHe flow rates the cryoplant smoothing procedure has no significant impact on the TF conductor temperature in the critical point of the TF winding. The obtained data were used for the ITER design at the stage of the ITER EDA.

F-3

COOL-DOWN SIMULATIONS FOR THE ITER

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A comprehensive cool-down simulation of the ITER magnets should be provided for the proper engineering of the cooling system and cool-down scenarios. Such simulation, as known, requires a lot of computation resources. An appropriate finite element model of the TF coil is proposed for analysis with the use of COND code developed at the Efremov Institute. The main attention of the work is focused on the cooling of the TF coil as the most massive component of the magnet system. The original results are presented. A cool-down procedure of the ITER machine is mainly determined by cool-down of the magnet system. The total mass of the magnet system is approximately 8732 t. The total helium flow that is available for cool-down of the magnet system is 3 kg/s. It is subdivided between different components of the magnet system in an approximate proportion to the mass of each component. The cool-down of the magnet system is performed in two stages, from 300K to 80K and from 80K to 4.5K. The cool-down scenario of the ITER machine can be quantified on the basis of a cool-down analysis of the TF winding packs and their cases as they have the largest cold mass and the most restrictive hydraulic parameters. The PF coils and the CS will follow the same gradual cool-down scenario as that for the TF coils, because all portions of the magnet system must be simultaneously cooled to keep them in thermal equilibrium as closely as possible.

The TF case contains a set of cooling channels. These channels are embedded in the walls of the TF case and connected in parallel by helium flow. A half of these channels is used for cooling the inner leg portion of the TF case and another half for cooling the outer leg of the TF case. The rate of the gradual decrease of the helium temperature at the inlet of the TF case is 0.5 K per hour. The gradual cool-down of the magnet system is provided by compressed gaseous helium pre-cooled by liquid nitrogen. It is roughly evaluated that the temperature gradient of 50K satisfies mechanical limits for the magnet system during the cool-down.

The paper is dedicated to a numerical thermal-hydraulic analysis of cool-down from 300K to 80K for the TF case. The model takes into account the actual complex structure system of the TFC case and includes the pre-compression structures, inter-coil structures, supports for six PF coils and gravity support. The cool-down model allows evaluation of evolution of the temperature gradients both along the cooling channels and over the thickness of the TF case and the height of the structures. Two possible design options for cooling the structures are under analysis: 1) passive cooling of the structure system (no cooling channels for any structural component) and 2) a part of the structures has the cooling tubes for cool-down. The present analysis is focused on admissibility of a passive cool-down for the structures of the TF case.

Simulation results show the unacceptable temperature gradients for majority of the structures. Nevertheless some structures can be passively cooled. An additional cooling circuit should be provided for other structures for their active cooling.

F-4

ITER TOROIDAL FIELD MODEL COIL TEST: ANALYSIS OF HEAT TRANSFER FROM PLATES TO CONDUCTORS

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The first phase of the tests of the International Thermonuclear Experimental Reactor (ITER) Toroidal Field Model Coil (TFMC) took place in 2001 in the TOSKA facility at Forschungszentrum Karlsruhe (Germany). As the main objective of this coil is to demonstrate the industrial feasibility of the ITER TF coils, the TFMC design includes the particular features of these coils, such as the insertion of the conductors inside stainless steel radial plates (see Figure). During ITER transients such as the safety discharge of the magnet, eddy currents are induced in the plates, generating a heat power which is transferred into the conductor by thermal diffusion through the conductor insulation, and eventually quenches the system. The delay introduced by this diffusion process influences significantly the development of the quench which occurs in the ITER coils at the end of the discharge. The coil current at which this quench is triggered is a key parameter for the design of the system.

The thermal diffusion model is presented. The diffusion equation is added to the thermo-hydraulics code Gandalf, which enables to take into account the helium circulation through the conductor channels. For a given transient, the time and space evolution of the conductor temperature can be calculated in this way.

The ITER TFMC tests offered a unique possibility to check this model especially during the 4 safety discharges carried out during the system operation. The discharges were intentionally initiated from a low 25 kA current to keep a stable thermohydraulic regime in the circuits. The results of these experiments, compared to the predictions prove that the model is relevant and can be used for modelling the ITER TF coil behaviour. The assumptions made in the model for heat conductivity and specific heat are discussed.

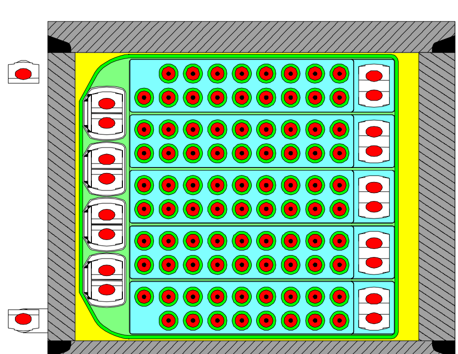

Figure. TFMC cross-section showing circular conductors embedded in the radial plates.

F-5

FRICTION ANALYSIS OF THE MECHANICAL BEHAVIOUR OF THE TORE SUPRA TOROIDAL FIELD MAGNET

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The Tore Supra Toroidal Field (TF) magnet is made of 18 superconducting NbTi coils bolted together to form a mechanically rigid torus, sustaining itself the electromagnetic loads during operation. Each coil is made of a double pancake wound winding-pack, operated at 1.8 K and enclosed inside a 316LN thick case, operated at 4.5 K. This magnet, operated at Cadarache since 1988 is mechanically monitored with strain gauges installed on the case of 12 coils.

The comparison of the experimental results given by the strain gauges to the finite element analysis performed when the magnet was put into operation showed a non-linear mechanical behaviour of the coils, which couldn't been modelled with the linear elastic analysis carried out. The non-linear phenomenon was characterized by a strongly different path for the evolution of certain strains versus the intensity of the current in the magnet from ramping-up to ramping down. This behaviour was explained by the influence of friction between the winding-pack and the case, but couldn't be modelled properly by the finite element code used, CASTEM. The development by CEA in the last ten years of a new finite element code, CASTEM 2000, including now the possibility to model friction between surfaces in contact with each other, allows to build a more refined model of the Tore Supra TF magnet. This model is made with brick and shell elements, in a similar way to the model previously used, but takes into account the possibility of friction between the winding-pack and the case. The winding-pack is modelled with an orthotropic equivalent material featuring the different elasticity according to the direction, radial, axial or longitudinal. The electromagnetic loads are computed directly at the nodes of the mesh through a semi-analytical application of the Biot and Savart law and integrated on the whole winding-pack. To address properly the friction phenomenon requires the knowledge of the load history at each point where friction occurs. The different steps of the coil manufacture, magnet assembly and cooldown have thus been modelled to feature as much as possible the initial stress pattern when the current is ramped up in the magnet. A parametric analysis is carried out to investigate the influence of the friction coefficient on the variation at the strain gauge location of the strains and stresses versus the loading. A comparison to the experimental results from the strain gauges allows to determine the friction coefficient range giving the best agreement with the measurements.
The paper describes the model and the analyses performed, including the comparison with the experimental results and a discussion on the validity of the study.

The friction analysis of the Tore Supra TF magnet demonstrates that the mechanical behaviour of a large superconducting magnet during operation can be well addressed by the now available finite element models.

F-6

TESTING OF THE CICC JOINTS FOR SUPER-CONDUCTING MAGNETS IN SST-1

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The basic conductor for all the superconducting TF and PF magnets in SST-1 is NbTi/Cu based multifilamentary Cable in Conduit Conductor (CICC) [1, 2]. All the 16 TF superconducting magnets are double pancake wound and have 80 number of interpancake joints along with 15 intercoil joints. Four of the nine PF superconducting magnets are double pancake wound and have 12 number of interpancake joints. The joint configuration is soldered shake hand type and similar for all the joints. The estimated joint resistance is 2.0 nΩ at 10 KA current and 4.5 K temperature. The maximum temperature rise of the joint in the worst case of the plasma disruption of 330 KA is estimated as 0.4K. In the first campaign, the full size prototype CICC joint has been tested at 4.2 K and 13.5 KA. The current decay method has been adopted to measure the DC resistance of the joint [3]. In this test a high current is induced in a closed joint loop by rapidly discharging a primary superconducting magnet coupled with joint loop. The joint loop current decays with a time constant governed by joint resistance and joint loop inductance, which is measured by Hall sensors. The measurement of current decay time constant and loop inductance finally gives the joint resistance. In the second campaign, the current transfer pattern in the joint, the stability of the joint in parallel magnetic field disturbances, and the DC resistance of the joint will be tested. The cryogenic test facility for these tests has been developed in house at IPR. The paper will describe the joint design, test methodology, test facility, and the test results.

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F-7

ANALYSIS ON NUCLEAR HEATING IN THE SUPERCONDUCTING COILS OF HT-7U TOKAMAK FUSION DEVICE

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HT-7U is a completely superconducting tokamak experimental device under construction at Institute of Plasma Physics, Hefei, China. The tokamak will generate a neutron rate of the order of 10^{15} per second during the D-D operation of long pulse up to 1000 seconds. It is necessary to have the exact nuclear heating source in the superconducting magnet for the purpose of the design of cryogenic system and the quench analysis. Nuclear heating in the magnet mainly comes from D-D fusion neutron energy deposition and induced Gamma energy deposition. Based on a peak D-D fusion rate of about $2x10^{15}$ D-D and 3% D-T neutrons per second, the nuclear heating density in the TF (Toroidal Field) and PF (Poloidal Field) winding and the coil cases have been estimated by using the Monte Carlo particle transport code MCNP and the latest version of the Fusion Evaluated Nuclear Data Library (FENDL-2) based on the three-dimensional geometrical configuration where the first wall, vacuum vessel, inner and outer liquid nitrogen radiation shield, cryostat, poloidal field coils (PF) and toroidal coils (TF) are included. The calculations show that nuclear heating in the coils is contributed some 10% by neutron energy deposition and some 90% by induced gamma energy deposition. To reduce nuclear heating in the superconducting magnet of TF and PF (on another hand, to reduce neutron/gamma radiation dose in the environment), a shield water envelope (with double walls) is designed with the water layer of thickness of 5 cm. The boron concentrations in borated water and the enrichment of ^{10}B have been optimized considering the requirement of shielding, limit of available space, limits of boron solubility and erosion problem etc.

F-8

THE GROUND FAULTS DETECTION SYSTEM FOR THE TORE SUPRA TOROIDAL PUMP LIMITER

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The Toroidal Pump Limiter (TPL) of Tore Supra is electrically insulated from the vessel by ceramic rings, to allow its polarization at a voltage of up to 1 kV. The TPL, in its start-up version, was composed by six sectors, insulated from each-other and from the vessel. Each sector was connected outside the vessel to the machine through a 1 Ohm resistor to fix its potential and to measure the currents collected from the plasma. These measurements have been used to validate the physical model of current distribution in the scrape off layer.

The continuous monitoring of the TPL isolation has been considered necessary due to the large complexity and extension of the TPL structure, the many insulating joints, often impossible to inspect, and the small insulating distances. In case of a single or multiple short-circuits from the TPL to the vessel, large Halo or induced currents might flow in the fault, putting in danger the machine integrity and the TPL insulation.

The TPL Ground Faults Detection System has been designed in collaboration with the Consorzio RFX using a multiple-channel structure similar to the one used for RFX machine ground loop detection system. The TPL Ground Faults Detection System is composed by an high frequency voltage generator board that feeds several transformers through a multiplexing card. The transformers are installed in each earth connection of each TPL sector. If a ground loop is present, current circulates in the earth connection and is detected by a second transformer. A detection board is connected to each detection transformer in turn, through a second multiplexing card. The system is synchronized and supervised by a local Programmable Logic Controller. It can measure earth fault impedances comprised between 10 and 4000 Ohms, fault detection time is less than 3 seconds and impedance measurement error is less than 5%. The status of the system and of the TPL insulation can be remotely monitored from the Main Control Room. Self diagnosis circuitry has been implemented to improve the system reliability.

In 2001, during the Tore Supra experimental campaign, the TPL Ground Faults Detection System has correctly detected twice the loss of TPL insulation caused by major plasma disruptions. In 2002 the development of a second multi-channel system, integrated with the first one, is foreseen. The new system will improve the existing monitoring of the ground loops of the Tore Supra Machine: the vessel and the mechanical structure, the magnetic circuit and all diagnostics and auxiliary systems linked to the machine potential.

The paper describes the design features, the performance, the operational experience and the future developments of the TPL Ground Faults Detection System and the progresses in the design and commissioning of the new Machine Ground Faults Detection System for Tore Supra.

F-9

IMPROVED SUPPORT CONCEPT FOR THE HELIAS REACTOR COIL SYSTEM

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Coil systems with four toroidal field periods for the Helias Reactor (HSR) are considered. They are more compact than the former five periodic systems. The major radius is 18m at an average coil radius of about 5m. The magnetic field on axis is chosen to be 5T, resulting in a total stored magnetic energy of less than 100GJ. The thermal output of the Helias reactor amounts to about 3000MW. The magnetic field has been optimized in view of plasma confinement and ignition. The reduction from 5 to 4 field periods and the related reduction in size will also reduce the cost of the Helias reactor.

The HSR coil system comprises 40 modular coils with NbTi-superconductig cables. The operational current of the conductor is 40kA. The magnetic field at the coils has been reduced by using a trapezoidal shape for the coil cross-section and splitting the winding pack into 8 double-pancakes with a total number of 18 x 2 x 8 = 288 turns per coil. This diminishes the overall current density at the location of high magnetic field. Consequently, the maximum magnetic field at the coils is about 10T, which is in the range of NbTi-technology at a cooling temperature of 1.8K. The coil winding pack of each individual coil is surrounded by a strong coil housing of stainless steel. This is necessary to maintain of the large value of the virial stress which characterizes the specific magnetic load of the coil system. The weight of the winding pack of one coil is 45t, and the total weight including the coil housing and all necessary reinforcements amounts up to 200t.

The electromagnetic field and force distribution of the coil system and the net coil forces of the individual coils are calculated. In order to balance the large magnetic forces the common vault support concept of earlier studies has been modified: Two toroidal support rings are introduced at the inner part of the coil system. These new rings form the basic part of the vault support. They connect the coils and carry the main part of the centripetal forces. In this way decoupling of the system's vault support and of the adjacent support of the individual coils is achieved. The mechanical stress and strain distributions of half a field period are computed, using the ANSYS finite element code and taking the stellarator symmetry into account. The values are analyzed for the coil housing, for the intercoil structure, and for the toroidal support rings. Nonlinear calculations are done in order to investigate the embedding of the pancakes of the winding pack inside the coil housing.

F-10

ANALYSIS OF A POSTULATED UNMITIGATED QUENCH IN A POLOIDAL FIELD COIL OF ITER-FEAT

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The unmitigated quench is one of the most severe accidents for the superconducting coils of ITER-FEAT. In the course of such an accident it is assumed that the redundant and diverse techniques to detect a quench and to safely discharge a coil fail at once, i.e. the probability of such an event is practically zero. Nevertheless, in the context of identification if the maximum possible damage to the coils of ITER could exert to the barriers of radioactivity, such unlikely events are investigated.

The coil selected for the present analysis is the PF3 coil, the largest out of the poloidal field coils. To maximize the initial stored energy and so the damage, the maximum operating current for this coil is taken as initial condition. This current is additionally increased by 30% to simulate induced energy contributions from the other poloidal field coils due to their possible current changes. The location of the initiating quench is assumed in the innermost turn of the central double pancake of the PF3 coil.

The analysis is done with the code system MAGS. It takes into account the transient magnet field, the electrical circuit of the coil and the inductively coupled spacer plates between double pancakes. The thermal model considers one double pancake of the PF3 coil, wound with two conductors in hand. Helium flow is not analyzed, a simple analytical formula to determine quench propagation along the cable is used instead. Quench however can not only propagate along the cable, transversal propagation is also determined via 3d heat conduction and module OHM, identifying conductor quench due to field, temperature and current density. In case of insulation failure short circuits in the coil are analyzed with module COINLOSS.

The analysis shows that the cable space near the quench initiating spot reaches melting temperature of copper at about 30 s after the start of the transient. The thick walled jacket surpasses also the copper melting temperature, however, it stays well below the melting temperature of steel. At about 150 s the current in the coil approaches zero and the copper melt starts to solidify. At 400 s all copper is again solid.

F-11

SUSPENSION OF THE W7-X COILS

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A large steady state plasma fusion experimental device WENDELSTEIN-7X (W7-X) of the stellarator type is presently being built at the Max-Planck-Institute for Plasmaphysic (IPP). The aim of the experiment is to prove the reactor validity of the stellarator research line in addition to the tokamak research line. The inherent property of the stellarator is a true steady state operation which is of major importance for fusion reactors. The stellarator W7-X is characterised by special shaped nonplanar superconducting magnet coils which are assembled to a toroidal arrangement, to the core of the device. Main features of the magnet system are: Modular concept with 5 field periods by 5 different nonplanar field coils and 2 ancillary planar coils per period. The major radius of the experimental device amounts approximately 5.5 m, the mean coil radius 1.1 m, the magnetic induction on plasma axis 3.0 T, the maximum induction at the coils 6.0 T, stored magnetic energy 620 MWs, operation temperature 4.0 K, and total weight about 400 tones.

The magnet coil system of the main and ancillary coils is carried by a central support structure and kept in strongly predefined space position. The central support structure together with the additional lateral stiffening elements between the coil housings generate a complex 3D-framework system. This system is responsible to balance the considerable electromagnetic force typically in the range of 1 to 4 MN for the single coils and 20 MN per period as residual force in the centripetal direction.

The analysis and design of the central support structure is a long term task with considerable man-power quota. The milestones to develop the support structures have been published in several journals. The process of the general design definition of the central support structure is completed, the particular components have been already ordered and are in the process to be manufactured. This paper shows some special investigations of vital components inside the structure. Especially the central support elements, so-called extensions which connect the coil housings with the support ring, have to carry very high forces and moments.

These elements of the structure consist of two parts which have to be joined in the state of assembly. The aim of the investigations, described in this paper, is to explain the coherences between particular elements inside the structure. As a consequence, the design for connecting elements under high stresses is recommended. In addition, the methods and possibilities of the assemblage are discussed in this paper.

F-12

PROTECTION SYSTEM FOR THE SUPERCONDUCTING COILS IN WENDELSTEIN 7-X

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One of the goals of W7-X is to demonstrate steady state operation which is an inherent property of stellarators and essential for a future fusion power plant. Magnetic confinement in W7-X will be achieved by 50 non-planar and 20 planar coils with diameters of up to 5 m and masses of typically 6 t. To achieve steady state operation, all coils are superconducting.

The main field of W7-X is produced by 5 groups of 10 non-planar coils each. In addition 2 groups of 10 planar coils each allow to vary the magnetic configuration. The electrical parameters of all groups of coils are quite similar with inductance being typically 1.1 H res. 0.4 H.

The big amount of energy (up to 1 GJ) stored in the coils requires a fast and reliable protection system. In case of a quench or the need for a rapid shut-down the current in the coils has to be commuted to external resistors and ramped down within 10 s. Several technical alternatives have been studied to find the most reliable and economic solution.

All main components of the protection system were qualified by tests prior to be implemented in the design. This holds in particular for the power switches where currents up to 20 kA resp. voltages up to 8 kV have to be switched within a fraction of a second. In addition operation of the components was verified in a magnetic environment of up to 80 mT to ensure proper function in the vicinity of the stellarator and the high current lines.

The presentation will describe the results of the tests of the main components and the final design of the coil protection system.

F-13

CRYOTECHNOLOGY FOR WENDELSTEIN 7-X

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The magnet system of the stellarator WENDELSTEIN 7-X (W7-X) consists of 50 non-planar (npl) and 20 planar (pl) superconducting coils which are arranged toroidally with a mean large diameter of 11 m. All coil windings and the interconnecting bus system are composed of NbTi/Cu cable-in-conduit-conductors with Al-alloy jackets. The winding packs are retained by steel casings which in turn are shored up against each other, and which are attached to the ring-shaped coil support structure. The coils are combined in 5 and 2 groups of 10 series-connected npl and pl coils each, respectively. Every one of the 7 groups can be supplied independently with currents of ≤18 kA via a corresponding current lead pair.

This whole assembly has to be cooled down to and operated at ≈4 K. In order to make this possible, the cold components are situated within a cryostat where they are thermally insulated by high vacuum, an actively cooled thermal shield, and multi-layer insulation. Shield and insulation layers cover the cryostat walls which are composed of the plasma vessel, the outer vessel, and the port ducts which allow access to the interior of the plasma vessel.

In addition to the cold components associated with the magnet system, also the cryo-vacuum pumps of the divertor have to be supplied with refrigerant at about the same temperature as that of the coils.

After terminating the R&D works, and after completion of the cold component design phase, the cooling requirements for all cold components could be finally determined. In particular, the coil casing cooling was significantly influenced by experiences gained from the DEMO CRYOSTAT and DEMO COIL projects. Design of the He-manifolds and quench gas piping was governed by extreme space restrictions and mounting requirements. When laying out the cryogenic system, special emphasis was placed on high cooling efficiency during the long zero field periods of W7-X which amount up to 90% in the years' average. This requirement determined especially the design of the current leads which will be of conventional, but "overloaded" type. Another design feature is the possibility of day-night refrigeration power levelling: Liquid helium, produced by utilising excess refrigeration capacity during nights, will be used for cooling the current leads, and for supporting the refrigeration capacity in maximal power modes. Another reserve power booster will be liquid nitrogen to support the refrigerator by precooling via heat exchangers - a technique which will be used anyway during W7-X cool-down. All these features allow to limit the refrigerator capacity to ≈5 kW at 4.5 K entropy equivalent.

The cooling requirements for all expected W7-X operation modes, the design and layout of the corresponding magnet system cooling circuits including the quench gas relief lines, the cryogenic instrumentation, and the refrigeration system will be presented.

F-14

THE ITER THERMAL SHIELDS FOR THE MAGNET SYSTEM: SPECIFIC DESIGN, ASSEMBLY AND STRUCTURAL ISSUES

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The ITER thermal shield system minimises heat loads transferred by thermal radiation and conduction from warm components to the components and structures that operate at 4.5K. Reduction of these heat loads by over two orders of magnitude facilitates the removal of the residual heat load at 4.5K by the ITER cryoplant with reasonable capacity. Thermal radiation to the superconducting magnets is minimised by operating the thermal shields in a temperature range of 80–100K and providing surfaces with low emissivity by silver coating. Conduction through thermal shield supports is minimised due to their relatively small conductance.

The thermal shields comprise the vacuum vessel thermal shield (VVTS), which for space reasons has to closely follow the shape of the vacuum vessel (VV) and VV ports. It is therefore of a segmented, toroidal design. Its complexity, apart from the overall shape, lies in the fact that the design of the structure should be compatible with overall tokamak assembly and repair procedures, the required number of electrical breaks, its surface coating, the narrow clearances with adjacent components, the high requirements for reliability for the lifetime of the ITER machine. This paper presents assembly procedure and the design of unique bolted joints and support system, which fully meet all design requirements specified for the VVTS.

The structure of the ITER neutron beam (NB) VV ports located in three adjacent sectors of the machine is completely different from that of “regular” VV ports. The toroidal inclination of the port aggravates design problems peculiar to the VVTS around regular ports. The paper describes the specific structural design and assembly aspects of the thermal shield for the NB ports.

The efficiency of the thermal shield system depends strongly on the interface between its components. The paper presents the results of the detailed study of this issue.

The ITER thermal shields rely on a low emissivity for the whole time that the magnets and thermal shields are cold. Relatively thin solid layers of cryocondensed gas on the reflective surfaces can increase the apparent emissivity beyond acceptable limits. The paper describes critical aspects of ITER vacuum and cooldown scenarios as well as results of 3D supporting cryovacuum calculations.

The paper describes details of the design and assembly procedure for the VVTS, present some results of cryovacuum, thermal-hydraulic, seismic and structural analyses and also include reasons for adopting or rejecting design and assembly concepts.

F-15

THE DESIGN OF 192 SADDLE COILS FOR RFX

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The torus assembly of the RFX machine is presently being modified in view of the experimentation of the active control of the plasma MHD modes. The thick stabilizing shell has been removed and the machine will be equipped with a system of 48x4 Saddle Coils, specifically conceived for the control and the active stabilization of the typical Reversed Field Pinch (RFP) plasma modes.

The system will be driven by fast switching power supplies and is designed to produce a stationary or time-varying magnetic field configurations with a spectrum of toroidal mode order up to (m=1, n=20) and with an amplitude up to 40 mT plasma edge (for mode m=1, n=8). The nominal current on each coil is 400 A, the maximum operating voltage is 600 V and the operating frequency band is DC to 200 Hz. The Saddle coil will be operated either as a feed-back controlled system, for active plasma mode stabilization, or as a feed-forward system, in order to obtain a specified single- or multiple-helicity magnetic boundary. In any case the active system will be backed-up by a thin conductive shell (passive stabilizer), whose time constant for the penetration of the vertical field (for mode order m=1, n=0) will be ~50 ms , i.e. 1/10th of the time constant of the original shell.

Since original stabilizing shell also had the purpose of mechanical support for the poloidal and toroidal coils and because the geometry of the rest of the machine will not be modified, the Saddle Coils, together with a new toroidal supporting structure and the new shell, will have to fit in the space formerly occupied by the shell. Moreover, since the electromagnetic effectiveness of the Saddle Coils would be strongly reduced if the full coverage of the torus surface is not assured, each coil will have to be strictly adjacent to the previous and to the next, both in toroidal and poloidal direction, without allowing open gaps between coils. These constraints have obliged to a careful and very space-saving design of the 192 Saddle Coils and of the supporting structure.

The Saddle Coils will be installed in 4 toroidal and 48 poloidal grooves on the external surface of the toroidal support structure, immediately below the 48 toroidal field coils. The Saddle Coils will be clamped against the supporting structure by the toroidal Field Coils, whose clamping systems also preload the Saddle Coils by means of an interposed silicon-rubber layer.

The paper presents the various aspects of the design: the evaluation of the electrodynamic loads, the simulations of the thermal and mechanical working conditions, the mechanical analyses developed for the design of the coils and their fixing system, and illustrates the solutions adopted for the definitive proposal.

The Saddle Coils are presently under construction.

F-16

T-15M MAGNET SYSTEM

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Tokamak T-15M is under manufacturing design by cooperation of Russian Scientific institutions and industry. The T-15M magnet system consists of twenty Toroidal Field Coils (TFC), six Poloidal Field Coils (PFC), six coils forming the Central Solenoid (CS) and Support Structures. The copper windings with water cooling are used. The magnet system provides the possibility to realize the ITER similar plasma configuration with major radius of 1.55m, minor radius of 0.5m, plasma current of 1.7MA, discharge duration of 5 sec. TF coils provide a toroidal field of 2.5 T at the plasma center.

The electromagnetic loads due to interaction of the magnetic fields with the currents in the coils have been calculated for normal operation and plasma disruption events. The temperature fields due to Joule heating and radiation have been also analysed.

Stress analysis of the magnet system under the electromagnetic, thermal and dead weight loading has been carried out. The Global model of the magnet system sector has been developed and analysed with the help of ANSYS code. The Global mechanical behaviour of the TFC, PFC and support structures has been obtained. The local finite-element models have been built for the most loaded section of the TFC inner leg and for the key-bolt connection of the outer inter coil structure. The mechanical behaviour of the free standing CS has been analysed. The design of the pre-compression structure providing the required axial pre-loading force on the CS has been fulfilled. The structural materials for the magnet system components with the mechanical properties required for satisfying the structural design criteria have been chosen. The structures are designed to withstand 50,000 full-power operational pulses.

F-18

EXCITATION OF TORSIONAL OSCILLATIONS IN GENERATOR SHAFT LINES BY PLASMA FEEDBACK CONTROL

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The ASDEX Upgrade (AUG) tokamak requires an electrical power up to a few hundred MVA for a time period of 10–20 s. The power and energy is provided by two separate networks based on the flywheel generators EZ3, EZ4 and by the generator EZ2 dedicated to the toroidal field coil. EZ3-EZ4 supply the high voltage power supplies for additional heating and the thyristor converters allowing fast control of the DC currents in the poloidal field coils.
In 1999, during a routine check performed on generator EZ3, it was discovered that the coupling bolts of the flywheel-generator shaft line were deformed. Given that the active load of the generator (~100 MW) in service is well below the design value of the shaft (~1000 MW), the damages may only be explained by torsional resonance of the flywheel-generator shaft, itself excited by active power transients from the converter loads. Using a simplified model of the shaft line, eigen-frequencies between 20 and 30 Hz have been calculated.

Devices for measuring the active and reactive power output of the generators EZ3 and EZ4 were installed in order to investigate the load curves during plasma shots. Normally, the power demand of a tokamak can be characterised by a high active power (P) demand during plasma ramp-up, a relative small demand of P during the plasma flat-top phase and a feedback of active power ($P < 0$) during plasma ramp-down. However, plasma feedback control can cause fast changes of the P demand in the power supply (>1000 MW/s) and active power oscillations with amplitudes in the range of 100 MW and frequencies between 0 and 100 Hz.

Frequencies between 10 and 30 Hz have been identified in the spectrum of the dynamic load curves of feedback controlled plasma experiments. Since torsional shaft oscillations are characterised by a very low damping, torsional resonance can become dangerous even for over-dimensioned generator shaft lines. Therefore, a novel 'torque' measurement system, developed by the University of Dortmund (Prof. Kulig) and ITWM (Fraunhofer Institut für Techno- und Wirtschaftsmathematik), was installed on generators EZ3 and EZ4. The system is unobtrusive as it measures the shaft torsional stress taking advantage of the magnetostrictive effect of iron. It provides for an online monitoring of the shaft line torsion at the sensor location and for a generator protection in immediately aborting the plasma pulse once a critical stress value is exceeded. In addition, a finite element analysis of the EZ3 flywheel-generator shaft line, aimed at better defining the eigen-frequency and the dynamic behaviour of the shaft line, is presently being performed.

The paper will present the measurement results showing that generator shaft lines can be excited to resonance by plasma feedback control. Therefore, devices capable to measure the stress in the shaft line are stringent for generator protection.

F-19

MECHANICAL RESPONSE OF THE ITER TOROIDAL FIELD MAGNET SYSTEM DURING FAULT CONDITIONS

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The magnet system of ITER consists of 18 superconducting toroidal field (TF) coils, a free-standing central solenoid (CS), six poloidal field (PF) coils, and 18 correction coils (CCs). The TF coil case, which encloses the winding pack, is the main structural component of the magnet system. Wedging along the sidewalls of the inboard TF coil legs resists the radial in-plane electromagnetic load, with friction playing an important role in supporting the out-of-plane magnetic forces. In the curved regions above and below the inboard leg, four upper and four lower poloidal shear keys support the out-of-plane loads. In these regions, the coils are linked by means of two upper and two lower pre-compression rings, which provide a radial centripetal force and improve the operation of the shear keys. In the outboard region, the out-of-plane support is provided by four sets of outer intercoil structures (OISs). The TF coil operates at high peak field (11.8T), uses Nb_3Sn-type superconductor and is cooled with supercritical helium in the range 4.4–4.7K. The operating current is 68 kA.

During normal operation, the TF coil currents are identical. In case of electrical fault, such as a short circuit or a quench in a TF coil without discharge of the others, the current distribution is non-uniform, the electromagnetic force distribution is not symmetric and the mechanical response to the out of plane loads is altered. If the current inbalance between coils is high, wedging force and friction force can be overcome, large displacements can take place and the clearance with other components reduced. The analyses of these fault events require large and extremely detailed finite element models. Details such as the CS and the preload structure and contact surfaces have to be properly modelled. Non-linear effects such as contact and friction forces have to be included. We have developed a model to perform such analyses, which describes 180° of the magnet system and includes all the important non-linearity and components. The analyses have been carried out with the finite element code ANSYS.

This paper describes the main features of the 3D finite element model and reports the results of the mechanical response of the TF coil under not symmetric loading conditions. The main consequences of these events have been assessed including stress in mechanical structures of the TF coil and the reduction of clearance with other components such as the CS, thermal shield (TS) and vacuum vessel (VV).

F-20

THE UPGRADE OF THE RFX FAST PROTECTION SYSTEM IN VIEW OF THE NEW OPERATING SCENARIOS AND MACHINE MODIFICATIONS

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In RFX, the main central protection system is composed of two parts. One of them, named RGM in the following, provides for the protection of the RFX windings against electric faults; it is based on a fault detection system, which consists of about a hundred electromagnetic transducers, located on the windings and their bus-bars, and of a central processing unit. The other part, named SGPR in the following, collects the protection requests coming from RGM and all the power supply equipment, processes and coordinates such requests and finally produces the commands to the protection devices. The overall intervention time is within one ms. Since many types of faults can arise of different severity, the system is based on hierarchical levels of intervention.

RGM was originally designed taking advantage of the symmetries of the windings and of the circuits topology. In normal operation, some pairs of fluxes, currents or voltages present equal values, while, in fault or anomalous conditions, unbalances occur and stated thresholds are exceeded, giving rise to protection request signals. The new operating scenarios, for the control of the MHD plasma modes, require setting voltage or current imbalances in machine windings, which would be considered as faults by the original logic system. The revision, which will be described in the paper, required the implementation of a new set of probes for the detection of coil short circuit in the TF coils and the rearrangement of other sets of probes dedicated to the detection of earth faults.

As for SGPR, the main modification consists in a re-definition of the hierarchical levels: in particular, the level called ST (Soft Termination) is split into three sub-levels. In the past, the protection action of the ST consisted in the free-wheeling command to all the four groups of main ac/dc converters, including those providing for plasma equilibrium and toroidal field control. At high plasma currents (1MA), fast terminations were occasionally observed, caused by the sudden loss of the toroidal field control, subsequent to a ST request. Moreover, with the new copper shell characterised by a smaller vertical field penetration time constant, a loss of plasma equilibrium, caused by turning off the converters for the vertical field control, would be damaging to the vacuum vessel. Therefore, the splitting of the ST level will allow switching off the four converter groups independently, according to different protection requests, thus guaranteeing equilibrium and reversal also at the end of the discharge and avoiding plasma fast termination.

The paper will describe the experimental evidences and the new requirements due to the machine and circuit modifications, which called for the revision of the systems, their new architecture and finally all the technical solutions adopted to further improve their reliability and flexibility.

F-21

REALISATION OF THE COOLING SYSTEM OF THE NON PLANAR COILS FOR W7–X

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During the design phase of the Wendelstein 7–X coils it turned out, that the foreseen coating by spot welding of copper plates can no longer be applied due to the shape of the cases. Other methods had been investigated for coating with copper. Taking into account all existing boundary conditions it turned out, that thermal spraying is the best solution for this task [1]. In addition to the coating, the coils are equipped with sheets of high conductivity copper in thermally high loaded regions. We report in this paper about our investigations to characterise the copper coating with respect to its physical properties and the necessary tests and procedures for realisation of the cooling system.

As more than 75% of the coil surface, which is exposed to the cryovacuum, is covered with thermal sprayed copper, its thermal conductivity at cryotemperatures is the essential parameter for proper performance of the cooling system of the W7–X coils. We measured these values for several coatings, which had been prepared by several companies using different spraying methods: flame spraying using powder and wire, high velocity spraying and arc spraying. In parallel, also the adhesion strength of these coatings was measured. In order to apply the thermal spraying of copper, the gap that exists between a cooling tube and the case surface has to be filled. We investigated different possibilities using resins, glues and also spraying of different metals in order to find suitable material combinations. The second important step which needed development was the fixing of the copper sheets on the coil case. Because the coil case is sandblasted prior to spraying, it is necessary to prevent particles from protruding between the copper sheets and the coil case. Therefore, the volume between case and copper sheet has to be closed. At a first glance, welding seems to be the method to choose. However, this turned out to be a difficult and slow process due to both the difference in thickness between the coil case (ca 40 mm) and the copper plate (1mm) and the geometry of the case surface. Secondly, there is the danger of local buckling of the copper stripes due to plastic deformation during the blasting process. From this point of view, an alternative might be gluing, which in present reports is proposed also for other applications related to superconductivity [2]. This has the advantage that a full area connection of the copper stripes to the coil case is possible. The first tests with respect to this approach led to encouraging results.

As a result of all these investigations, a solution was found, how the cooling system for the non-planar Wendelstein 7–X coils can be realised in an industrial process.

[1] H. Scheller et al. Development of a Copper Coating Suitable for Application on the Nonplanar W7–X Coils, MT-17, Geneva, 2001.
[2] A. Perin et al. Study of Material and Adhesives for Superconducting Cable Feedthroughs, CEC–ICMC'01, Madison, Wisconsin, 2001.

F-22

MAGNETS OF THE KAZAKHSTAN TOKAMAK FOR MATERIAL TESTING

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This report describes the design of the magnets of the Kazakhstan tokamak for material testing (KTM). KTM is a spherical tokamak with aspect ratio A = 2 and vacuum vessel elongation. The toroidal field of 1.0 T (on the plasma axis) is produced by 20 four-turn water cooled coils, stretched in a vertical direction. Each coil consist of three parts bolted each other. The wedged parts of the toroidal coils form the central column to support the centripetal forces. The outer parts of the toroidal field attached to the support column and upper and lower installation units. The overturning loads are taken up by the mutual action of the support column and installation units and interblock structures. Poloidal field coils are fasten to the toroidal coils.

Central solenoid is wound on a special central rod and have four layers. The magnetic field at the central solenoid axis amounts to 7.0 T. It is located into the central column. The central solenoid is designed to sustain 100000 loading cycles.

The vacuum vessel is a welded stainless steel structure. The vacuum vessel contains horizontal manholes and vertical ports. The divertor components are located in the lower part of the vacuum vessel. The vessel is to be baked to 200° C. For this purpose the vacuum vessel is covered with thermal insulation.

The support structure serves for holding the toroidal and poloidal field coils, vacuum vessel, takes up the magnetic forces acting on the toroidal and poloidal field coils and transfers the weight of the KTM magnets to the foundation.

In the paper the basic parameters of the KTM magnets and PF coils and CS are given. The TF and PF coils fastening and vacuum vessel and support and the interblock structures are described. Results of mechanical and thermal calculations on the KTM magnets and its elements are shown in the paper.

F-23

DESIGN AND MANUFACTURE OF THE POLOIDAL FIELD CONDUCTOR INSERT COIL

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The Poloidal Field (PF) coils of ITER will supply the necessary magnetic field to initiate, shape, control and shutdown burning plasmas. They will feature NbTi cable-in-conduit superconductors, which use stainless steel square tubes as structural material. These conductors have to operate at maximum currents in the order of 45 to 60 kA and will have to experience large variations of current and magnetic fields during the flux swing needed to generate the plasma. In order to test full-scale NbTi superconductors at operational conditions relevant to ITER, the European Participant Team has been requested to design and manufacture the PF Conductor Insert (PFCI) coil. The Insert will be installed and tested in 2004 in the Central Solenoid Test Facility (CSTF) at JAERI Naka, Japan.

The PFCI coil consists of a single layer solenoid with nine turns, an intermediate joggle at the equatorial plane equipped with inductive heaters, temperature and voltage sensors, and an intermediate joint located in the upper half of the solenoid. This should allow to acquire sufficient data to understand the behaviour of these large NbTi superconductors and their joints. The coil will be inserted into the bore of the CSTF and connected at top and bottom to the existing Insert electrical and cryogenic supplies. The facility can generate magnetic fields up to 13 T in transient conditions and will provide the background field of about 6–7 T, which is the maximum tolerable for NbTi. The nine turns will be impregnated in a large cylinder made of fibre reinforced plastic. A pre-load system consisting of two flanges and twelve long tie-rods will be used to provide some vertical preload to the coil during assembly and to prevent debonding of the turns during operation. The pre-load system also restrains the Insert against the CSTF structures to react the electro-magnetic load during operation. Preliminary electro-magnetic and stress analyses of the PFCI coil have been performed by the Russian Participant Team (Efremov Institute, St. Petersburg). The terminal joint design and manufacturing procedures will be validated with a Full Scale Joint Sample (FSJS) to be tested in the Sultan facility, Switzerland.

The Russian Federation Participant Team (VNIIKP, Moscow) has produced 70 m of superconducting cable and 20 m of dummy conductor for the manufacture of the PFCI coil. The cables have been shipped for jacketing to Ansaldo Superconduttori, Italy, to produce the conductor lengths needed for the fabrication of the coil and the FSJS. Manufacture of the coil has started with the production of quality documents, manufacturing drawings and tooling.

This paper describes the design of the PF Conductor Insert coil, including the main results of the electro-magnetic and stress analyses, and a description of the interfaces to the test facility. The status of the manufacture of the superconducting cable, jacketed superconductor and coil winding is also presented. Preliminary testing scenarios have been evaluated and will be outlined in the paper, including a summary of the instrumentation required for the assessment of the conductor and joint performances.

F-24

SOUNDNESS EVALUATION OF SUPPORT STRUCTURE OF LARGE HELICAL DEVICE

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The Large Helical Device has a large-scale superconducting magnet system and stiff structure to support superconducting helical and poloidal coils and to sustain huge electro-magnetic force. The support structure was constructed with 100 mm thick stainless steel of type 316. The structure was divided into 10 sectors, and the lower and the upper halves were welded at the equator. The partial welding was applied to reduce the welding deformation, so the metal touched area remained at the middle of the thickness, when the toroidal cross-sections were welded. In the case of the equator welding, the full penetration was achieved but the thickness was reduced to be half. The stress acts on the support structure by the electro-magnetic force is expected to be less than 100 MPa so that the current center of the coil does shift very little. In order to monitor the deformation of the structure, strain gages were attached on the weld joints and the measurement was carried out carefully

Since the electrical break is not installed in the support structure, the eddy current runs and temperature of the structure rises when the coil currents are increased or decreased rapidly. The temperature rise affects on the strain measurement to generate the apparent strain. When the apparent strain occurs, it is not easy to distinguish the mechanical strain and the apparent strain. Therefore, the lower ramp rate of 0.02 T/min at magnetic axis is applied and the magnetic field is increased to 2.85 T. It takes about five hours. Such measurement was performed four times until now, October 4, 2000, January 19, 2001, November 29, 2001 and January 11, 2002, under the same conditions, and the strain behavior was investigated and compared with each other.

The relation between the strain and the square of the magnetic field, which is corresponding to the magnetic field at the magnetic axis, is plotted and the hysteresis curves are obtained. The hysteresis would be generated by the conductor motion, which showed non-linear behavior due to friction between conductors and spacers. To investigate the identity of the hysteresis curves obtained at the different measurement days, one date set of the strain is checked against the other set at the same magnetic field. All data on the ramp up and ramp down processes is examined and it is found that the all plots exist within the scatter band consisted of +/-3 digits. This is reasonable when one can consider that the error of +/-2 digits comes from each data sets and the additional +/-1 digit comes from the error of the magnetic field. Moreover, the change of the strains at ten sectors for one year and half is studied and it is confirmed that the significant difference does not generate. From these results, it is evaluated that the support structure works well and is in good condition.

F-25

CONSTRUCTIONS OF 50 SUPERCONDUCTING NON PLANAR COILS FOR W7-X EXPERIMENT

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An order for the construction of 50 non planar coils for the W7-X Stellarator (10 equal coils for each different type) has been placed by the Max-Planck-Institute for Plasmaphysics to a consortium of two firms (BNN and ASG). The winding-packs are made using an aluminium coextruded NbTi cable in conduit specially developed for this project. This special cable was ordered by the Babcock Noell Nuclear-Ansaldo Superconduttori Consortium to another Consortium of two European firms (VAC and Europa Metalli) and now, after several problems encountered during the development phase, the fabrication process of this cable seems to be established.

The coils to be made are similar to the prototype already made by BNN/ASG some years ago although the real coils presents several changes (the conductor dimension, the shape, the coil case, the electrical exits…).

Actually the first coils are under construction in Italy and Germany. The coils of type 2-3-4 are made by Ansaldo Superconduttori by winding the cable on an aluminium winding form that will also be used as mould for the heat treatment of the aluminium alloy and vacuum impregnation. The coils of type 1 and 5 are made by BNN in the ABB premise of Augsburg by winding the cable on an steel winding form before to move it into an aluminium mould for the heat treatment and vacuum impregnation.

After their final electrical test all the coils will be send to BNN at the premises in Zeitz where they will be mounted into the coil case and embedded before to be sent to CEA for the cryogenic test (see other papers in this conference).

In this paper a description of the fabrication technology as well as the test procedure used for the manufacturing of the impregnated winding pack both in Ansaldo and BNN is described. A preliminary result about the quality of the winding is reported.

In addition the development and the cryogenic test of the joints between the superconducting cable of the different layers is reported and described.

F-26

DC BEHAVIOUR OF NbTi CICC FOR ITER-FEAT

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To study the relation between the behaviour of cable-in-conduit conductors (CICC) and strands five NbTi sub-size conductors were fabricated from three different strands. The NbTi sub-size cable-in-conduit conductors are presently tested in the SULTAN Test Facility. As a basis for the comparison of cable and strand properties the critical current of the three different NbTi strands was measured in the temperature range of 4.2–6.5 K for magnetic fields between 4 and 7.5 T.

Strands A and B of 0.7 mm diameter are characterised by copper : non-copper ratios of 7.5 : 1 and 1.02 : 1, respectively. As a consequence of the different copper : non-copper ratios the filament number for Strand A is only 48, while 690 filaments are in Strand B. The number of filaments in Strand C of 0.87 mm diameter is 690. The resulting copper : non-copper ratio of 1.02 : 1 is the same as for Strand B. The measured Ic data of all three strands can be well represented by the scaling law $I_c(B,T) = (C_0/B)(1-(T/T_c)^{1.7})^{\gamma}(B/B_{c2}(T))^p(1-(B/B_{c2}(T))^q$ where $B_{c2}(T) = B_{c20}(1-(T/T_c)^{1.7})$. However, the values of the scaling parameters B_{c20}, T_c, C_0 and γ are slightly different for the three strands. In addition to the critical current, the index of the resistive transition n ($I = I_0(E/E_0)^n$) was determined for electric fields between 0.015 and 0.2 μV/cm. The n values have been found to decrease considerably with increasing temperature and applied magnetic field.

Three of the five NbTi sub-size cable-in-conduit conductors were fabricated from Strand B. These conductors are distinguished by the strand coating (electroplated Ni versus SnAg solder) and cable layouts with and without sub-cable wraps providing deeper insight in the effect of the effective transverse resistance on the DC performance. The two other conductors fabricated from Strands A and C are characterised by a large fraction of copper in the cable cross-section. In the conductor fabricated of Strand C the large copper cross-section is provided by segregated copper.

The DC behaviour (I_c, T_{cs}) of the different sub-size cable-in-conduit conductors is compared to the strand behaviour. The effects of cyclic loading on the effective transverse resistance, the critical current and the n values has been studied. First results suggest that cyclic load reduces both the critical current and the n value. This result indicates that the n value in a superconducting cable is closely related to current transfer effects.

F-27

SUPERCONDUCTING COIL WITH PURE COPPER WIRES FOR THE THERMAL PROTECTION OUTSIDE THE CABLE-IN-CONDUIT CONDUCTORS

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Generally, the cable-in-conduit conductors of superconducting coils for fusion reactors have pure copper wires for the necessary thermal protection. A number of the copper wires are decided by the fast discharge time of the coil when the coil was quenched. As the discharge time is shortened, a number of the copper wires are reduced and then the coil becomes compact. However, the discharge time length is limited to the varying magnetic field which is produced during the discharge, on a plasma-confinement vacuum vessel because the vessel is broken due to electromagnetic force produced with the current induced in the vessel by the varying magnetic field. In the case of the TF coil for ITER the discharge time constant is designed to be 11 s. Therefore, in this paper the realization of a compact ITER TF coil with the varying magnetic field corresponding to the time constant of 11 s is considered.

The coppers wires within the cable-in-conduit conductors are eliminated, but a copper coil fabricated with the eliminated copper wires is provided in the coil case in coaxial with the TF coil. A number of the turns of the copper coil are 10 and the time constant is estimated as around 14 s. If the quench detection time is 2 s, the TF coil is calculated as 200 K to be almost equal to the copper coil in the temperature rise during the discharge of the TF coil under the adiabatic condition. The temperature rise of the copper coil is permitted up to 300 K because of no superconducting material. It is the result that the number of copper wires of conductor of the copper coil can be reduced by a half of the copper wires. And furthermore the cross-section of the conductor becomes smaller because the conductor has not coolant helium space.

The cross-section of cable-in-conduit conductor for the ITER TF coil is reduced by around 40% due to the elimination of pure copper wires under the condition that the time constant of 11 s is kept. And also a protection resistor system provided outside the cryostat is reduced because a part of magnetic stored energy is dissipated into the coil case contacted thermally strongly to the copper coil. Finally it was found that the copper wires for the thermal protection could be eliminated from the cable-in-conduit conductors by providing the copper coil in the coil case and the TF coil became compact.

F-28

OPTIMISATION OF THE REACTION HEAT TREATMENT CYCLE OF INTERNAL TIN AND BRONZE Nb_3Sn STRANDS FOR ITER

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The heat treatment cycles recommended by the manufacturers of Nb_3Sn superconductor strands are very long. These cycles were designed to achieve the highest possible critical currents with acceptable low coupling losses at the same time. The heat treatment recommended by Europa Metalli for their internal tin strand is 200 h at 210°C, two steps at 340°C and 450°C 24 h each (to evaporate organic compounds from the surface) and 180 h at 650°C. This makes in total about 550 h (~23 d) including the time needed to heat up and cool down (10°C/h). A similarly long cycle is recommended by Vacuumschmelze for the bronze route strands, namely two steps at 340°C and 450°C 24 h each, 220 h at 570°C and 175 h at 650°C summing up to about 570 h including ramping time. These very long heat treatment cycles pose a manufacturing problem as they are interrupting and delaying the manufacturing process of the large superconducting (SC) magnet coils of the International Thermonuclear Experimental Reactor (ITER). Therefore it was in the interest of this project to investigate if the same or nearly the same performance of the strands could be achieved with shorter heat treatment cycles.

Strand material from both manufacturers as used for the ITER CS and TF model coils were submitted to various heat treatments and subsequently characterized with respect to critical current, coupling losses, RRR (Residual Resistance Ratio) and magneto-resistance. The measurements were supplemented by metallurgical and microscopical investigations.

It will be demonstrated that with considerably shorter heat treatment cycles (~200 h less) almost the same critical currents can be achieved in both types of strands, with a benefit of lower coupling losses (significantly for EM strands) and distinctly less RRR degradation. Even if the filaments were not always completely reacted the finer grain structure gave a better pinning. It can be estimated that the overall time needed to react all ITER TF coil pancakes could be dropped by 7200 h, equal to 300 days. Beside this benefit to the schedule there would be a not negligible reduction in cost and risk of heat treatment failures.

*This work has been carried out within the association EURATOM-OEAW.

F-29

THE STUDY AND DESIGN OF POWER QUALITY CONTROL IN HT-7U POWER SUPPLY SYSTEM

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The study and design of power quality control in power supply system of HT-7U superconductive Tokamak are carried out in order to ensure the safety and reliability in operation and to realize the compatibility between the power supply system and high voltage grid. On the one hand, in the design of power supplies of magnets, one of the principles is to reduce the impact of pulse power, reactive power, harmonic, and unbalance component in three-phase power system as much as possible. On the other hand, reactive power compensation and harmonic suppression are very important and indispensable.

The design of integrating static compensation with dynamic compensation, passive power filter with active power filter is proposed. The feasibility and availability of the design are demonstrated.

The pulse power demands of HT-7U has been estimated. And the scheme of static var. compensation and harmonic suppression is described.

Multi-group switchable capacitor banks and thyristor-controlled reactor/transformer, and hybrid active power filter are installed in the intermediate 10KV grid. The active power filter is set up between the inductance and the capacitor in the high pass filter in order to improve the performance of the compensation furthermore.

The R & D of key technology in compensation devices have been made progress. A high voltage AC compound switch that is based on the ordinary vacuum switch and thyristor valves for frequent operation in capacitor banks is developed. A new definition of instantaneous power theorem in the integration vector plane is derived and a novel approach to detect reactive power and harmonic current as well, which brings the advantages of simple arithmetic, real time and wide application. Moreover, the dynamic compensation control with high precision and fast response is investigated. The results of simulation and experiment will be presented.

With the power quality control in HT-7U power supply system, the grid voltage fluctuation and flicker and total harmonic distortion are within the limit of GB national standard and IEC standard. The compatibility between the power supply system and high voltage grid is realized.

F-30

POWER SUPPLIES FOR THE WENDELSTEIN 7-X STELLARATOR

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The magnetic confinement of the plasma in WENDELSTEIN 7-X (W7-X) will be determined by 50 superconducting non-planar coils and 20 superconducting planar coils. The coils are grouped in five periodic modules with 10 coils each connected in series. To fine tune the magnetic configuration, shift the plasma axis and to modify the plasma edge an additional set of 10 normal conducting control coils will be mounted. Different power supplies are required for the superconducting and the normal conducting coils, depending of their design parameters and their electromagnetic characteristics.

In order to be able to vary the magnetic configuration and hence increase the experimental flexibility the five groups of non-planar coils and two groups of planar coils will be powered individually resulting in seven independent power supplies. The magnetic coupling of the coils and the required high accuracy of the coil currents requires a very precise control system. To achieve a magnetic induction of 3 T the coils need to be powered by currents of up to 20 kA at voltages below 3 V. Steady state operation claims for additional requirements. The power loss of the power supply which should be as low as possible is not only determined by the design of the power supply but also by the length of the copper bus to the coils. Therefore the power supplies will be installed close to the stellarator where, however, the influence of the stellarator stray field is not longer negligible and has to be taken in account. Different configurations for the power supplies were studied in detail to fulfil the requirements of W7-X. The selected concept is based on a twelve pulse thyristor-based converter with interphase transformers. The choice of twelve pulses helped to reduce the ripple on the output current.

The 10 „control coils“ allow to modify the magnetic configuration at the plasma edge and further enhance experimental flexibility. The coils use a copper conductor and are situated inside the plasma vessel, behind the baffle and target plates. The power supplies for the control coils have to provide a direct current of 2500 A bi-directional and with high accuracy and low ripple. To allow to sweep the power deposition from the plasma across the target plates the power supply shall allow for periodic variation of the magnetic field locally. As a result the power supply has to provide an alternating current of up to 625 A with frequencies up to 20 Hz. The design of the power supplies is based on 10 independent MOSFET-based, four-quadrant power supplies. All ten power supply units have meanwhile been delivered and are currently installed at Greifswald.

The paper describes the design of the power supplies for W7-X as well as the results of the first component tests for the power supplies for the superconducting coils and results from the works acceptance tests for the power supplies for the control coils.

F-32

THE SUPERCONDUCTING BUSBAR SYSTEM OF WENDELSTEIN 7-X

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WENDELSTEIN 7-X (W7-X) is presently the largest fusion experiment of the stellarator type under construction. The superconducting magnet system consists of 50 non-planar and 20 planar coils grouped in 5 periodic modules. 10 coils of a given type of non-planar and planar coils will always be connected electrically in series with nominal currents ranging up to 18 kA. The busbars are routed bifilar and uses the same internally cooled cable-in conduit superconductor as the coils. To connect the coils in series and with the current supplies and provide the joints between neighbouring modules some 25 single superconducting lines with lengths between 4 to 14,5 m need to be routed in each module. Each busbar system will be pre-manufactured and tested as a single part to limit the number of joints and simplify assembly. Special requirements of the busbar system are helium tightness of the helium cooling loops, mechanical strength of the electrical insulation and a dielectric strength of 13 kV which may occur during a quench.

The busbar system needs to be supported by dedicated fixed bearings to balance the large electromagnetic forces caused by the magnet system and gliding bearings to allow for thermal contraction during cool-down. Electrical connection of the busbar system will require 184 disconnectable joints with a resistance below 5 nΩ. The joint design is based on the design for the connection between the double-layers of the non-planar coils. Some modifications were necessary to disconnect the contact area of the strands without damaging the strands.

The presentation will describe the design features of the busbar system and of the supports and focus on the design and qualification of the disconnectable joints. The complexity of assembly of the busbar and of the joints within the restricted space of the W7-X cryostat will be highlighted.

F-33

POWER SUPPLY OF THE CONTROL COILS OF WENDELSTEIN 7-X EXPERIMENT

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On the way to a fusion power station a new stellarator is being built in the Max-Planck Institute for Plasma Physics, IPP, in Greifswald, to prove the effectiveness of the continuous stellarator operation against pulsed Tokamak. Besides superconducting main field coils, there are 10 normally conducting coils, so-called control coils, to get precise relative position of the plasma by means of varying the magnetic field locally.

The 10 control coils in the stellarator are regulated by 10 power supplies, that must provide a controlled current compounded of direct current and 0–20 Hz bandwidth ac current in a range of almost 3 kA at low voltage, 30 V, in four quadrants. High current with low ripple and low harmonic distortion is required by load. Due to the closer location of the power supplies to the Torus of the stellarator, high magnetic fields (around 50 mT) must be withstood by the power supplies system, but keeping on regulation precision of the entire system.

During year 2001, JEMA has manufactured and successfully tested at factory all equipment corresponding to the power supply system of the Wendelstein 7-X Control Coils, 10 power supplies, the general cooling system and the power distribution system. Beginning of March 2002 start up has started on IPP facilities in Greifswald and expected to end before summer 2002.

There will be available last results and curves of the operation for final approval of the power supplies, so they will include in the paper and compare with the partial results got during tests at factory. Special attention will be paid to the tests done to avoid the high magnetic field affecting the right operation of the power supplies.

F-34

145 MVA MODULAR THYRISTOR CONVERTER SYSTEM WITH NEUTRAL CONTROL FOR ASDEX UPGRADE

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During the next few years, ASDEX Upgrade (AUG) will strengthen its efforts concerning investigations of advanced tokamak scenarios in connection with increased triangularity shaping and divertor operation. To fully exploit these operating modes a plasma current flattop time of at least 2–3 plasma skin times, i.e. 10 seconds, is required. Additionally, the currents in mainly the upper control coils have to be increased. To cover these demands, a new Thyristor Converter Group 6 was designed, installed and commissioned.

The converter has to fulfil three main objectives: Modularity and flexibility - to substitute any of the existing converters in case of malfunction; Power factor improvement - to optimise power consumption for future long pulse operation; Fast four quadrant operation - to improve the possibilities of plasma shape control.

All converters installed are powered by 10.5 kV flywheel generators starting at a frequency of 110 Hz, running down to 85 Hz during a pulse with a maximum short circuit power of 2,250 MVA at 85 Hz. The Thyristor Converter Group 6 consists of four identical modules. The nominal pulsed rating of every module is 22.5 kA, 1.25 kV (no-load voltage 1.61 kV) for ten seconds every five minutes. The modules can be operated independently or combined in parallel, series (both 12-pulse) or antiparallel by means of DC-switches. This results in a total of eight different modes of operation with currents up to +/-45 kA and voltages up to 3.0 kV in two- or four-quadrant operation, to cover the diverse demands of the various AUG magnets. The converter modules are designed and tested to be short-circuit proof on its DC terminals. The converter transformers have fixed voltage ratios. The no-load voltage can be limited electronically. The neutral connections of the transformers are made available for reduction of the reactive power consumption from the mains in controlling the phase to neutral voltage by means of neutral thyristors. The neutral control can be switched off, if required. For the fast four-quadrant operation a circulating current is necessary. The circulating current will not be switched off at higher DC output current levels in order to avoid any additional DC voltage transients. The voltage loop bandwidth of the converter is 300 Hz.

The paper describes the design and testing of the modular Thyristor Converter Group 6 with neutral control and four quadrant possibilities. It will present the various configurations on the AUG magnetic coils, analyse the results of measurements obtained during commissioning, compare them to the calculated (design) values and report on the performance achieved in fast four quadrant operation improving the possibilities of the AUG feedback control of plasma shape and position.

F-35

AC LOSSES DUE TO EDDY CURRENTS IN THE MAGNET COLD STRUCTURE OF THE ITER TOKAMAK

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Variation of the magnetic fields at the ITER operation causes eddy currents in the conductive structures, and results in the resistive losses. A part of this energy dissipates in the structures at a low temperature level and has to be removed by the liquid helium flow. The main objective of the analysis is numerical simulation of eddy current behavior and calculation of the energy deposition in the magnet cold structures (except for superconducting coil windings).

The cold structures numeric model were modelled in fine detail, while the other components models were much simpler.

Integrated numeric model of the ITER, was used with the following elements (sub-models):

- Magnet cold structures having thermal contact with Toroidal Field (TF) coils and Poloidal Field (PF) coils: TF coil case, intercoil and pre-load structures, PF coil supports;
- Magnet cold structures having thermal contact with the Central Solenoid (CS): CS pre-load and supporting structures;
- Double-walled Vacuum Vessel (VV);
- PF coils, as a source of field variation;
- Plasma, as a source of field variation.

The following operation conditions have been simulated:

- Reference plasma operation scenario;
- Fast downward Vertical Displacement Event (VDE);
- Fast downward VDE followed by CS/PF coil fast discharge.

For each plasma behavior scenario (except for the Reference plasma operation scenario), two transient electromagnetic processes have been analyzed separately and superimposed in automatic mode: one caused variation of the plasma full toroidal current, and another, caused by variation of the toroidal magnetic flux created by the plasma and Halo currents.

The calculations have been carried out using the TYPHOON code, developed at the Efremov Institute. TYPHOON code is designed for simulation of 3D transient electromagnetic processes in thin interconnected conducting shells.

F-36

DEVELOPMENT OF HL-2A POWER SUPPLY

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The HL-2A tokamak is under installation based on main components of ASDEX from IPP at Southwestern Institute of Physics(SWIP), Chengdu, China. This paper presents the design and development of power supply system for HL-2A tokamak.

The power supply system is designed and developed according to the requirements of the operation of HL-2A at following parameters: toroidal field of 2.8T, plasma current of 450 kA with a flat top of 5s. Thus, the peak power required is 300MW and energy content is about 1200MJ per shot. Three flywheel motor-generators (MG) are used to power the system by transferring the power and energy from commercial HV grid. Up to now, following progress has been made for this system: (1) Two identical existing MG are modified by three means: A. Replacing original flywheel of 40 ton to a new one of 90 ton. B. Raising the maximum speed of the shaft system up to 1650rpm from 1500rpm by using a static Scherbius consisting of cyclorectifier for each driving motor. C. Increasing the speed drop of the total shaft from 1488rpm–1320rpm to 1650rpm–1200rpm. After modification, the maximum apparent power for each generator can reach 90MVA from 80MVA and released energy can be increased to 500MJ from 100MJ. They are used to power the toroidal field coils via a 12 pulses diode rectifier. The current in TF coils is controlled by regulating the exciting field current of two MG sets. Another MG with output power of 125MVA is used to power the poloidal field system with transformers and thyristor rectifiers. The currents in PF coils are controlled by a feedback control system. (2) In order to check the system design and optimize the parameters of feedback control system, the power supply system has been simulated with EMTP code. Some special computation models for the system elements such as generator with six phases, transformers with three phases three windings and twelve pulse thyristor rectifiers including firing circuits are developed. Based on these models, the simulation of the entire system has been carried out. (3) A new digital trigging method for thyristor rectifier has been developed to adapt to the AC frequency changes from 120Hz to 96Hz corresponding to the MG shaft rotating speed slow down from 1650rpm to 1200rpm. By using this method, the control precision of the thyristor rectifier can reach 0.04^0. (4) Using the advanced technology develops the monitor control and protection system of the power supply. Its consists of microcomputers, PLC and application software such as Wincc and step7.

The simulation and primary tests show that the design of the HL-2A power supply system is able to satisfy the requirements of the operation of HL-2A.

F-37

OPERATIONAL EXPERIENCE WITH REACTIVE POWER CONTROL METHODS OPTIMISED FOR TOKAMAK POWER SUPPLIES

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The power and energy of the ASDEX Upgrade (AUG) tokamak are provided by two separate 10.5 kV 110–85 Hz networks based on the flywheel generators EZ3-EZ4 in addition to the generator EZ2 dedicated to the toroidal field coil. The 10.5 kV networks supply the thyristor converters allowing fast control of the DC currents in the AUG poloidal field magnet coils. Engineering activities devoted to the optimisation of the power demand by reactive power compensation were undertaken to satisfy future power requirements.

The large inductive voltage required by the poloidal field coils during plasma ignition and plasma ramp-up lead to a large reactive power demand from the thyristor converters during the plasma flat-top phase, as the firing angles are phased-back. Typically, the power factor of EZ3 load drops from 0.6–0.8 during the plasma current rise to 0.2–0.3 during the plasma flat-top.

Two methods for improving the load power factor in the present experimental campaign of AUG have been investigated, namely the control of the phase-to-neutral voltage in thyristor converters fitted with neutral thyristors, such as the new 145 MVA modular thyristor converter system (Group 6), and the reactive power control achieved by means of static VAr compensators (SVC).

In 2001, a 120 MVAr SVC system was installed and commissioned for the 10.5 kV network of generator EZ3 (144 MVA, 500 MJ). The characteristics of the reactive power load curve during AUG load pulses was the determining factor for the decision to compensate reactive power by capacitor banks energised by line synchronised vacuum breakers, as opposed to thyristor controlled compensation. The number of SVC modules per load pulse as well as the optimum time for switching each module is determined by the SVC control system and depends on the measurement of the total reactive power demand.

The paper will analyse the results obtained with the 120 MVAr SVC during the 2001–2002 operational campaign and show that reliable compensation up to 90 MVAr was regularly achieved. The paper will show that electrical transients in SVC modules can be kept at an acceptable level, even under stringent conditions, such as those experienced in variable frequency power networks with sudden load changes. The paper will discuss the results from the reactive power reduction by neutral thyristor control and draw a comparative conclusion as to the respective advantages and limitations of each method.

F-38

THE POWER SUPPLY SYSTEM FOR THE ACTIVE CONTROL OF MHD MODES IN RFX

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In RFX, the future installation of a new thin shell, with about 50 ms penetration time constant of the vertical component of the magnetic field, will require to provide an active control system both for the plasma equilibrium and for MHD modes. The first will be performed by a system of axisymmetric windings fed by high power ac/dc converters; while the MHD mode control will be provided by a new system of 192 saddle coils. They will completely surround the shell surface and will be equipped with as many flux measure probes, installed at the vacuum vessel outer surface, so as to assure a complete control of the magnetic field at this plasma distance. The system will be able both to minimise the radial component at the "plasma boundary" and to produce m=1, n=1–24 single or multiple modes, so as to investigate a wide range of control actions, like mode rotation, single elicity, resistive wall mode stabilisation.

The power supply system designed to allow these controls is composed of four ac/dc thyristor converters which feed 192 IGBTs dc/dc switching converters, one for each saddle coils, rated for maximum output voltage and current of 650 V and 400 A, respectively. Due to the consistent amount of reactive power absorbed by the load, capacitor banks are provided at the output of the ac/dc converters.

Some parts of the system are an upgrade of those designed for the radial field control at the portholes of the RFX shell outer equatorial gap, but there are several new aspects both in the power and control section, which required deep analyses to be defined. These will be described with particular emphasis in the paper, which will also report the description of the complete power supply system.

As for the ac/dc section, one of the most critical issue is the control optimisation of the ac/dc converters, aimed at limiting the drops of the dc voltage, which feeds the dc/dc converters, at the fast transients of the load current during the pulse.

Also the minimization of the noise to the plasma diagnostic equipment, placed near the saddle coils is a crucial point. It is mainly due to the fast variations of an high common mode voltage, which derives from the particular modulation law, selected for the control of the IGBTs commutations. Analyses and experimental tests on a dummy load were performed, to define the parameters of an EMI filter, provided at the output of each cc/cc converter.

F-39

SMES-UPS FOR LARGE-SCALED SC MAGNET SYSTEM OF LHD

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LHD is a large-scaled superconducting (SC) toroidal fusion device of heriotron type constructed at National Institute for Fusion Science in Japan. The total shot numbers of the plasma productions exceed 35,000 during four years. Cryostable condition of the SC coils has been kept for more than a half year in every experimental campaign. Large voltage sags or short interruptions caused by thunderbolts happened one or two times in a campaign. Almost of them were short-period phenomena in less than two seconds. The main circulating compressors and turbo-expanders of the helium liquefier/refrigerator immediately tripped, when a momentum power failure with more than 0.1 second occurred. After the recovery processes, such as restarting of the compressors, restarting of the turbo-expanders, and tuning of the temperature distribution of the heat exchangers in a cold box, the SC coils and their supporting structures of the main body should be reconnected to the liquefier/refrigerator system. Therefore, the recovery to the steady-state cooling will take at least more than one day. Since the liquid helium stored in the SC coils and current-leads evaporates and the pressure in a system rises in the meantime, the large volume of the helium gas has to relief into the atmosphere, when the cryogenic system accidentally stops.

To avoid these losses, the SMES for an uninterruptible power supply (UPS) application is investigated. The cryogenic system of the LHD has an equivalent refrigeration capacity of 9.1 kW at 4.5 K. It has eight oil-injected screw compressors with total electric power of 3.5 MW. Therefore, the SMES are designed to compensate the electric power of 3.5 MW during two seconds. The stored energy and available discharge rate are determined to be 14 MJ and 50 %, respectively, in consideration for the economical optimization of the electrical power converters. The analysis model which expressed the characteristic of the circulating compressors in detail is investigated by using the PSCAD/EMTDC analysis code. The effective operation methods of the SMES were surveyed, as the parameters of the voltage sag rate, delay time of sag detection, delay time between the circuit isolation and SMES compensation, and so on.

The induction motors for the compressors are slowed down between the delay times until power failure occurs and SMES backs up, reviving kinetic energy in part as electric energy like a motor-generator. Therefore, it is desirable that this dead time is short as much as possible. It was clarified that the SMES-UPS for the rotating machine of the cryogenic system should have the following functions; 1) to compensate the transient frequency decrease, 2) to regulate the voltage decrease, and 3) to synchronise the phase shift between the load voltage and recovered grid voltage.

F-40

APPLICATION OF SiC-BASED POWER ELEMENT TO HIGH CURRENT AC/DC CONVERTER SYSTEM

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The extremely high DC current, likely 20 MA, should be supplied to the single turn toroidal field coil of a spherical torus (ST) type fusion experimental device. To yield such high DC current with keeping low operational loss in the power supply, the AC/DC converter system is designed to employ its switching unit having the minimized conduction loss. As the switching unit, the Si-based power electronics element of conventional power-MOSFET has been investigated to confirm its advantages of on-state resistance reduction in cooled down and parallel connected conditions.

The crystallized material of SiC has excellent electrical characteristics, and it is now under development to fabricate it into the power electronics element. In comparison to the conventional Si-based power electronics element, SiC-based one can have the advantages of fast switching frequency and high withstanding voltage. From the point of high current operation, the SiC-based power electronics element can promise another electrical advantages that it has lower on-state resistance value and more wide allowable range of operational temperature than that of Si-based one. And also, the SiC-based power-MOSFET, which is one of attractive major carrier power electronics elements, will show the good characteristics of on-state resistance reduction in low temperature and parallel connected operation as same as Si-based one.

In this paper, it is proposed that SiC-based power-MOSFET is applied to the high current AC/DC converter system. The possibility of using SiC-based power-MOSFET depends on operational loss reduction and circuit configuration simplification of the high current AC/DC converter system. Concerning the operational loss reduction, the lower on-state resistance of SiC-based power-MOSFET decreases the conduction loss of switching unit, and it means that the temperature rise of switching unit in operation is lowered. Therefore, the auxiliary cooling equipment for AC/DC converter system will be minimized. The allowable operational temperature of SiC-based power-MOSFET will assist also the minimization of cooling capacity. Whereas many parallel connected elements are required for high current operation, as our previous studies have examined, it is effective for on-state resistance reduction that the parallel connected elements should be cooled down to liquid nitrogen temperature. In the point of circuit configuration, the high withstanding voltage of SiC-based power-MOSFET allows to reduce the series connected switching units, and then the required number of switching unit for the high current but low voltage AC/DC converter would be only one. The reduced series connected SiC-based power-MOSFET elements realize the desirable design for the simplified converter system.

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SUPERCONDUCTIVE CABLES CURRENT DISTRIBUTION ANALYSIS

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Non-uniform current distribution inside the conductors of superconductive magnets might affect their performance and transient stability. The paper reports about the ongoing efforts to study the cable current distribution, both by developing detailed computer codes and by their experimental validation. The paper describes a code which combines detailed on-purpose developed electromagnetic representations of multi-stage cable in conduit conductors and joints with thermo-hydraulic models.

The joint model is made up with discrete equivalent electrical networks representing the cable segments, the saddle 3D network and the cable-to-saddle interface. The cable model is instead based on distributed parameters which are better suited for the description of very long structures. A single channel thermo hydraulic code (possibly to extended if necessary to a multi-channel model) is finally interfaced with the electro-magnetic sections thus allowing for a self-consistent solution. The main results of cross checking with other existing codes are briefly reported in the paper. As a part of the code validation, calculations of the self and mutual inductances have been checked against experimental results.

In parallel to the code development, an important testing activity has been planned on the ITER Toroidal Field Model Coil (TFMC) aimed at measuring the current distribution inside the busbars. The paper describes the theoretical studies on the optimal positioning of Hall magnetic probes to be installed around the busbar and on the design and manufacture of the measuring heads. The results of tests carried out on a resistive mock-up of the TFMC busbars are reported in the paper.

F-42

TECHNOLOGY OF HIGH POWER BREAKING BY COMBINED OPENING SWITCH (VACUUM INTERRUPTER AND PEOS)

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Description and work of unit cells of the technology of breaking by combined opening switch (COS) in circuit of inductance storage and load are offered for consideration. The properties of both interrupters, vacuum interrupter (VI) and plasma erosion opening switch (PEOS), and engineering state of each of them are allow to integrate them into single construction, exactly into single vacuum gap. Under certain conditions the COS is able to break the circuit with IS in 100 nsec under energy storage time ~1sec. The power under it can reach ~10^{12}W and more. [1,2]

The action of COS has four steps: – the first one is energy storage; – the second one is arc discharge; the third one is stage of current zero and the fourth is stage of PEOS and magnetic insulation. Number of movable electrodes in COS construction equal number of IS turns. Fixed electrodes, cathode and anode, are executed as sections integrated into flat or axis symmetric construction and PEOS electrodes. Each section of cathode and anode is connected with a turn of IS. Movable electrode is connected with anode section through flexible buses.
At energy storing the movable electrode is shorted out with cathode section -it's the first stage. The movable electrodes are activated, when current runs maximal value and arc is burned between electrodes, it is step two. When movable electrode is drawn out of interelectrode space to the length equal the gap between stationary electrodes, the arc is extinguished by back current. The strength of vacuum gap is recovered for current zero time, it is the third step. Plasma is injected into the interelectrodes gap and current IS is shorted out through the PEOS after break of current zero. When the PEOS acted the whole vacuum gap changes over to state of magnetic self-insulation and energy from IS is transmitted to a load. It is the fourth step of combined switch work.

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F-43

DESIGN AND MANUFACTURE OF A PROTOTYPE NbTi FULL-SIZE JOINT SAMPLE FOR THE ITER POLOIDAL FIELD COILS

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The design of the Poloidal Field (PF) coils of the International Thermonuclear Experimental Reactor (ITER) relies on the use of 45 kA NbTi Cable-in-Conduit Conductors. An R&D programme is carried out jointly between CEA and ENEA to acquire knowledge on the behaviour of such conductors. In addition to the manufacture and tests of two subsize joint samples at CEA/Cadarache, the design of a Poloidal Field Full-Size-Joint Sample (PF-FSJS) has been carried out. After manufacture, this prototype sample will be tested in the Sultan test facility at Villigen and will serve as a model for the manufacture of joints for the PF coils .

The PF-FSJS is composed of two conductor legs made of stainless steel-jacketed NbTi full size cables, of a lower joint built according to the “twin box” concept developed at CEA and of two upper terminals to be connected to the facility current leads. Each leg is made with a specific NbTi strand: one is Ni-plated and the other contains an internal CuNi barrier. Both legs use conductors made with circular cables including 1152 0.81 mm diameter strands and jacketed with a square 316LN jacket.

The technology used for the manufacture of the terminals is derived from the experience gained in the manufacture of previous niobium-tin samples and in the manufacture of the Toroidal Field Model Coil busbars, which used niobium-titanium strands.

The paper describes the conceptual design of the sample, the techniques used for the different manufacturing steps of the conductor legs, the terminations and the assembly. It includes a survey of the instrumentation and an outline of the testing programme. The main modifications in the sample design, manufacturing steps and instrumentation with previous samples will be analysed and discussed. The quality inspection plan will be presented and the results of the manufacture commented. If available by the date of the conference, the first results of the tests at low temperature will be shortly presented.

A first step in the European development programme for the ITER PF coils is achieved with the manufacture and test of a prototype full-size sample, allowing to investigate as well the conductor behaviour as the joint behaviour. This development will bring qualification for the manufacturing techniques and controls to be used in the manufacture of conductor and joints of these large niobium-titanium coils.

F-44

MECHANICAL ATTACHMENT OF THE CONDUCTOR ENDS IN THE ITER POLOIDAL FIELD COILS

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The ITER PF (Poloidal Field) coils are wound from a large cable-in-conduit-conductor, the jacket of which is made of stainless steel. Tapered bonded tails, consisting of shaped steel profiles welded to the conductor ends, are used in these PF coils to mechanically attach the conductor ends to the winding pack. Their main function is to transfer the tensile force from the end of the outermost turn to the adjacent turns by shear through an appropriate thickness of insulating material (glass epoxy). These tails are embedded in the winding pack thus avoiding any local protrusion.

Similar tapered bonded tails have been extensively used in large copper coils. This design concept has been applied to the ITER PF coils. However, compared to a standard copper conductor, the tensile force to be transferred to the winding pack is larger in a ITER PF conductor because of the higher tensile stress experienced by the stainless steel jacket (average tensile stress up to 200 MPa). This has led to a new hollow tail design capable of transferring the large tensile force carried by the PF coil conductors on a length in the range 600–650 mm. Hollow tails, as opposed to solid tails, provide a larger bonded perimeter for the same effective tail cross section which also contributes to limit the peak shear stress in the insulation. Another possible design option based on a bi-metal solid tail concept is also described.

The tail geometry is obtained based on simple analytical tools and then validated by a detailed FEA (Finite Element Analysis).

In the first step, the shape of the tails is optimized for each of the PF coils through a 1-D analytical straight model so as to limit the insulation peak shear stress below the allowable. This 1-D model is the basis for a set of routines that solves the force balance between shear in the insulation and tension in the tail along its length. In the second step, a 3-D model is built based on the resulting geometry and a mechanical FEA is finally performed in order to validate the tail design.

The tail shape optimization process is described in detail as well as the resulting tail geometry of a typical PF coil. The detailed results of the mechanical FEA are compared to the expectation of the 1-D model thus allowing the validation of the 1-D model.

F-45

MEASUREMENTS OF NON-AXISYMMETRIC COIL-RELATED ERROR FIELDS IN DIII-D*

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Non-axisymmetric (error) fields in tokamaks lead to a number of instabilities including so-called locked modes [1] and resistive wall modes (RWMs) [2]. They can also cause errors in magnetic measurements made by point probes near the plasma edge and measurements made by magnetic field sensitive diagnostics also they violate the assumption of axisymmetry in the analysis of data. Most notably, the sources of these error fields include shifts and tilts in the coil positions from ideal symmetry, coil leads, and nearby ferromagnetic materials magnetized by the coils. New measurements have been made of the n=1 (and n=2) coil-related field errors in the DIII–D plasma chamber using a set of 24 probes measuring B_R,B_Θ, and B_Z at 8 locations equally spaced on a ring at R=1.64 m near the plasma major radius. This apparatus is similar to the one described in earlier measurements [3], but a number of improvements were made in the apparatus and technique to improve the accuracy of the measurements. These include: collecting a complete set of 8 data points for each field component, specializing to n=1 by using one probe of each field component as reference and subtracting the others from it, realigning the apparatus in situ, making measurements at three different elevations (–0.75 m, 0, and +0.75 m) inside of the plasma chamber, and rotating the apparatus through a large toroidal angle (65°) to identify systematic errors associated with the probes. During preparations for these measurements, a previously unrecognized large iron structure near the top of the machine, capable of causing substantial error fields (and present during the earlier measurements), was identified and removed.

Preliminary results of the measurements are that the centers of the poloidal field coils lie within a circle in the horizontal plane of radius 8 mm and are tilted from vertical by <0.2°. Both the center of this circle and the vacuum vessel are displaced from the center of the toroidal field by ~5 mm. The single largest displacement of a poloidal field coil from the axis of the toroidal field is 12 mm. The aggregate field errors of the poloidal field coils are about half as large as previously reported [3]. No new error field contributions associated with the toroid field coil were identified at the level of less than ±10 G (not previously reported). Non-linear effects due to coils operating together (e.g. the toroidal field coil distorting in the presence of a poloidal field) were not significant. These measurements failed to confirm the presence of an unrecognized substantial error field deduced from locked mode [1] and RWM [2] measurements. The difficulties of making these measurements in existing and new devices and potential sources of error including iron structures too small to affect the plasma will be discussed.

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F-46

DESIGN OF COMPACT HIGH FIELD TOKAMAK BASED ON VIRIAL THEOREM

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A novel tokamak device with a new type of toroidal field (TF) coils and a central solenoid (CS) whose stress is reduced to the theoretical limit determined by the virial theorem is designed. Recently, we had developed a tokamak with force-balanced coils (FBCs) [1] which are multi-pole helical hybrid coils combining TF coils and a CS coil. The combination reduces the net electromagnetic force in the direction of major radius by canceling the centering force due to the TF coil current and the hoop force due to the CS coil current. This excellent feature of FBC and its capability of tokamak operation were investigated and demonstrated by the first FBC tokamak "Todoroki-I" [2], while working stress in coils has not yet been investigated whereas the net electromagnetic force is reduced.

Therefore, we have extended the FBC concept using the virial theorem which shows that strength of magnetic field is restricted by working stress in the coils and their supporting structure. High-field coils should accordingly have same averaged principal stresses in all directions, whereas conventional FBC reduces stress in the toroidal direction only. In Ref. 3, we have obtained the poloidal rotation number of helical coils which satisfy the uniform stress condition, and named the coil as virial-limited coil (VLC). According to Ref. 3, VLC with circular cross section of aspect ratio A=2 reduces maximum stress to 60% compared with that of TF coils.

In order to prove the advantage of VLC concept, we have designed a small VLC tokamak "Todoroki-II" whose parameters are listed in Table 1 where the achieved values of Todoroki-I are also presented. The vilial-limited winding of Todoroki-II is illustrated in Fig. 1, in which a VLC is shown by darker hatch. Because the power supply of Todoroki-I and the same coil conductors are used, magnetic field strength and plasma current are restricted. The plasma discharge in Todoroki-II will be presented.

Table 1. Size and operation parameters.

Parameters	Todoroki-I	Todoroki-II
Major radius	0.297 m	0.300 m
Minor radius of FBC/VLC	0.115 m	0.140 m
Winding pitch of FBC/VLC	5	3
Pole number of FBC/VLC	8	8
Plasma minor radius	0.055 m	0.070 m
Maximum toroidal field at axis	0.90 T	1.55 T
Maximum plasma current	10 kA	40 kA

Fig. 1. VLC winding of Todoroki-II.

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F-47

QUENCH DETECTION & DATA ACQUISITION SYSTEM FOR SST-1 SUPERCONDUCTING MAGNETS

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The diagnostics of SST-1 Superconducting magnets will be in a noisy environment that makes quench detection and data acquisition difficult. The presence of high voltages induced in the magnets during off-normal events like VDE, plasma disruption, and PF magnet ramp ups, along with a large number of channels has made quench detection and data acquisition a challenging task. A hybrid of analog electronic circuits and software controlled data acquisition system has been designed to safeguard the magnets. The paper will describe the electronic hardware circuits developed for high voltage suppression, fail proof quench detection system, along with the noise elimination algorithms.

The SST-1 Superconducting magnets will have large number of sensors like venturi flow meter, strain gages, hall probes, pressure sensors, temperature sensors and voltage taps. A real time data acquisition system has been designed using VME bus for monitoring and storing signals from all these sensors and initiating control action in off-normal events.

The paper will describe the hardware developed for quench detection and its integration with the data acquisition system, along with the overview of the data acquisition system designed for SST-1 Superconducting magnets.

F-48

CONSTRUCTION OF TOROIDAL FIELD MODEL COIL FOR HT-7U*

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The Chinese National Fusion Research Project---The HT-7U superconducting tokamak physics experiment device has been approved by Chinese Government. The HT-7U Superconducting Tokamak is an advanced steady-state plasma experimental device to be built in ASIPP.

On-site winding of magnet coils of HT-7U started in September 1998, using a numerically control winding machine. Magnet coils are all forced-flow superconducting coils, which are designed to satisfy several criteria regarding stability margin, well-cooled regime and protection. Magnet coils consist of superconducting toroidal field (TF) coils and superconducting poloidal field (PF). All of TF coils will use NbTi Cable-in-Conduit Conductor (CICC) cooled with supercritical helium. This paper reports on set up of a new winding facility with unique capabilities for continuous winding long length CIC conductor. Analytical method used to predict conduit springback before winding is present in comparisons to results obtained during winding. The principle of winding procedure has been successfully demonstrated. A winding method was made by trial of winding. The TF Dummy Coil has been fabricated in 1999 and the fabrication of all full TF Dummy Coil was finished in early this year.All-full TF Model Coil is under way and will finished in May of this year.

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F-49

DESIGN AND ANALYSIS OF KSTAR PF MAGNET STRUCTURE

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The Poloidal Field (PF) magnet system of the Korea Superconducting Tokamak Advanced Research (KSTAR) consists of three pairs of outer PF coils, PF5, PF6 and PF7. Each coil using cable-in-conduit conductors is cooled by forced super-critical helium. The conductor of PF5 is Nb_3Sn superconductor with Incoloy 908 conduit, whereas the PF6 and PF7 conductor is NbTi superconductor with SS 316LN. The PF coils are arranged symmetrically with respect to the equatorial plane.

All PF coils are connected to the TF coil structure by links and pins that can sustain the gravitational and electromagnet loads with the capability of allowing relative radial movement during cool down. The flexible plates type supports are also carefully considered as an alternative design of the link and pin type support system in order to solve the functional problems caused by misalignment and backlash of the pin. But the flexible plates type support has the disadvantages such as low buckling strength and difficult installation.

Several three-dimensional finite element models for the PF coils and structures have been developed to investigate the structural characteristics and reliability of the PF magnet system. To obtain more accurate analysis results for the PF coils, the detailed analyses, the global and local analyses have been performed applying the sub-modelling technique. The major loads acting on the PF coils are dead weight, assembly forces, thermal load due to cool down, and electromagnetic forces under the reference scenario operations and the maximum vertical force condition. Under the most load cases, the stresses in PF5 are higher compared to those in other coils, and the critical load case is the maximum attractive force condition. The strength of the PF structure complies with the requirements of the structural design criteria of the KSTAR magnet system.

F-50

STRUCTURAL ANALYSIS OF THE KSTAR CS MAGNET SYSTEM

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The Korea Superconducting Tokamak Advanced Research (KSTAR) device is a steady-state-capable experimental fusion device with fully superconducting magnet system. The Central Solenoid (CS) magnet system of the KSTAR consists of four pairs of superconducting coils compressed with a CS coil structure. The CS coil structure is operated as a pre-load structure to give an axial compression on the CS coils. The required axial compression is applied partly by using the pre-compression components at room temperature and by the thermal contraction difference between the coil winding pack and the structure as well.

In order to investigate the structural integrities and to build the structural reliabilities of the KSTAR CS magnet, global and local structural analyses of the CS coils and associated structures have been conducted for various operation scenarios by developing three-dimensional finite element model. The model consists of eight winding packs with ground wrap, insulations, top and bottom blocks, bearing plates, wedge, tie plates, flexible joints, and toroidal ring. Orthotropic smeared mechanical properties have been used for the winding pack consisting of multi-phase material. The extreme contact conditions are considered such as the fully bonded and sliding without friction between contact surfaces. The electromagnetic loads are applied based on 225 plasma equilibrium states as the applicable operation scenarios.

From the analysis results, it has been found that the CS magnet system can safely withstand the reference scenario operations, and the structural strength of the CS structure complies with requirements of the KSTAR magnet system. The critical conditions on the preload occur at room temperature because the allowable stress of the metal at room temperature is much lower than that at 4K. All analyses have been performed for both bonded and sliding conditions at the contact surfaces between the CS coils and structures. The bonded condition gives more conservative results for all load cases. The sliding could occur at the interface of bearing plate, insulation, and winding pack for some load conditions.

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STRUCTURAL ANALYSIS OF THE KSTAR TF MAGNET SYSTEM

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The Toroidal Field (TF) magnet system of Korea Superconducting Tokamak Advanced Research (KSTAR) device consists of 16 superconducting coils enclosed in the steel cases. The TF magnet structure protects the cable-in-conduit conductor from mechanical, electrical, and thermal loads, and also supports three pairs of the Poroidal Field (PF) coils as well as four pairs of Central Solenoid (CS) coils.

A three-dimensional finite element model including winding pack, insulation and filler material has been developed using solid brick elements, and a hybrid model with beam-shell elements has been also built for the verification purposes. To investigate the system integrity and to build the structural reliability of the KSTAR TF magnet system, linear global analysis, local analysis for the high stress region, and detailed analysis for the conductor jacket of the winding pack have been carried out. The fatigue analysis based on the linear elastic fracture mechanics has been also performed according to the structural design criteria of the KSTAR magnet system.

The results reveal that the maximum stress intensities of the TF magnet structure are below the allowable stress limit, and it can safely withstand the reference scenario operations. In addition, the design loads can be determined for the detailed structural design of other supplementary components.

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INFLUENCE OF REACTOR IRRADIATION ON THE MECHANICAL BEHAVIOR OF ITER TF COIL CANDIDATE INSULATION SYSTEMS*

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The International Thermonuclear Experimental Reactor (ITER) fusion device is based on the magnetic confinement principle to control the high temperature plasma. These superconducting fusion magnets require a high material performance due to exposure to neutron and γ-radiation and to high mechanical stresses at cryogenic temperatures. In particular, radiation induced effects in the insulating materials for the windings of the ITER Toroidal Field (TF) Coil have been identified as an area of concern for an undisturbed reactor operation over the plant lifetime.

Basically, these insulation materials are glass fiber reinforced plastics (GFRPs) consisting of different fibers and organic matrix materials (e.g. S-and R-glass multi-component epoxy resins) and are composites with excellent mechanical behavior (e.g. high stiffness and high radiation resistance). Both candidate insulation systems fabricated by European industry consist of R-glass fiber reinforced tapes, vacuum impregnated in a DGEBA epoxy system. In addition, Kapton H-foils were integrated as electrical barrier materials.

The static mechanical and the fatigue properties of these GFRPs have to fulfill the ITER-FEAT design criteria. In order to assess the mechanical material performance at the magnet location, the systems were irradiated at ambient temperature (~ 340 K) in the TRIGA reactor (Vienna) to neutron fluences of $5x10^{21}$ and $1x10^{22}$ m^{-2} (E > 0.1 MeV). Tensile (ASTM D638 and DIN 53455), short-beam-shear (SBS) (ASTM D2344) as well as double-lap-shear (DLS) tests were carried out at 77 K prior to and after irradiation. Furthermore, stress versus cycle-to-failure diagrams (stress-lifetime diagrams, “Wöhler-curves”) were assessed for the tensile and the DLS-specimens under tension-tension fatigue load up to 10^6 cycles.

The extensive test programs show that the mechanical strength of both insulation systems is strongly influenced by the winding direction of the tapes and the radiation induced delamination process, which is documented impressively by photographs. Furthermore, the ultimate tensile strength (UTS) and the interlaminar shear strength (ILSS) degrade significantly after irradiation to the ITER-FEAT dose level of $1x10^{22}$ m^{-2} (E>0.1 MeV). In addition, the radiation damage causes swelling accompanied by the formation of bubbles inside the fiber composites.

*This work has been carried out within the association EURATOM-OEAW.

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RADIATION HARDNESS OF NEWLY DEVELOPED ITER RELEVANT INSULATION SYSTEMS

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The application of glass fiber reinforced plastics (GFRPs) as insulating materials for the superconducting (SC) magnet coils of the International Thermonuclear Experimental Reactor (ITER) fusion device imposes high demands on the material performance. In general, the radiation spectrum and the mechanical stresses due to the pulsed Tokamak operation lead to critical conditions, which significantly influence the mechanical behavior of the magnet insulation. Therefore, GFRPs with improved mechanical and electrical properties are of special interest. Recently, advanced fiber composites were introduced based on technologies originally developed for aerospace or cryogenic engineering.

One of the most promising matrix materials for higher radiation resistance is the cyanate ester system. This resin has become of major interest for insulation materials, especially due to the low moisture uptake and its stability at low and high temperatures. Thus, various innovative GRFPs with boron-free R-and S-glass fiber reinforcements and cyanate ester based matrix compositions, partly inter-leafed with Kapton H-foils, were manufactured by European and US industry.

In order to assess their radiation hardness, the specimens were irradiated at ambient temperature (~ 340 K) in the TRIGA reactor (Vienna) to neutron fluences of 1, 5, 10 and $50 \times 10^{21} m^{-2}$ (E > 0.1 MeV). Mechanical static tests were carried out at 77 K in tension (ASTM D638 and DIN 53455) as well as in the interlaminar shear mode using the short-beam-shear (SBS) test (ASTM D2344) and the double-lap-shear (DLS) test before and after fast neutron irradiation. To obtain more information about the radiation damage and the lifetime of these GFRPs, stress versus cycle-to-failure diagrams (stress-lifetime diagrams, "Wöhler-curves") were assessed for the tensile and the DLS-specimens under tension-tension fatigue load up to 10^6 cycles. In addition, swelling in the through-thickness direction and the weight loss were investigated prior to the mechanical measurements.

Both static and dynamic experiments show good material performance with minor degradation of the ultimate tensile strength and the interlaminar shear strength up to the ITER-FEAT design neutron fluence of 1×10^{22} m^{-2} (E > 0.1 MeV). No significant influences on swelling and weight loss were observed.

*This work has been carried out within the association EURATOM-OEAW.

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TECHNOLOGICAL DEVELOPMENT OF THE RFX ENERGY TRANSFER, PROTECTION AND SWITCHING SYSTEMS

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In the framework of the recovery activities regarding the Energy Transfer, Protection and Switching Systems - severely damaged by the blaze which occurred to the RFX power supply system at the end of 1999 - the former design has been modified in order to take the maximum advantage from both technological development and operation experience. Furthermore, the issue of the improvement of the safety level has been always considered in the design modifications.

The most significative modification is the replacement of ignitron valves with Light Triggered Thyristors (LTTs) for HVDC application. In fact, the progressive increase of thyristor performances makes them competitive with respect to ignitron for high current - high charge applications (typical in fusion experiments). This replacement improve also the system safety; in fact, being the ignitron technology based on low-pressure arc inside mercury vapor, faults of this component can produce mercury release in the environment, leading to serious safety problems. On the other hand, thyristors are much more sensitive to fast electrical transients (e.g. voltage and current derivative limits) than ignitrons; for this reason, detailed calculations have been carried out to design an efficient protective network for all normal and fault conditions.

The thyristor switches for all the systems are made of series-parallel connections of a unique type of components, rated for 7500V–3750A. In particular, a six-series/two-parallel LTT switch has been adopted for the Energy Transfer units and Protection units; for the Switching units a switch made by hybrid thyristor-diode series of four components has been adopted. Important modifications have been introduced also in the control and monitoring system. In particular, the fast control electronics, formerly based on Siemens C2-Simatic system, is now based on a PLD (Xlink FPGA Spartan 20XL) system. The measuring system has been completely changed, moving from a fiber optic analogue transmission based on voltage to frequency conversion, to a digital fiber optic transmission, where receiving modules have built-in programmable comparators used for fault discrimination; this solution increases the reliability of the protective action performed by the PLD system. Finally, the slow PLC-based control has been upgraded from Simatic S5 to Simatic S7, using the standard bus Profibus DP for the communications with the peripherals. The human interface has also been improved, moving from synoptic panel to a visualization system based on WinCC software running on a PC670 computer.

All systems are presently undergoing the erection phase while the commissioning phase is foreseen from June through September 2002. It has to be underlined that the commissioning will be oriented also to the realization of the ITER Bypass switch and Vacuum Tube combined tests, to be carried out within the end of 2002.

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WEDNESDAY 11th September

TEST RESULTS OF THE ITER TOROIDAL FIELD MODEL COIL EXPERIMENT IN THE TOSKA FACILITY OF THE FORSCHUNGSZENTRUM KARLSRUHE

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The ITER Toroidal Field Model Coil (TFMC) is one of seven large projects of the ITER Engineering Design Activity (EDA). The ITER TFMC is a Nb_3Sn superconducting coil designed and constructed by the European Home Team (EUHT) in collaboration with European industry under the leadership of EFDA / CSU (European Fusion Development Activity / Close Support Unit), Garching. The shape of the TFMC is a racetrack coil representing the design principles of the ITER TF full size coils. The aim of the project is the confirmation of the feasibility of manufacture and the proper function of the ITER TF coil design principles.

The TFMC is being tested in two phases in the TOSKA facility. In phase I, the TFMC was tested as single coil up to its rated current of 80 kA. In phase II, the coil will be tested in 2002 in the background field of the EURATOM LCT coil in order to achieve relevant mechanical stresses. The TOSKA facility was extended concerning its cryogenic and high current supply system as well as control and data acquisition system performance for that test within an ITER Task Agreement. The test program was prepared within the EUHT by code development and accompanying experiments for getting the input data needed for calculations in the areas of electromagnetic, thermohydraulic, mechanical and dielectric insulation properties. In each area suitable codes are now available from which predicted values have been deduced.

The TFMC was successfully cooled down in two weeks to its operating temperature. After zero current checkouts (leak test, flow distribution, sensor continuity, high voltage insulation) and low current checkouts of the proper operation of the interlocks, the quench detection system, the power supply and the safety discharge circuit, the current operation was started. Afterwards, the current in the coil was ramped up in steps of quarters of the stored magnetic energy. Each current level a slow ramp down, a controlled inverter mode discharge (i.e., negative voltage applied to the load) and a safety discharge (i.e., exponential current decay) was performed to become acquainted with the behaviour of the whole system in normal and emergency conditions. The rated current of 80 kA was achieved without any unexpected behaviour in July 2001. That is the highest current level ever achieved for a large superconducting magnet up to now and has exceeded the ITER FEAT TF current level by about 19 %. The resistance of all windings and bus bar joints was between 1 and 2 nΩ as expected. The mechanical stress level and coil deformations were in fair agreement with the finite element calculations. One pancake was intentionally quenched by increasing its inlet temperature by stepwise increase of the heating power of a gas heater optimised by the code Multiconductor MITHRANDIR. The determined current sharing temperatures have met the predictions from the design criteria within the measurement error bar. The TFMC was quenched 6 times at 80 kA. No degradation of the current sharing temperature was noticed. The investigated thermohydraulic behaviour and AC losses fitted well to the predictions.

The main conclusion from the first test phase is that the engineering design principles and manufacturing procedures are sound and suitable for the ITER TF full size coils. Presently, the TFMC test is going ahead with preparation for test phase II in second half of 2002.

IN-10

ANALYSIS OF DC TEST RESULTS OF THE ITER TOROIDAL FIELD Conductor Insert-coil experiment

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The ITER Toroidal Field Conductor Insert-coil (TFCI) is a Nb3Sn superconducting single layer coil designed and constructed by the collaboration of Russian enterprises (NIIEFA, VMIINM, VNIIKP, SC "Izora Plant") headed and co-ordinated by the Efremov Institute (NIIEFA). In the interim July–October 2001 the TFCI was tested in the ITER CS Model Coil (CSMC) by the team of the Japan Atomic Energy Research Institute (JAERI), Naka, Japan in collaboration with the International CSMC Test Group. All the work was a part of the ITER Model Coil Program carried out jointly by the participants of the ITER project from EU, Japan, Russia and USA to support the Engineering Design of the Superconducting Magnet System of the ITER tokamak reactor. The main research goal of the TFCI project was to validate performance of significant length of the full size conductor under relevant conditions of the magnetic field, current and mechanical strain. The present work is devoted to discussion of DC test results.

The TFCI has reached its specified operating point – the magnetic field of 13T and conductor current of 46 kA – just in the very first energizing campaign, and demonstrated stable adequate performance under the ITER operating conditions also after the lifetime cyclic and quench tests.

However some of the design margins appear somewhat less than originally expected. A comparison between the witness strand and the TFCI performance has revealed that there is a difference in the current sharing temperature TCS for strands and the coil, and that the n-factor for the coil is a half of that for the strand. The difference in Tcs (ΔTCS=TCSstrand-TCScond) depended on the coil test conditions (applied field, operating current) and there was recorded a variation of ΔTcs along the TFCI conductor: the highest ΔTCS was measured near upper terminal joint and the helium outlet, the lowest ΔTCS – near lower terminal, and helium inlet. Besides, the Tcs values, estimated from "up" (conductor heating) and "down" (conductor cooling) branches of Voltage-Temperature characteristic (VTC) during the low current tests, were not the same.

There could be several explanations for the effects observed. This paper considers a non-uniform current distribution (and re-distribution) as a possible cause of Tcs "degradation" and "hysteresis-type" behaviour of the VTC. The source of such a current "non-uniformity" could be the upper TFCI terminal joint, which was repaired during manufacturing and shows higher resistance compared with the lower one.

FO-7

STATUS OF FABRICATION OF THE SUPER-CONDUCTING COILS FOR WENDELSTEIN 7-X

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The fusion experiment WENDELSTEIN 7-X (W7-X) continues the line of stellarator experiments at IPP. To allow steady state operation the magnet coils of W7-X are superconducting. The 50 non-planar and 20 planar coils of W7-X are grouped in five equal modules, each consisting of two mirror symmetric half modules. The half modules are assembled from five different non planar coils, two planar coils and a sector of the coil support structure. All cryogenic parts are enclosed in a cryostat to protect them from ambient temperature.

The coil system of W7-X is currently manufactured in industry and uses the advanced conductor. The advanced conductor has a cross section of 16x16 mm² and consists of 243 copper stabilised NbTi strands which are enclosed in an aluminium jacket. The nominal current of the non-planar coils is 17.6 kA and of the planar coils 16 kA. Manufacture of the coils includes winding of the conductor, impregnation of the winding package and embedding it in a steel casing, connection of electric joints with a resistance of less than 1 nΩ and helium cooling pipes which are electrically insulated from the conductors and mounting of instrumentation to measure temperatures, electrical potentials and mechanical tensions. Whereas the steel casings for the non-planar coils are cast as half-shells and welded along the perimeter the casings for the planar coils are made from plate material and joined by screws. The coils are cooled by circulating supercritical helium both through the void in the cable-in-conduit conductor and through tubes which pass around the casing. Copper cladding of the coil casings improves the heat conduction across the casings to the helium coolant.

Manufacture of the 50 non-planar and 20 planar coils is in process. The first winding packages have been fabricated and delivery of the first non-planar coil to the cryogenic test facility at CEA/Saclay is expected for autumn 2002. Fabrication of the coils is presently considerably delayed due to difficulties in superconductor production.

The presentation will give an overview on the status of coil fabrication and will describe details of the production of the conductor, qualification of single components and the design of the improved cooling of the coil casings.

FO-8

SUMMARY, ASSESSMENT AND IMPLICATIONS OF THE ITER MODEL COILS TEST RESULTS

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The ITER model coils consist of the Central Solenoid Model Coil (CSMC) and 3 associated insert coils, the Central Solenoid Insert (CSI), the Toroidal Field Insert (TFI) and the Nb3Al Insert (NALI) and the Toroidal Field Model Coil (TFMC). The CSMC and associated inserts produce a high field (~13T) and were designed to qualify the ITER Nb3Sn conductors at and beyond the ITER operating conditions. The stored energy of the system is about 640MJ. The TFMC was intended primarily to qualify the manufacturing procedures for the TF coils, and operates up to about 8.5T with a stored energy of about 80MJ.

Testing of the coils is close to completion, with 3 test campaigns since 2000 on the CSMC and inserts lasting a total of about 5 months and one on the TFMC in 2001 lasting about 2 months. A further test of the TFMC this year and the fourth test of the CSMC with a NbTi PF insert next year are outstanding. The main task remaining is the assessment of the results and the implications for the ITER conductor/coil design (including where necessary modifications to the design criteria and the design itself and further R&D). This assessment has been supported by extensive testing on subscale components (such as strand and conductor samples), in several cases initiated as a result of the model coil test results themselves.

The assessment of the model coil results has concentrated in the area of the conductor performance, since the coils have allowed the conductor design margins to be assessed. The coils have also qualified most of the manufacturing processes proposed for the ITER coils themselves, as they performed completely satisfactorily as regards the electrical and structural characteristics. Assessment of the conductor margins is a critical issue as in ITER a minimum cost, maximum current density conductor is required. To achieve this, margins must be kept to the minimum required to cover performance uncertainties and operational disturbances. The main issues assessed have been conductor temperature margin, AC losses, stability, thermohydraulic characteristics, quench behaviour and sensitivity to cyclic magnetic loads, as well as joint resistance and stability to pulsed fields. The temperature margin is the main design driver and has been considered in detail, using computer simulations of the current distributions in the conductor and the heat transfer processes between conductors.

The assessment has indicated that, although the conductor performs very close to expectations regarding the AC losses, the transition of the conductor from superconducting to resistive behaviour does not. It occurs more gradually, and at a lower temperature, than expected from the strand results. There are also indications that the local magnetic loads on the conductor may be influencing this transition. To recover adequate margins, possible design modifications include the use of a Titanium jacket for the conductor (as used in the TFI) to reduce the performance drop due to differential thermal contraction between strand and jacket, or, if a steel conductor is retained, the use of a void fraction in the cable of 32% instead of the 36% in the model coil cables.

FO-9

DEVELOPMENT OF ACTIVE CONTROL SYSTEMS ON ASDEX UPGRADE IN VIEW OF ITER DISCHARGE SCENARII

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Active control of tokamak discharges has always been an important tool for performance optimisation. For next-step devices such as ITER, it is clear that many discharge parameters will have to be controlled simultaneously. In addition to the established feedback control of the magnetic geometry and simple plasma parameters such as line-averaged density, present research also includes active control of MHD stability and confinement quality. Although this can be achieved by control of both pressure and current profile, the large amount of self-heating in ITER will allow only limited control of the pressure profile. Therefore, active control of the current profile has recently received a lot of attention.

On ASDEX Upgrade, we study both conventional and advanced scenarii as candidates for ITER. While control of the global current profile seems to be necessary in the advanced scenario to maintain ITBs, the conventional ELMy H-mode mainly calls for current drive to control the MHD stability. This can be achieved by local current drive around the resonant surfaces of interest. Both scenarii need control of global quantities such as density and stored energy and therefore, in a next-step device, fusion power, but also control of the power exhaust, e.g. by injection of radiating impurities. In addition, study of control or mitigation of disruptions is an important point in ASDEX Uprade, too.

ASDEX Upgrade has a wide variety of control tools available. For heating and current drive, 8 MW of ICRH, 2 MW of ECRH and 20 MW of NBI, 5 MW out of these with tangential injection for optimum off-axis current drive, are available. Fuelling can be done with a number of gas valves, but also with a high-speed (up to 1200 m/s) centrifuge pellet injector with high repetition rate (up to 80 Hz). All these tools will be discussed in detail with respect to their suitability for active control. Recent highlights from plasma control will be shown, in particular the application of ECRH to stabilise MHD modes such as NTMs and sawteeth and the control of the ELM frequency by pellet injection.

The complex interlinks between the quantities that have to be controlled requires a sophistiated control system. We will present the ASDEX Uprade approach based on a fully digital control system that directly interacts with various real-time diagnostics. In this approach, the control system also knows about the plasma state and can, depending on which state it is in, adapt its control strategy. Examples for this 'regime recognition' will be presented. Scenarii for future feedback control of e.g. the current profile will also be discussed.

IN-11

ADVANCED TOKAMAK OPERATION USING THE DIII-D PLASMA CONTROL SYSTEM*

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The principal focus of experimental operations in the DIII-D tokamak is the advanced tokamak [1] regime, which will require highly integrated and flexible plasma control to achieve. In a high performance steady-state advanced tokamak, accurate regulation of the plasma boundary, internal profiles, pumping, fueling, and heating will need to be well coordinated with MHD control action to stabilize such instabilities as tearing modes and resistive wall modes. Sophisticated monitors of the operational regime must provide detection of off-normal conditions and trigger appropriate safety responses with acceptable levels of reliability. Many of these capabilities are presently implemented in the DIII-D plasma control system (PCS), and are now in frequent or routine operational use. The present work describes recent development, implementation, and operational experience with AT regime control elements for equilibrium control, MHD suppression, and off-normal event detection and response. The highly flexible DIII-D PCS allows rapid development and testing of such subsystems, and is also in use at both NSTX and MAST.

A central characteristic of AT plasma regimes is the strongly shaped plasmas which must be accessed and the high degree of accuracy with which they must be regulated. The DIII-D isoflux shape control system [2] has demonstrated that it can produce a wide range of shapes including AT-relevant high triangularity, high elongation, double- and single-null diverted configurations. Density has been controlled by regulation of pumping via strikepoint placement relative to pumping plenums. Beam power regulation has allowed fine control of stored energy, and impurity gas puffing has been regulated to control radiated power.

Another key characteristic of the AT regime is active suppression of MHD instabilities. The DIII–D PCS has stabilized the neoclassical tearing mode (NTM) by controlling the relative location of gyrotron ECCD deposition and the mode islands. This suppresses the modes by replacing the "missing" bootstrap current that characterizes the instability. PCS algorithms for feedback control of nonaxisymmetric radial field coil currents based on measurements of n=1 magnetic signals have stabilized the resistive wall mode (RWM) beyond the no-wall β-limit.

Because AT plasmas operate close to stability limits in order to yield high efficiencies, off-normal events can conceivably produce large excursions from the operating point and result in uncontrollable instability. Monitors have been developed to detect the impending onset of such instabilities and take corrective or mitigating action. The use of such monitors has allowed recognition and early termination of discharges with performance-limiting instabilities, as well as triggering of high-pressure gas injection to almost completely mitigate disruption effects.

[1] D.A. Humphreys, et al., "Design of a Plasma Shape and Stability Control System for Advanced Tokamaks," Proc. 18th Symp. on Fus. Technol., Karlsruhe, Germany, August 1994, **1** (Elsevier Science Publishers B.V., Amsterdam, 1995) 731.
[2] J.R. Ferron, et al., Nucl. Fusion **38** (1998) 1055.

DO-10

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CONTROL SYSTEM OF WENDELSTEIN 7-X EXPERIMENT

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WENDELSTEIN 7-X is a superconducting stellarator under construction and capable of running pulses of up to 30 minutes duration with full heating power. The control system of W7-X will support all discharge scenarios compatible with this capability, i.e. short pulses with arbitrary intervals, steady state discharges, and arbitrary sequences of phases with different characteristics in one discharge. These phases are called “segments” and represent the basic temporal unit of a discharge. The use of segments substantially reduces the time required for parameter and diagnostic scans and permits short tests of new settings without interfering with the main programme.

A hierarchical layout of the control system will reflect the structure of the experimental device. Each technical component and each diagnostic system including its data acquisition will have its own control system permitting autonomous operation for commissioning and testing. The activity of these devices will be co-ordinated by a master controller during the experimental sessions or when co-ordinated activity is required.

The component control systems as well as the master controller will make use of industrial PLCs and computers running a standard operating system or a real time operating system. Industrial field busses, several Ethernet networks and a proprietary unidirectional fibre network of the “TTE” (trigger, time, event) system [1] will distribute most of the information required for control purposes including the precise time, event messages and measured data in real time [2].

The real time computers will run cyclically at a kHz frequency synchronised by the TTE-system. System parameters and algorithms as far as they are relevant for the experiment, will be exclusively controlled by complex software objects. By synchronous changing references to these objects in all computers the whole system behaviour can be changed from one cycle to the next. This allows to switch between segments or to the end of the discharge. The switching may be determined by fixed timing, by logical conditions or by operator action. All data required for the construction of the software objects will be stored in one central SQL database. With the information in the database the objects are constructed in the control computers well before they are required. The operator of the experiment will trigger the construction of the objects, chain segments, start execution of a chain, re-order or supplement chains during execution and stop the discharge.

[1] J. Schacht, H. Niedermeyer, Chr. Wiencke, J. Hildebrandt, and Wassatsch A., "The Trigger-Time-Event System for the W7-X experiment," in Proc. 12th IEEE-NPSS Real *Time Conf*, 2001, pp. 240–244.

[2] H. Laqua, H. Niedermeyer, I. Willmann: Ethernet based Real Time Control Data Bus. To be published in: *IEEE Trans. Nucl. Science.*

DO-11

TRENDS IN COMPUTING SYSTEMS FOR LARGE FUSION EXPERIMENTS

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The major European fusion facilities, such as JET, Tore Supra and Asdex Upgrade, have a 30-year history from conception to the present day. This period has seen a rapid evolution in computer techniques, and Fusion has been in the vanguard of this development. Experience has shown that IT architectures and standards adopted early in the construction phase of a project are carried into the main programme and affect the project for its entire lifetime. In this paper we select a few of the more important computer architecture lessons we have learned in Europe, particularly with respect to the EFDA-JET facility, and we endeavour to make some recommendations towards the next generation of machines.

The subjects discussed will be:

- Documentation Infrastructure: Many products are in use around Europe, none of which are entirely satisfactory.
- Subsystem Interface Specification: Required early in the design phase, yet flexible to state-of-the-art developments.
- Integrated Control and Feedback: Of increasing importance in the last ten years, and paramount to the flexibility and stability of the plasma.
- The Central Interlock and Safety System: The dual-hardware CISS system at JET shuts down to a safe state in case of faults. However the triple-system approach as used in the aviation industry may be more appropriate for the next generation of machines.
- The changes required towards "steady-state" operation: The traditional separation in Fusion between the pulse-related architecture of the data and the continuous machine control will require merger, with far reaching consequences.
- Computer and machine security, yet transparency of information: JET has demonstrated that it can be licensed as a nuclear installation, yet provide full transparency of access both from offices and off-site.
- The site external network connection: Of increasing importance, we will discuss the connection to the world research networks.
- Options for "Remote Participation" – Drawn from the experience of the EFDA and EFDA-JET Remote Participation projects.

We will also discuss the routes taken in some non-Fusion research areas such as the high energy physics community and nuclear fission. We will further mention our experience in the choice between "in-house" computing versus "external contracts". Finally we note that important choices will be required very early in the construction phase of the next project and we give a rough timetable of our estimation of key decisional dates for computer architecture topics.

This work has been performed under the European Fusion Development Agreement.

EO-12

THURSDAY, 12th September

Poster session P3
Chairmen: Dr. Bahn, W. and Prof. Rintamaa, R.

Poster presentations

SURVEY OF IN-VESSEL CANDIDATE MATERIALS FOR FUSION POWER PLANTS. THE EUROPEAN MATERIALS R&D PROGRAMME

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The long term development towards fusion power plants (DEMO and a PROTOTYPE reactor) aims for materials which can withstand high neutron wall loading and heat fluxes, high fluences and coolant pressure conditions at temperatures attractive for efficient thermodynamic working cycles. In addition the materials should be “low activation” type to maintain one of the most attractive features of fusion.

The European Long Term programme is focussed on the R&D of the materials and fabrication technologies (joints, HIP, coatings) needed for the DEMO reactor. They should allow a reasonable extrapolation of the present technologies in order to be developed and demonstrated for the DEMO design in the time frame of ITER. The Reduced Activation Ferritic Martensitic (RAFM) steel EUROFER is the primary choice for first wall and breeding blanket structural application. Two breeder materials are presently considered: the Pb-17Li and the lithium orthosilicate or metatitanate pebble beds breeders. The test of DEMO relevant modules of breeder blankets in ITER will guide the choice for DEMO. The most probable plasma facing material for high heat flux components (thermal wall loading > 1MW/m2) will be a W alloy. For the divertor heat sink high conductivity Cu-based materials for water cooling and W alloys for helium cooling are investigated (design heat load of 15 and 10 MW/m2 respectively, have been assumed). In order to improve the thermal efficiency in advanced fusion power plants the maximum operating temperature of the structural materials should be increased. In the European programme a limited effort is dedicated to the development of ODS-RAFM steel with the objective to increase the maximum temperature from 550 C (EUROFER limit) to 650 C or higher. In addition SiCf /SiC ceramic composite is considered as a promising alternative for a very advanced power plant concept operating at very high temperature (1000 C). The characterisation of relevant materials under power plant neutron irradiation condition is a key issue. An intense neutron source with fusion relevant neutron spectra is needed. IFMIF has been proposed at international level as the best solution to meet fusion requirements. Modelling and numerical tools are essential for the analysis of the neutron defect production, accumulation and microstructural evolution of materials and to qualify small scale test technology. These tools should allow the comparison and the extrapolation to power plant conditions of the irradiation test results obtained in different types of neutron sources (fission reactors, spallation source, IFMIF).

The paper will include a short description of the European material development programme strategy, the ongoing R&D activities and the main results obtained so far.

IN-12

THE DEVELOPMENT OF EUROFER REDUCED ACTIVATION STEEL

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High energy, 14 MeV, neutrons from the plasma of a TOKAMAK type fusion power plant affect the first wall, blankets, and divertors. The structural materials in such components will show high displacement damage, and contain high levels of helium and hydrogen. Conventional austenitic stainless steels will swell unacceptably and suffer from helium embrittlement. Ferritic martensitic steels show limited swelling and are less susceptible to helium effects, which made them a first candidate for the development in the EU with conventional 9–12 Cr steels. In the IEA framework the US, Japan and the EU took the second development step including the change to steels with chemical compositions that lead to the class of reduced activation ferritic martensitic, RAFM, steels. The IEA results paved the way for the EUROFER97 specification and subsequent production by an EU supplier.

This steel shows attractive mechanical properties also after long ageing times, provided the heat treatment has been properly carried out. The designer is further supported by a wide selection of manufacturing routes including different types of fusion welds: tungsten inert gas, electron beam welding, and diffusion welding. The feasibility of hot iso-static pressing, HIP, of both solid parts and compacted powder has been demonstrated to result in objects with attractive mechanical properties. The compatibility of EUROFER 97 has been shown satisfactory in water of PWR like quality. The behaviour in flowing Li-Pb of blanket quality will have to be characterised more in detail in appropriate loops. From preliminary test results it is observed that the susceptibility for hydrogen embrittlement is strongly dependent on temperature. Additional work will have to confirm the longer term implications of the presence of hydrogen. The early results of EUROFER97 irradiation effects confirm that the ductile to brittle transition temperature, DBTT, shifts to higher temperatures: around RT. The DBTT of other laboratory scale heats with near EUROFER compositions stay below 0 degrees C. There is thus still room for improvement of the radiation resistance of the EUROFER steel type. The integrated results of the EUROFER97 development program form the basis for the next step towards improved chemical compositions and heat treatments for this class of steel. EUROFER97 steel has also been converted into powder to allow the alloying with yttrium oxide. The test results serve the development of Oxide Dispersion Strengthened, ODS, RAFM steel that promises a 100 K higher operating temperature.

Recent EUROFER97 results show that an RAFM steel can strike an attractive balance between strength and toughness for near plasma components. The environmental effects such as neutron irradiation and hydrogen on mechanical properties, will be characterised with fission neutron simulations up to levels representative for end of life conditions. ODS RAFM steels are in a very early stage of development, but the preliminary results are promising enough to continue this path to higher operating temperatures. For the simultaneous effects of neutron damage and helium generation modelling must enhance the prediction for end of life properties until IFMIF will be available for experimental verification.

JO-13

NUCLEAR ASPECTS OF MOLTEN SALT BLANKETS

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Molten salt is an attractive liquid blanket material due to its much less chemical reaction concern than lithium when exposed to water and air, and its significantly reduced MHD effect when used as the coolant in a tokamak reactor. This paper examines the nuclear aspects of two molten salt blankets: FLIBE - a mixture of LiF and BeF2 (2 to 1 ratio in molecules), and FLINABE - a mixture of LiF, NaF, and BeF2 (1:1:1 ratio in molecules). The blanket model used in the study is that for the ARIES-RS power plant. The structural material investigated is the reduced activation ferritic steel. Tritium breeding, nuclear heating rate, neutron activation in the molten salt/coolant and in the structural material, and nuclear transmutation are among the nuclear aspects to be considered. Related issues considered include corrosion of structural material due to free-fluorine, tritium extraction, and safety of activated molten salt.

Tritium breeding for the FLIBE blanket, without an external beryllium component, can reach about 1.11 tritons per D-T neutron, when the blanket has a 3 mm first wall, and inboard and outboard breeding zones of 0.4 m and 0.5 m, respectively. The structural content in the breeding zones is 10% by volume. With a 0.2 m neutron-multiplication zone (60% beryllium by volume) in inboard and outboard, the tritium-breeding ratio can go up as high as 1.33. The total nuclear heating rates, with and without external beryllium, are about 18 and 16 MeV per D-T neutron, respectively. For the FLINABE blanket, tritium breeding will not exceed 0.9 tritons per D-T neutron without an external neutron multiplier. With a 0.2 m of neutron multiplication zone containing external beryllium, the tritium breeding ration can be as high as 1.21. The nuclear heating rate is also approaching 18 MeV per D-T neutron.

Activation of FLIBE and FLINABE with neutrons is dominated by F18 [half-life 1.83 h, due to F19 (n,2n) reactions], and by F18 and Na24 [half-life 15 h, due to Na23 (n,γ) reactions], respectively. The induced radioactivity will impose safety considerations during power plant operation and accidents. The required confinement factor to prevent the release of radioactive hazard is about a factor of 2 higher for the FLINABE blanket than the FLIBE blanket.

Nuclear transmutation changes the elemental compositions of the ferritic steel. After 20 MW-y/m^2 exposure, the relative changes are –2%, 0%, +10%, and –8%, respectively, for the constituting elements, Fe, Cr, Ti, and W. Additional elements produced due to transmutation also include Mn (1.2 wt%), V (0.26 wt%), Re (0.14 wt%), Ta (820 wppm), H (260 wppm), He (190 wppm), and Sr (110 wppm). Neutron activation of impurities in the reduced activation ferritic steel may complicate the waste management issues. Concentration limits for low-level waste disposal and for materials recycling of several dominating impurities such as Ag, Al, Mo, and Nb were evaluated.

Corrosion of structural material due to free fluorine induced by neutron reactions is a concern. Mechanism to eliminate free fluorine and options to control corrosion of structures in the FLIBE and FLINABE blankets were explored. Other issues such as tritium extraction and make-up materials to maintain the chemistry balance of molten salts were assessed.

JO-14

TOWARDS ADVANCED WELDING METHODS FOR THE ITER VACUUM VESSEL SECTORS

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The problem of joining the ITER VV sectors, considering the tolerance requirements of the blanket attachments, and the time required for TIG welding, continues to stimulate EU R&D into power beam welding techniques for fewer passes and lower distortion by a factor of up to 5. The work on reduced pressure e-beam welding showed that penetration varied with position, fit-up, distance and pressure and single-pass weld control seemed impractical so work on a combination of rest-current-control root weld followed by wire-fill passes promises a practical solution. The NdYAG laser welding has yielded useful and stable results and demonstrated the possibilities of improved performance over TIG. Design on the welding robot has been simulated by the successful use of computational virtual modeling techniques. A robust, singularity-free, hydraulic, parallel-configuration robot is under construction to demonstrate welding and control scenarios and the possibility of active suppression of vibrations induced by machining forces.

JO-15

THE DEVELOPMENT OF TECHNOLOGIES RELEVANT FOR THE DIVERTOR AND THE ROLE OF SPIN-OFFS

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In this paper the development at Plansee during almost two decades on divertor relevant technologies will be described. It will be highlighted how existing technologies and products like X-ray targets for the medical industry and high voltage circuit breakers for the energy transmission have influenced the developments for the high heat flux components and the different concepts envisaged within this period.

In the eighties the medical industry required joints between Molybdenum alloys and fine grained isotropic graphites with high reliability during exposure. At Plansee this requirement led to the development of suitable brazing techniques and specially modified joint interfaces in order to obtain joints for heterogeneous material composites with high thermal shock resistance. When in the mid eighties the series production of graphite backed X-ray targets started, similar joining concepts have been applied to first prototypical components for nuclear fusion.

At the same time the energy industry already applied for high voltage circuit breakers components comprising joints between tungsten and CuCrZr. These joints have been and are still realised by casting pure copper onto tungsten followed by electron beam welding of this composite onto a CuCrZr supporting structure. In the mid eighties ideas came up to replace tungsten by C/C composites for such components. Applying the copper casting process onto C/C composite materials led to the development of the Active Metal Casting process.

As the available C/C composites at that time suffered from low mechanical properties these concepts have not been pursued any longer for high voltage circuit breakers. Nevertheless the technology has been available in the early nineties to be applied and further refined for prototypical plasma facing high heat flux components for nuclear fusion.

At Plansee these technologies have significantly influenced the related developments and manufacture of plasma facing components accompanied with the development of suitable non-destructive inspection methods. Based on these technologies Plansee manufactured different prototypical components for e.g. EFDA, CEA and the IPP leading to the first series-like production of high heat flux components between 1998 and 2002 for the CIEL project of Tore Supra. This production has been accompanied with specially adapted non-destructive examination methods like thermography-, radiographic-, ultrasonic- and pressurized hot Helium leak testing. At this time also advanced concepts for the Divertor of ITER have been pursued leading to a series of prototypical small scale components and finally the manufacture of the first European full-scale Vertical Target for the Divertor of ITER.

Especially the light weight C/C monoblock-components being developed for EFDA attracted the interest of companies involved in the manufacture of advanced combustions chambers for future air- and space-craft engines. This led to first co-operations for joint developments of corresponding high heat flux components. Based on the results obtained with the components for nuclear fusion the activities are currently continued on advanced concepts for high voltage circuit breakers, which have been stopped in the eighties.

For Plansee the involvement in nuclear fusion has therefore been possible due to activities in different markets requiring similar approaches. Whereas the references obtained due to the involvement in nuclear fusion enable Plansee today to participate in new programmes requiring the application of advanced joining and material processing technologies.

IN-13

NEW CRITICAL ASSESSMENTS OF CHAMBER AND WALL RESPONSE TO TARGET IMPLOSION IN INERTIAL FUSION REACTORS

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The chamber walls in inertial fusion energy (IFE) reactors are exposed to harsh conditions following each target implosion. Key issues of the cyclic IFE operation include intense photon and ion deposition, wall thermal and hydrodynamic evolution, wall erosion and fatigue lifetime, and chamber clearing and evacuation to ensure desirable conditions prior to target implosion. Several methods for wall protection have been proposed in the past, each having its own advantages and disadvantages. These methods include use of solid bare walls, gas-filled cavities, and liquid walls/jets. Detailed models have been developed for reflected laser light, emitted photons, neutrons, and target debris deposition and interaction with chamber components and have been implemented in the comprehensive HEIGHTS software package. The hydrodynamic response of gas-filled cavities and photon radiation transport of the deposited energy has been calculated by means of new and advanced numerical techniques for accurate shock treatment and propagation. Detail models of photon radiation transport are developed for either the gas-filled cavity or in the evolving vapor cloud layer above the wall surface. These models include non-LTE multi-group for both continuum and line radiation. Fragmentation models of liquid jets as a result of the deposited energy have also been developed, and the impact on chamber clearing dynamics has been evaluated. The focus of this study is to critically assess the reliability and the dynamic response of chamber walls in various proposed protection methods for IFE systems. Of particular concern is the effect on wall erosion lifetime of various erosion mechanisms, such as vaporization, chemical and physical sputtering, melt/liquid splashing and explosive erosion, and fragmentation of liquid walls.

LO-16

ACTIVE CONTOURS APPROACH FOR PLASMA BOUNDARY RECONSTRUCTION

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Snakes and active contour models [1] have been widely accepted as a powerful and elegant method for object analysis and feature tracking in computer vision and pattern recognition applications [2]. They are characterised with an "energy function" that takes into account both geometric and physical constraints and seem particularly suitable to simulate a deformable material.

The quality of a Tokamak control system depends on the availability of fast and reliable measurements of plasma position and shape. These measurements can only be obtained through a real time boundary reconstruction method like XLOC, which has been successfully used in JET for this purpose since 1994. XLOC code reconstructs the magnetic flux function fitting the magnetic measurements to a set of polynomial functions each valid in a different region of the poloidal plane [3].

In this article the theoretical approach of applying active contours to boundary description is studied. Containing energetic information of the plasma geometric configuration, the deformable curve is allowed to slide on a sort of potential function field until it locks on some local minimum. This function would contain all the information about the magnetic flux map as well as the mathematical term needed for regularisation. "Internal forces" are considered so as to impose smoothness constraints to the curve, while "field forces" are used to attract the snake to salient characteristics, and "external forces" allow the user to pull the curve away from undesired features or to push it towards desired solutions [4].

The validation with a non-linear equilibrium code and the comparison with the currently available reconstruction code will be presented.

This methodology could provide a more accurate reconstruction of the plasma boundary for a real time tracking of the shape: better fit to the physics model and the measurements, while presenting a reduced degree of freedom.

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D-1

PLASMA MODELING FOR POSITION AND CURRENT CONTROL IN FTU

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Many tokamak plasma position and shape control systems have been based on relatively simple PID controllers whose gains have been typically estimated on basic plasma models and then finely tuned according to experimental results. This approach is time consuming and in most cases does not allow to obtain optimal performance.

In order to predict the behavior of the controlled plasma, and, possibly to improve the performance of the controller a model of the plasma-circuit system is needed, together with a model of the power supplies. Usually these plasma-circuit models are linear or linearized about an operating point, time-invariant, and finite dimensional. The availability of a reliable model is an essential part in the design of the plasma controller, in order to minimize the number of experiments needed for a fine tuning.

In this paper we describe a model of the FTU tokamak which allow to simulate the FTU plasma current and radial position control loops. This model is made up of the following four elementary parts: a) the plasma-circuit CREATE_L model [1], which on the basis of the currents in the active coils, of the values of the poloidal beta and of the internal inductance, evaluates the flux and field measurements, in the probe locations; b) the identification algorithm which gives an estimate of the variable to be controlled (a horizontal flux imbalance) on the basis of the available flux and field measurements, of the current in the poloidal field active coils and of the preprogrammed values for the internal and external radius; c) the radial and plasma feedback controllers, which on the basis of the errors on the controlled variables, computes the currents needed in the active coils; d) the converters of the two active coils used for the radial and current control; these converters are equipped with internal PID digital controllers which try to guarantee that the currents flowing in these active coils are as close as possible to the values requested by the feedback radial and current controllers.

A detailed analysis of the converter of the coil which is used to control the horizontal flux imbalance has shown that, with the hardware currently used on FTU, this converter introduces a time delay of about 4 ms which does not allow any improvement of the closed-loop performance, especially in terms of response promptness.

The aim of this paper is to demonstrate that the proposed simulation model is reliable, in the sense that it can predict the system behavior when some control parameters are changed. To show this we carry out two series of experiments: 1) in the first two experiments, with nominal currents during the flat-top of 360 kA and 500 kA respectively, we modify the plasma current and external radius references during the flat-top and reproduce the experimental results in simulation; 2) Successively we modify the radial controller gains and predict the behavior of the plasma current and of the horizontal flux imbalances using the simulation model. These predictions are validated against experimental results.

The final version of the paper, besides describing in more detail the simulation model, will include the experimental results checking the accordance between the simulation results and the experimental results.

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D-2

ANALYSIS OF ITER ERROR FIELD AND CORRECTION COILS

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Perturbation of axial symmetry of tokamak magnetic field (error field) could cause disruptions. The paper analyses error field anticipated in ITER and capability of Correction Coils (CC) to reduce this error field to the acceptable level. The analysis comprises error fields from different sources. These are misalignment of the coils producing Toroidal Field (TF), Poloidal Field (PF) and Central Solenoid (CS), the coil joints, feeders and terminals, the test blanket modules (five of them have martensitic steel as a structural material), the magnetic field reduction system of three neutral beam injectors (two heating and current drive and one diagnostic).

Analysis of the error field is provided for helical magnetic field perturbation for the modes (m, n) = (1, 1), (2, 1), (3, 1) on the magnetic surface q = 2. The 3-mode criterion specifies the level acceptable for ITER plasmas.

The error field expected from misalignment of the TF, PF and CS coils can be calculated only statistically. The calculation uses Monte Carlo method based on the results of tolerance analysis. Other sources provide systematic error fields calculated with simplified magnetic models of the corresponding elements. The error field is calculated for 15 MA inductive scenario at the end of plasma current ramp-up, when the plasma current is high and its density is relatively low. The calculations are made using the codes KLONDIKE and PRORCODE developed at the Efremov Institute.

The ITER error field correction system comprises 6 top CC, 6 side CC, and 6 bottom CC. Toroidally opposite coils are connected to produce the error field with toroidal number n = 1 and have one power supply. Calculation of the currents required for the error field reduction is an ill-posed problem, where the solution is chosen as a trade-off between the performance (the value of “3-mode” error field after correction) and the design capability (the maximum currents in CC). A regularized solution of the problem is obtained with the code REGULA developed at the Efremov Institute.

The analysis has shown that ITER CC are capable of correcting error field practically for all possible combinations of the coil misalignments and systematic error fields. The corresponding probability is 97–98%. The capability is limited by the bottom CC having design current 0.18 MA turn, which is slightly below the required maximum. Currents required for the side and top CCs are well within their capability.

The design of ITER CC (maximum currents) was done on the basis of the study reported in this paper and the results were included in the ITER documentation.

D-3

ELECTROMAGNETIC MODELING FOR THE ACTIVE CONTROL OF MHD MODES IN RFX

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The RFX load assembly was characterized by a thick aluminium shell (400 ms time constant) to provide passive stabilization of MHD modes and to assure plasma equilibrium throughout the plasma discharge, whose nominal duration can be up to about 250 ms. A new load assembly has now been designed featuring a thin copper shell (time constant of 50 ms) enclosing the vacuum vessel and a set of 192 saddle coils mounted inside grooves machined on the outer surface of a stainless steel supporting structure. The saddle coils will be used to perform experiments on the active control of MHD modes. In correspondence to each saddle coil a similarly shaped saddle probe will be mounted on the outer surface of the vacuum vessel, as close as possible to the plasma boundary, to provide signals for feedback and magnetic diagnostics purposes. Moreover, an integrated system of magnetic field pick-up probes, to be installed both on the vessel outer and inner surface, could provide further signals to drive the saddle coil power supply. The main foreseen scenarios include stabilization of single MHD modes with growth time of the same order of the shell time constant, named Resistive Wall Modes (RWM); creation of rotating magnetic field components to interact with the localized deformation of the magnetic configuration due to phase locking of internally resonant tearing modes, in particular, to mitigate plasma-wall interaction by dragging the localized deformation along the torus; creation of single static components trying to establish Single Helicity configuration; independent minimization of the magnetic field radial component measured by each saddle probe ("intelligent shell").

The design of the plasma active control system requires the preliminary development of electromagnetic models capable of taking into account the interaction between saddle coils, plasma modes, passive conducting structures and poloidal and toroidal field systems in the various scenarios. Open loop experiments of mode rotation were already successfully performed in the past operation phase; on the contrary, feedback active stabilization of RWM is a new, mandatory issue to extend the pulse duration beyond the shell time constant and thus it was addressed first. On the basis of the linear MHD stability theory, a model has been worked out to account for the growth rate of typical RFP harmonic components (m,n) of the magnetic field in a cylindrical geometry producing a transfer function from currents in the coils to fluxes in the probes. This model must be integrated into a larger model including the dynamic relationships between the voltages applied to the saddle coils and the actual circulating currents. Axisymmetric field shaping winding and toroidal field winding produce field components which are coupled with both saddle coil and probe systems according to the selected operation scenario. Moreover, self and mutual couplings of the saddle coil system are affected by the close passive structures. All the above mentioned terms were calculated by FEM analyses carried out on 3D models of the load assembly.

The integrated electromagnetic model will allow to run simulations of the saddle coil system and the plasma and to provide a first assessment of the effectiveness of an active stabilization system.

D-4

PLASMA INITIATION STAGE SIMULATION IN TOKAMAK T-15M WITH USE OF TRANSMAK CODE

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Modern code TRANSMAK has been developed in Russia (TRANSience in tokaMAKs). This code was used as basical code for the plasma initiation stage simulations in ITER-FEAT /1/. 2D TRANSMAK code is intended for transient processes simulation at the initial stage of plasma discharge in tokamak-type installations with power supply system limitations taken into account.

The TRANSMAK code include SCENPLINT code (SCENario of PLasma INitiation in Tokamaks). The SCENPLINT code allows to calculate of plasma resistance behaviour time dependence and is based on modernization of B.Lloyd e. a. model /2/.

The TRANSMAK code solves simultaneously several tasks:
- creation of conditions necessary for the breakdown in choosen region of plasma vacuum vessel (i.e. necessary values of vortex electrical field E and stray magnetic field B_{str} in the breakdown region);
- simulation of PF currents and plasma current dynamics at the stage of plasma initiation.

Toroidal eddy currents on the tokamak conducting elements and power supply system limitations are taken into account.

The TRANSMAK code was used for the calculation of plasma initiation from outboard in tokamak T-15M (major radius R_o = 1,55 m, minor radius a = 0,5 m, plasma current $I_{P,max}$ = 1,7 MA, magnetic field at plasma axis B_t = 2,5 T, geometrical configuration is similar to ITER in proportion ¼, see report Status of tokamak T-15M Project in this Symposium) and gave the following results:

- poloidal flux losses at the stage of breakdown conditions creation are ≈ 0,1 Wb;
- T-15M power supply system allows to achieve successful plasma initiation (breakdown and plasma current ramp up to the value I_P ≈ 30 – 40 kA when plasma has configuration with closed magnetic field lines and becomes to be possible use of more standard transport codes (ASTRA, DINA));
- in the case of outboard plasma initiation in T-15M with R = 1.9 m, a_o = 0,2 m, E ≈ 0,8 V/m, B_{str} ≈ (0,7–1) mT, initial gas pressure p ≈ 1,7 mPa and level of carbon impurity ≈ 2%, ECRH assist at the level of Q_{ECRH} ≈ 200 kW is necessary;
- the plasma current rise with the speed ~ 1,5 MA/s is possible with taken into account the power supply limitations.

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D-5

REAL-TIME DETERMINATION OF GLOBAL CONFINEMENT PARAMETERS IN JET

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The main confinement parameters, like the internal inductance l_i and the diamagnetic poloidal beta β_{DIA}, are of particular relevance for a reliable real-time control system of next step Tokamaks. These quantities have been obtained at JET (Joint European Torus) using the Shafranov integrals S_1, S_2 and S_3. Indeed they allow the direct calculation of the Shafranov parameter $\Lambda = \beta_{MHD} + l_i / 2$ and, together with the diamagnetic parameter μ, of β_{DIA}. Moreover, in discharges with a sufficiently high elongation (k > 1.3 typically), the internal inductance can be separated from the MHD poloidal beta β_{MHD} and calculated independently, through the Shafranov integrals, with a precision which is more than satisfactory for real-time applications.

Since S_1, S_2 and S_3 are integrals defined on the plasma boundary, an upgrade of the fast code XLOC, that satisfy the real-time constraints of good precision and high computational speed, has been expressively developed to determine the last closed flux surface. The method has been verified on several experimental plasma configurations, giving very encouraging results both in the limiter and x-point phases of the discharges. As the code has to supply data to other real-time applications at least every 10 ms, the compatibility with the time restrictions has also been tested. The computational time, necessary to determine the plasma boundary and more than 30 signals, is only of about 1.5 ms on a PC equipped with a 400 MHz Pentium II, well below the previous constraint.

This application has therefore been implemented and will be extensively in operation during next experimental campaigns, in particular for the control of Internal Transport Barriers and next step Tokamak scenarios, and for the avoidance of MHD instabilities.

D-6

DESIGN AND ANALYSIS OF THE PLASMA POSITION AND SHAPE CONTROL IN SUPERCONDUCTING TOKAMAK JT-60SC

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The modification of JT-60 is planned to a full superconducting tokamak (JT-60SC). The mission of JT-60SC program is to establish scientific and technological bases for advanced operation in economically attractive nuclear fusion reactors. One of the established major research objectives in order to accomplish above missions is to realize high beta steady-state operation using non-inductive current drive with a high bootstrap current fraction. As known well, while a highly elongated plasma shape has an advantage to increase the plasma pressure, the stability of vertical position control becomes a critical issue with plasma shaping. Furthermore, since the JT-60SC is designed under maximum utilization of existing power supplies, the available voltage of magnet power supply would be considerably limited. Therefore, the accurate and reliable investigation of plasma position and shape control capability is required to optimize the modification on the power supply system.

In the present design of JT-60SC, it has four central solenoids and six equilibrium field coils. Furthermore, two sets of copper coils for producing horizontal and vertical fields are installed in vacuum vessel to achieve a rapid response of magnetic field control, because the induced eddy current of vacuum vessel may cause a time delay of the magnetic field penetration if the coils are located outside the vacuum vessel. We modelled the vacuum vessel and stabilizing plates installed for resistive wall mode (RWM) instabilities and selected the dominant 56 modes of the eddy current using 3D eddy current analysis code(EDDYCAL). The numerical model of plasma behaviour was obtained using ACCORD3 code which is based on the linearlized Grad-Shafranov equations with assumptions of small plasma deformation and flux conservation. The modelled plasma parameters are as follows: plasma current of 3 MA, major and minor radii of 2.8/0.77 m, normalized beta of 3.6, elongation (κ_{95}) of 1.86 and triangularity (δ_{95}) of 0.38. A large and sudden poloidal beta drop of $\Delta\beta_P$=0.5 is assumed as a plasma disturbance, because such a phenomenon is often observed as a beta collapse related to the internal transport barrier in high beta experiments in JT-60U. The simulation result shows that the plasma will shift by ~3 cm inward instantaneously. The vertical position oscillates slightly but it can be stabilized within about 0.1 s by the in-vessel horizontal field coil. The required voltage at the coil power supply is evaluated to be ~150 V which corresponds to about one third of the available voltage of thyristor converter to be used. Similarly, the plasma vertical position is successfully controlled in case of applying disturbance of the horizontal field of 0.1 mT. The actual simulation was performed using a conventional code of Matlab/Simulink with Power System Blockset to take into account the delay of magnetic sensing system, feedback control computation time and non-linear property of the power supply system.

In conclusion, the rapid control capability of plasma position was obtained by the in-vessel coil with affordable voltage power supplies. Other simulation results, e.g. the control of plasma radial position, X-point and plasma-wall gaps, will be presented in the conference.

D-7

NEUTRAL POINT DETECTION IN JET

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Any plasma current, position and shape controller must counteract a number of plasma disturbances – e.g. a fast plasma current quench due to a density limit disruption, or a poloidal beta drop due to a giant ELM. Such disturbances induce eddy currents in the passive structures that interact with the plasma; the imbalance of attractive forces between such currents and the plasma can cause the plasma itself to move vertically in a preferential direction (i.e. upward or downward). It can be expected that there exists a position for the plasma magnetic axis (called the neutral point) such that ideally this imbalance is zero and the vertical position displacement after a given time interval is zero (practically very small).

In a perfectly up-down symmetric device with a perfectly up-down symmetric plasma any point lying on the $z = 0$ axis would be neutral according to this definition. In non-symmetric configuration, the existance and location of such points is not obvious. In the past [1–3], the existance of such a neutral point has been experimentally demonstrated on JT60 and confirmed numerically via TSC simulations.

In this paper, we will describe a similar experimental and numerical exercise performed on JET and using the CREATE_L code [4].

Once the neutral point will be identified experimentally and characterized numerically, we will have reached two goals.

a) Operation at the neutral point could be advisable in particular situations, since the vertical position displacement for a given perturbation after a given time would be minimum, giving a higher chance to the controller to counteract successfully the perturbation.
b) A reliable and experimentally validated numerical tool will be developed, able to successfully predict the existance and the location of the neutral point for other (currently operating or future) devices.

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D-8

PLASMA RESPONSE MODELS FOR CURRENT, SHAPE AND POSITION CONTROL IN JET

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In the next future, many physical experimental activities on JET will require ITER-like plasmas with high triangularity and elongation. There is a project aimed at investigating the possibility of obtaining extremely shaped plasmas with the existing active circuits and control hardware [1]. To achieve this objective, it is planned to enhance the present shape control and vertical stabilization systems on the basis of a reliable plasma response model.

Here we present the features and the performance of the JET plasma response models based on an upgraded version of the CREATE-L code [2], which takes into account several aspects:
- an equivalent axisymmetric model of the iron core;
- the response to the coil currents and to the plasma current density profile parameters;
- the set of magnetic diagnostics available;
- the eddy currents induced in the passive structures.

The equivalent axisymmetric model of JET has been assessed from the electromagnetic point of view on a set of dry runs, with both linear and nonlinear magnetostatic and eddy current analyses. The response to the coil currents and the main plasma current density profile parameters, i.e. the plasma response model of interest to the shape control, has been assessed on a set of JET pulses, by comparing the simulated open loop response of the magnetic measurements and the plasma shape to the experimental measurements. The shape and current control oriented model assumed the ten PF circuit currents and the plasma current I_p as state variables, taking as input quantities the applied voltages (or, alternatively, the circuit currents) and the time behaviors of and the two quantities $\beta_p I_p$ and $l_i I_p$ (related to the poloidal beta β_p and the internal inductance l_i). For the design and the assessment of the vertical stabilization system, a model was set up including the eddy currents in the passive structures (vessel and Mark II) as state variables, simulating the open loop growth rate of the vertical instability.

The main results of this analysis are that:
- a good axisymmetric model of the iron is obtained assuming the correct geometry of the polar shoes, limbs and legs;
- the non-linear effects due to the B-H curve are small, above a certain m.m.f. threshold;
- there is a significant effect of the eddy currents;
- there are some offsets in the measured currents and voltages;
- it is possible to reliably predict the growth rate of the vertical instability of an elongated JET plasma;
- the model provides a reliable base for the design and the assessment of a new current, shape and position control system in JET.

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D-9

THE CONTROL SYSTEM OF THE ITER FEAT VERTICAL STABILIZATION CONVERTER

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In the ITER machine, the stabilization of the plasma vertical position is assured by the passive reaction of the vacuum vessel to the plasma movements and by an active position control system. It acts by means of a special power supply, named VS converter, feeding the four central poloidal field (PF) coils. The converter-coils connections are such that the coil current components, fed by the VS converter, produce an horizontal component of the poloidal field, so as to restore the desired vertical plasma position.

The VS converter is rated for 8 kV no load output voltage and 22.5 kA maximum output current.

It is based on two identical basic units connected in series; the basic unit is made of two identical sub-units connected in parallel. Each subunit is composed of two back to back thyristor bridges in series, respectively connected to a star and delta transformer secondary winding so as to obtain 12-pulse operation. The basic unit operates on four quadrants with the two subunits supplying the same current, till the load current is higher than a certain threshold; then a circulation current is set to avoid discontinuity in the zero crossing region.

As for the control system, the main requirement is a fast transient response, which is not a typical characteristic of line-commutated converters of such high power. In fact the VS converter is required to assure transition of the output voltage from full negative voltage to full positive voltage and vice versa with a maximum delay of 2.5 ms and a time constant less than 7.5 ms.

The paper describes the structure of the VS control system. It is composed of an open loop for voltage regulation, providing however feed-forward blocks to compensate for the converter voltage drops, and a circulation current control loop. One of the most peculiar aspects of the design is the search for the best compromise between the fast response requirement and the limitation of the circulation current peak in any operating condition. Another important issue is the optimization of the dc interface inductance, again related to the limitation of the circulation current peak.

In the paper, we also report the analyses performed, both with simplified and complete circuit, to evaluate the converter performances and to verify their accordance with the specification.

D-10

N=2 COMPENSATION AND VARIABLE GAINS FOR JET VERTICAL STABILISATION

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In all JET plasma discharges the plasma vertical position is unstable. In order to obtain plasma lasting more than the vessel field penetration time constant, a system called vertical stabilisation had to be employed [1]. In order to calculate the plasma vertical speed (using the vertical moment time derivative) a passive resistor network combines the non integrated signal produced by 2 sets of 32 coil, placed poloidally around the machine and in 2 opposite toroidal locations (to reject noise from the n=1 modes).

Since 1994 JET has been running with four new internal shaping coils and a divertor. At that time, to be able to cope with the increased complexity of the machine, the vertical stabilisation had to be upgraded to a digital controller. The vertical velocity was still provided by the same combination of signals, while the main control algorithm was now performed in software [2][3].

Unfortunately the presence of the new shaping coil, and moreover the introduction in the vessel since the Mark2 divertor of a toroidally conductive support structure, has resulted in a reduction of quality of the signal produced by all the measurement coils in the lower part of the machine. While most of the desirable plasma configurations are well stabilised, during the years a small group of extremely elongated plasma has pushed the vertical stabilisation control system to its limits. Furthermore, the new power amplifier used for control, the FRFA (a GTO based very fast switching amplifier producing 9 voltage levels)[4], while proving excellent at controlling very elongated plasmas, proved very sensitive to oscillation caused by n=2 plasma modes disturbances. The excessive switching caused the amplifier to overheat and trip. Finally, giant ELMS have proven to be more than a challenge for the present system.

This paper will discuss a new system that will try to solve the problems introduced in the vertical stabilisation since divertor operations. 4 sets of 32 coils will be used instead of 2 in order to remove the disturbances created by n=2 plasma modes. The signals will be separately isolated and combined using a newly developed digital-analogic hybrid card. This card will provide a higher bandwidth, which should extend the range of controllable plasmas. At the same time the weights of the combinations will be dynamically adjustable, thus providing a mean to try measurement schemes potentially immune to the disturbances created by ELMS.

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D-11

JET REAL-TIME OBJECT-ORIENTED CODE FOR PLASMA BOUNDARY RECONSTRUCTION

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The plasma boundary reconstruction code XLOC [1] has been used routinely in JET in the plasma shape control since 1994 [2]. An array of digital signal processors produced plasma wall distances measurements in less than 1 ms was adequate for the needs of the newly developed shape controller [3][4]. The code was written in "C" language starting from an original FORTRAN implementation that had been used for offline analysis [5].More recently, the difficulty in obtaining a steady state reverse shear mode of operation has generated the need for more real-time internal plasma parameters, such as confinement and current profile. For this reason the real time project has been initiated at JET.

A new implementation of XLOC has then been designed with the aim to facilitate the development of the several new real time subsystems. The original "C" DSP code has been first transformed to portable "C++" in order to produce a reusable component and then has been improved in its flexibility and in its efficiency.

This paper will describe the XLOC algorithm as implemented in both the C and the new C++ version, and how the different applications have made use of it.

The real-time calculation of many important plasma parameters, such as confinement parameters and profile reconstruction, has been the first user of this new library. These applications require the development of a more efficient boundary reconstruction that avoids convergence problems caused by the X-Point. They also need a reconstruction of the field along the plasma boundary in order to calculate volume integrals using Shafranov integrals. A newly developed real time equilibrium code EQUINOX will use as inputs the magnetic field reconstructed by XLOC in a fixed closed line following the limiter. The plasma current internal profile will be measured in real time using information from several polarimetry and density interferometers that scan the plasma along different line of sight. The real-time reconstruction of the plasma boundary will be provided by XLOC. The library has also been used to produce a tool to simulate the magnetic measurements. This software package has been used to help the specification of the JET magnetic refurbishment.

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D-12

NON LINEAR MODEL OF THE GAS INTRODUCTION MODULE FOR PLASMA DENSITY CONTROL AT JET

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The performance of a Plasma Density Feedback System relies on an accurate estimation of the number of particles provided by the Gas Introduction System. In JET, the Gas Introduction System is composed of ten Gas Introduction Modules (GIMs) placed in different poloidal locations around the vacuum vessel (divertor region, outer midplane, top of the machine). An accurate model of the GIMs able to relate the actual particle flow to the real-time voltage control signal is essential for the robustness and the performance of the control system.

Because of the unavailability of an accurate flow sensor able to cope with radiation and electromagnetic disturbances, a flow model, based on the GIM reservoir pressure measurements, had to be developed. The static non-linear relations used since now, have shown not to be appropriate for a precise and reliable flow estimation because of the large operating range which is not taken into account by such a model. Furthermore, the thermal exchange during the gas flow cannot be neglected because it greatly affects the dynamic behaviour.

In this paper, a dynamic non-linear model of the GIM is proposed. It has been designed as the cascade of two subsystems so as to better describe the physics involved in the process.

In the first subsystem the pressure evolution in the GIM reservoir is related to the voltage signal driving the piezoelectric valve that handles the amount of gas delivered from the reservoir. In the second subsystem, the non-linear differential equation between the pressure measurement and the particle flow is calculated. In both subsystems the thermal exchange between gas and the walls is estimated and included.

The two models have been validated separately with experimental data, providing accurate estimations of the particle flow. This GIM model can be effectively used both as a modelling tool for the design of an improved Plasma Density Control System, and for a real-time gas flow diagnostics.

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D-13

NEXT-GENERATION PLASMA CONTROL IN THE DIII–D TOKAMAK*

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The advanced tokamak (AT) operating mode which is the principal focus of the DIII–D tokamak will require highly integrated and complex plasma control. Simultaneous and high performance regulation of the plasma boundary and internal profiles will require multivariable control techniques in order to take into account the highly coupled influences of equilibrium shape, profile, and stability control. This paper describes progress towards the DIII–D AT mission goal through both significantly improved real-time computational hardware and control algorithm capability.

Routine operational plasma control requires providing algorithms which operate with all normal experimental plasma configurations as well as dealing with a number of operational constraints. In the course of normal experimental operations at DIII–D, a large range of plasma shapes, currents, and fields are used, requiring design and implementation of controllers covering this large range, as well as methods for switching between these controllers as the plasma regime changes. Operational constraints include sometimes severe limits on coil currents and voltages, an extremely nonlinear set of shaping power supplies, actuators which are shared by both vertical position stabilization and shape control, a linearized X–point response which changes sign during an experimental discharge, and a hard constraint on achievable combinations of shaping coil currents.

When multivariable controllers replace the current PID control, a number of different control methods will be used to deal with the various constraints and nonlinearities. Gain-scheduling of multivariable linear controllers will be used to accommodate the range of plasma regimes. Anti-windup and bumpless transfer methods will be used to ensure smoothly varying performance during and between transfers of controllers. Several novel nonlinear control methods will be used in conjunction with the multivariable linear controllers in order to handle the constraints and nonlinearities imposed by operational hardware and real-time changes in plasma response.

A major enabling aspect of DIII–D PCS development is the next-generation plasma control system (PCS) hardware and network configuration upgrade which has been underway for the last two years. This upgrade extends the architecture of the PCS to much higher performance and flexibility, producing a factor of 10 or more increase in processor speed and adding a scalable network which can accommodate a large number of real time cpus and peripherals. Phase I of the upgrade plan is complete with all implementation scheduled for completion in 2004. Three of the previous six VME based i860 real-time processors have been replaced with new PCI based computers and the current hybrid VME/PCI system is currently in use in experimental operations. A new 2 Gigabits per second Myrinet network is in use and pre-integration testing of new 32 channel simultaneous sampling PCI form digitizers has begun.

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D-14

PENETRATION OF OSCILLATING MAGNETIC FIELD THROUGH ITER DOUBLE WALLED VACUUM VESSEL

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This work relates to active stabilization of the Resistive Wall Mode (RWM) instability in the ITER. The RWM resembles another mode - instability of elongated plasma to the vertical displacement as a whole. Both these modes are stabilized by combination of the magnetic fields, created by eddy currents and by feedback currents in some coils. The difference is that vertical displacement mode is stabilized by the radial magnetic field, but the RWM – by helical magnetic field. For both modes, periodic (oscillating) and non-periodic regimes are mixed at real operation, but can analyzed separately in parametric studies.

Specific work scope was to simulate periodic and non-periodic magnetic field penetration processes into the ITER Vacuum Vessel (VV), with special attention paid to the frequency-dependent electromagnetic shielding properties of the double-walled VV. External magnetic field was applied in n=1 mode, and created by a pair of correction coils with co-directional magnetic fields. These coils were located at two ends of one diameter. A virtual magnetic probe was located at the same diameter, near the VV inner wall, and measured the field component normal to the wall. The study was performed with the TYPHOON code developed at the Efremov Institute. Double walled VV was simulated by two enclosed toroidal shells, covering ¼ of the full torus.

The major conclusions from numerical study are in frequency-dependent parameters of the double-walled VV. With frequency grow from 0.05Hz to 32 Hz, still far enough from the VV wall skin time, the following major observations were made:

- the phase shift between the coil voltage and probe’s field increases noticeable faster then for a single wall model, and reaches asymptotically 270 degrees, instead of 180 degrees possible for a single wall model.
- characteristic time (defined as L/R, where L and R are effective self-inductance and resistance of the total eddy current loop), was found in a scale of 300 ms for the lowest frequencies, and drops almost in twice at frequency in a few Hz.

These observations were proven with additional numerical studies and explained qualitatively. For example, at elevated frequencies, the eddy currents in two walls have very different densities and phase shifts. The inner shell sees the field penetrating through the outer shell, and this makes all processes in the inner shell delayed at almost 90 degrees relative to ones in the outer shell. This explains the total asymptotic phase shift enlarged to 270 degrees.

These results were included in the Design Description Document of ITER activity and used for designing the ITER machine. They will help in development of advanced plasma control algorithms, which reflect better the electromagnetic behavior of the double-walled VV.

D-15

THE NEW ASDEX UPGRADE REAL-TIME CONTROL AND DATA ACQUISITION SYSTEM

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ASDEX Upgrade investigates the integration of confinement, stability and exhaust issues into an operating scenario for ITER and a future fusion reactor. Since commissioned in 1990 the systems used to feedback control plasma position and shape as well as performance have continuously been enhanced. To overcome performance limitations and improve connectivity and steady state capability, a new plasma control system is being implemented.

For the new distributed system, adequate and reliable communication mechanisms are essential to integrate the real-time discharge control and data acquisition. During discharges, process information is distributed periodically among control and diagnostic applications via a common real-time network.

Application tasks are connected, via the network, to supporting tasks such as control cycle management, reference value injection, protocol value extraction and machine protection.

In the preparation and postprocessing phases of a discharge, complex data structures and other information, must be exchanged between heterogeneous systems. Applications must be parametrised prior to a discharge. Protocol and logging data must be archived. This communication, however, is not subject to hard real-time constraints. The middleware NDDS, has been selected for this job and runs under VxWorks, Solaris and Linux and on a variety of hardware platforms.

Tests have shown that the middleware´s performance is also sufficient to pass process information during discharges. This permits the extraction of control information for close-to real-time visualisation and even allows the integration of diagnostic systems, with moderate data rates, not connected to the real-time network.

We present communication methods and the process organization of the new distributed system and show that the new concept allows easy performance scaling with the availability of powerful standard hardware components. We demonstrate how existing periphery and new real-time diagnostics interface to control applications and how this facilitates the realisation of novel and sophisticated control tasks combining multiple diagnostics and actuators for common physical goals.

D-16

THE MAST DIGITAL PLASMA CONTROL SYSTEM

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A digital control system is in development at Culham as an upgrade to the present MAST plasma control system [1]. The existing hardware is primarily analogue-based, but allows weights to be set digitally, and controller gains can be programmed to vary in time during the shot. The primary motivation for the move to an all-digital system is the vast increase in flexibility and capability that will be gained, especially for non-linear and adaptive control.

The hardware for the new system primarily comprises of:

1. digitisers that capture the signals from plasma and plant diagnostics and output converted data immediately via Front-Panel Data Port (FPDP [2]),
2. the multi-cpu real-time computer, which receives data from the digitisers via FPDP, and
3. VME-based digital-to-analogue converters to send the computed response to the plant.

The system is VME-based, because VME is a stable, long-life platform. FPDP was chosen for the transfer of acquired data because it has low latency and high bandwidth. Since the number of outputs is much smaller than the number of inputs, the VME bus was considered adequate to handle the output traffic. The real-time computer is hosted by a single-slot embedded SPARC workstation, which provides host system services such as peripheral and network I/O, thus freeing up each real-time processor to run only the actual real-time control tasks.

Under a collaboration agreement, General Atomics have provided their “PCS” infrastructure software [3], which has been successfully ported to support the MAST device, the above control hardware, and the local data archiving format. To gain acceptance and facilitate testing, the first algorithms will mimic the behaviour of the existing system. The software already has the flexibility to allow much of this testing to be done in simulation mode.

Further collaboration is in progress with General Atomics to develop a ‘code generator’ tool. This will allow new control algorithms and the data structures that they use to be defined in XML (eXtensible Markup Language) source files, which encapsulate the programmer’s requirements in a neat and logical way. The code generator parses these files to generate the C source code with a self-consistent set of data references. Documentation is also automatically generated from the same source. This tool will be of considerable use for rapid development of the next set of control algorithms.

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D-17

DINA SIMULATIONS OF TCV DISCHARGES

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Increased reliance on detailed models of the operational behaviour of a tokamak discharge requires positive validation of the reliability of any particular code. The DINA 1.5D plasma discharge simulation code is becoming widespread in its use and substantial effort has already been dedicated to benchmarking it. As part of this overall exercise, DINA was set up and validated for TCV discharges in two specific Ohmically heated plasma conditions.

Firstly, the experimental responses of limited and diverted plasmas to PF coil voltage pulses were compared with DINA simulations. This required correct simulation of the diagnostics, the feedback control system and the power supplies. In a second exercise, "free-fall" Vertical Displacement Events were simulated during the very large vertical movements possible in the highly elongated TCV vacuum vessel. These two validation exercises increased our confidence in DINA for simulating a full discharge evolution including the free-boundary evolution of the plasma shape itself. This work has now been extended to additionally heated discharges, using the TCV electron cyclotron heating and current drive. Specific features of these discharges include fully non-inductive current drive, intense additional heating with no current drive, and a mixture of the two. Discharges with off-axis current drive and with substantial bootstrap current are also being simulated.

For Ohmically heated discharges, the evolution of the plasma temperature in the simulation and the experiment are naturally stable, finding a stationary operating point. The positive temperature dependence of the current drive efficiency adds an additional problem to the simulation of TCV discharges using DINA, which is discussed in this paper. DINA is being implemented in the user-friendly Matlab environment for rapid execution of discharge simulations. The suitability of this platform will also be presented.

D-18

SENSORLESS SENSING OF PLASMA HORIZONTAL POSITION ON HT-7

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In a nuclear fusion reactor, diagnostic systems should play a role of measurement necessary for reactor protection and plasma control. Since the sensors are used under severe irradiation circumstances, the number of diagnostic sensors is better to be small or zero. Therefore it is important to study on sensorless control where we need no sensors of the controlled object except sensors of the actuator.

In a feedback control system of plasma horizontal position x_P, the x_P is controlled by vertical magnetic field made by vertical field coil current I_V driven by vertical field coil voltage V_V. When the plasma shifts horizontally, a voltage is induced in the vertical field coil. In a power supply of voltage control type, I_V is increased, and in a power supply of constant current control type, V_V is reduced. Therefore we can obtain some information on x_P from the V_V and I_V, and we can deduce the x_P from them. We consider electrical equivalent circuit equations of plasma and vertical field coil. Since the mutual inductance between them depends on x_P,

$$\left(I_P \frac{dM_{VP}}{dx_P}(sx_P) + M_{VP}(sI_P) \right) + \left(\Omega_V + sL_V \right) I_V = V_V \, , \tag{1}$$

where s is an operator for Laplace transformation. Using this equation, we can calculate x_P from the V_V and I_V.

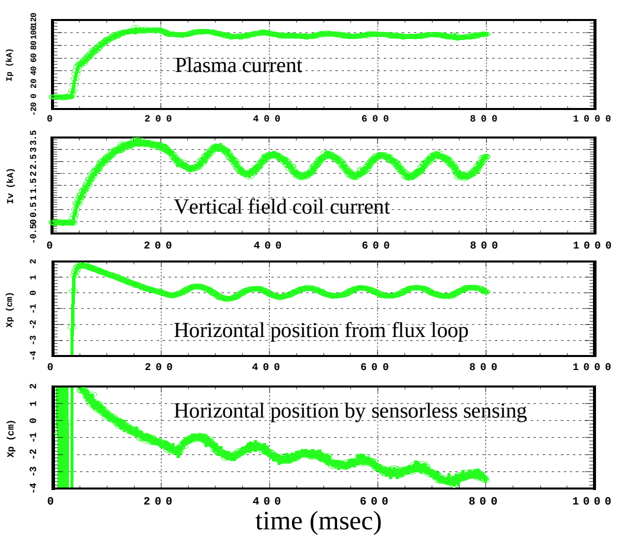

Fig. 1. Waveforms of the shot number 45165.

In the superconducting tokamak HT-7 (B_t =.2.5 T, R = 1.22 m, a = 0.26–0.30 m), effect of eddy current in thermal radiation shield must be taken into account. We swept the plasma horizontal position by 7 Hz in the shot number 45160. We calibrated the equivalent circuit parameters by a least square method. Using these parameters, we calculated the x_P of the other shot number 45165 where the x_P was swept by different frequency 10 Hz. The result is shown in Fig. 1. Although it is necessary to improve the low frequency characteristics of sensorless sensing method, the calculated sinusoidal waveform coincides the waveform obtained from flux loop signal

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D-19

SIMPLIFIED MAGNETOSTATIC MODELS FOR THE TESTING OF IDENTIFICATION ALGORITHMS

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The information used by the plasma shape and position control system to drive the shaping coils currents come mostly from magnetic sensors, either flux or field probes. The rationale behind this choice is that the main plasma shape quantities are tightly related to the magnetic flux map inside the chamber. It is then evident that the performance of the identification subsystem (comprising either the measurement system and the pieces of software dedicated to their analysis) is critical for the achievement of satisfactory performance.

Although the plasma current is governed by the complex equations constituting the MHD model, quite a lot of information about the flux map structure can be obtained by adopting the so called "Equivalent Current" models.

In the design of identification systems, the interest is focused on the flux map structure rather than on the physics behind the MHD models; therefore simplified models, able to reproduce the same structure of the flux maps present in the full MHD models, can be effectively adopted. The advantage is the significant reduction of computational times, allowing the generation of a very large number of flux maps in a short time, with the required statistical distribution.

Such a large number of maps can be either used to assess the performance of an identification system on statistical basis, or as a data base of possible "equilibria" for those algorithms requiring large number of cases for their setup (i.e. neural networks or simplified model extractor such as PCA methods).

In addition, in the case that the control system is designed for "slow" phenomena (e.g. control of evolution at flat top), the eddy currents induced in the conducting structures can be neglected, obtaining for the simplified models a magnetostatic set of equations.

In this paper an effective technique for the simplified magnetostatic modelling of a tokamak device is proposed, taking into account the possible presence of iron parts. The algorithm is applied to the flat top of ITER and JET tokamaks and its performance is evaluated by considering the main magnetic moments of the current density distribution.

D-20

REAL-TIME SAFETY FACTOR PROFILE DETERMINATION IN JET

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During the last experimental campaign at JET, we have demonstrated the feasibility to control in real-time Internal Transport Barriers (ITBs) characteristics using a simple criterion to determine the ITB's strength, existence and location. One of the main conclusions of these experiments was that the control of the pressure alone was not sufficient to sustain high performance regimes in steady state and active control of the current profile was also required.

In this paper we present the recent implementation in the JET real-time system of new algorithms to calculate the electron density and the safety factor profiles directly from the measurements of both the line integrated electron densities (with real-time fringe jumps corrections) and the calibrated Faraday angles. Current density, electron temperature and density profiles are now available in real-time using magnetic, interfero-polarimeter, and electron cyclotron emission data thus allowing real-time feedback control of these profiles with the heating and current drive actuators available on JET. The geometry of the last magnetic surface is determined by a fast boundary reconstruction code using external magnetic measurements. The density and current profiles are obtained by inverting the interferometric and polarimetric measurements.

Details will be given on the hardware and software architecture. The system is formed by three subsystems: two VME systems and one real time PC, communicating via the JET real-time ATM network. The first VME system is devoted to the Line Integrated Density (LID) calculation and the fringe jumps correction the second to the calculation of the calibrated Faraday angles (FAR). The LID subsystem uses a fast 16 bit ADC running at 400 KHz sampling frequency, passing data to the CPU (PowerPC 400 MHz) via a fast Front Panel Data Port (FPDP) interface. The FAR subsystem uses a standard JET ADC module operating at 1 kHz sampling frequency and produces as output an ATM packet containing the LID and FAR data. The real time PC acquires via ATM the packets coming from the LID and the FAR subsystem together with information from the fast boundary code. Then it calculates in real-time the density and the poloidal field profile. Real time calculation of density and q profiles are showing good agreement with the Thomson scattering LIDAR profile and equilibrium code EFIT respectively.

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JET FIRST WALL AND DIVERTOR PROTECTION SYSTEM

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and contributors to the EFDA-JET work programme

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In modern tokamaks the plasma shape is one of the most important parameters determining the fusion performance. For this reason a digital PPCC (Plasma Position and Current Control) Shape Controller has been introduced at JET [1][2]. Thanks to the introduction of both the Gap control and a much improved user interface, the Session Leaders are now able to easily design and run extremely complex plasma scenarios. It is now possible to slightly modify a discharge setting without having to resort to the extremely lengthy process of a full simulation. This, together with the difficulty of taking in account all the possible events, increased the probability of erroneously placing hot plasma on a non power-handling part of the first wall.

A Wall Loading Limiting System (WALLS) has then been developed as a complement to PPCC. This system protects the plasma facing structures from excessive power deposition. WALLS checks in real-time whether the plasma is too close to a predetermined set of first wall areas. When the gap clearance is smaller than a pre-set threshold, it is then assumed that all the additional heating power is transferred to the wall at that gap. Each gap is rated with a maximum energy handling capability, when the total energy discharged to it reaches a value greater than its limit the additional heating is removed and the pulse is promptly terminated. Recently, the divertor dome has been replaced with a plate (Septum Replacement Plate, SRP) with limited power handling capabilities. WALLS has been recently modified in order to take into account the new needs arisen from the different divertor structure (Mark II SRP). The maximum allowed power load on the SRP depends on the value of the perpendicular field line angle. The power is delivered to the divertor tiles in the position determined by the strike points. The power density and the area of the wetted surface depend on the size of the flux expansion in those locations. WALLS has been upgraded to protect in real-time also the SRP and, for this, to include ohmic power as well as additional heating and to partition the total power among radiation, inner and outer leg of the plasma separatrix.

All the real-time data are provided by the real time network (RTDN) via an ATM link.[3]

This paper will principally focus on the recent upgrade for the new SRP divertor.

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D-22

DESIGN, FABRICATION, INSTALLATION AND TESTING OF IN-VESSEL CONTROL COILS FOR DIII-D*

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Since 1995, DIII-D has performed correction of magnetic field imperfections using a set of six external picture frame coils located on the vessel mid-plane. Recently, these coils have also demonstrated significant benefits when used for feedback of the resistive wall mode, an instability that limits the plasma performance at high beta. Modeling has shown that substantial performance improvements can be achieved by installing new coils inside the vessel and expanding the poloidal coverage both above and below the mid-plane. Two prototype internal coils were installed in 2001 and have tested successfully. Installation of a full set of twelve internal coils and related magnetic sensors in the DIII-D tokamak is scheduled for completion in December 2002.

The design requirements for the new coil system were to maximize the magnetic field at the plasma edge, operate with a frequency range of DC to 1000 Hz, and fit behind the existing graphite wall tiles to avoid any plasma contact. A technical comparison of expected operation of in-vessel and ex-vessel designs indicated a significant performance advantage of the in-vessel coils due to the closer proximity to the plasma, better field penetration, and lower inductance. However, the in-vessel coil environment requires a more robust design because of high cyclic electromagnetic forces in vacuum and temperatures to 350°C during vessel baking.

The in-vessel coil design adopted and installed is water-cooled hollow copper conductor insulated with polyamide and housed inside a stainless steel tube, which forms a vacuum boundary. The coil conductor system is 19 mm outside diameter and is engineered to operate up to 7 kA DC with cooling water flow of 5.5 m/s. The coil set is installed as 2 sets of 6 coils each located on the inside of the outer wall above and below the mid-plane. The coils are single turn with a conductor length of 5 m and an area of 1 m^2. In order to minimize error fields and electromagnetic forces, the radially oriented power and cooling water feeds are coaxial. The coils are shielded from the plasma by the plasma facing graphite tiles.

The primary challenge in the design of these coils was in joining of both the copper conductor and the stainless tube without overheating the polyamide insulator. The challenge was met after significant design evolution by using induction brazing (815°C) for the copper and automated orbital welding of the stainless steel tube. The brazing and welding processes are performed both outside and inside of the vacuum vessel.

In operation, protective instrumentation monitors water flow and temperature to prevent water boiling during off normal conditions. To protect against water leakage into the vessel from a damaged conductor, both the copper conductor and the stainless steel tube form redundant barriers. The polyamide insulator space is filled with nitrogen and its pressure is monitored to detect any water or vacuum leak of a single barrier. During vessel baking, the cooling water is automatically replaced with dry nitrogen to minimize copper oxidization at 350°C exposure temperature.

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A NEW SHAPE CONTROLLER FOR EXTREMELY SHAPED PLASMAS IN JET

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In the next future, many physical experimental activities on JET will require plasmas with high triangularity and elongation. This work has been carried out in the framework of a project aimed at investigating the possibility of obtaining extremely shaped plasmas with the existing active circuits and control hardware [1]. The aim is to reach ITER-like shaping with high value of triangularity. One of the steps needed to achieve this objective is the redesign of the JET shape controller, since the actual controller does not take into account the possibility of operating with highly shaped plasmas. The new shape controller has been designed on the basis of a plasma-circuit response model which has been extensively validated against JET experimental data [2].

This paper presents the features of this new controller and some preliminary results obtained in simulation. To control the plasma shape in JET, there are 8 knobs, namely the currents flowing in 8 PF coils, while the P1 circuit is devoted to control the plasma current. In the present JET shape controller, a number of coil currents are feedback controlled at given preprogrammed time trajectories, while the remaining coil current are used to control a limited number of geometrical parameters, usually no more than 5. Extensive simulations have shown that, in the presence of highly shaped plasmas, this approach, even guaranteeing small errors on the controlled geometrical variables, is no longer able to ensure a satisfactory invariance of the overall shape. In this paper we propose a new approach to the shape control in which all the available coil currents are used to control, at the best, the overall plasma shape remaining within the current limits imposed by the power supply. In particular, the plasma shape is defined in terms of 48 geometrical descriptors: 40 gaps chosen all around the vacuum vessel and 8 additional geometrical descriptors for the X-point, the strike points and the plasma current centroid. The proposed control approach gives the following flexibility:

- it is possible to weight separately the 48 geometrical descriptors;
- besides the geometrical descriptors, one can choose to control linear combinations of them, so as to account for variables like the linearized pseudo-elongation, triangularity, ...;
- it is possible to weight the current amplitudes, so as to account for the distance of each actual current from its upper and lower limits;
- it is also possible to directly control some currents; this can be useful during critical phases of the shot, e.g. plasma formation, current ramp-up and ramp-down, …

The performance of the new shape controller have been tested in simulation in two different situations: i) in the rejection of the disturbances (namely β_p, l_i and I_p variations) acting on the plasma during flat top phases; ii) in the transition from a non-extreme diverted equilibrium to an extreme equilibrium, with high triangularity.

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D-24

UPGRADE OF THE PRESENT JET SHAPE AND VERTICAL STABILITY CONTROLLER

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The present JET Shape Controller has the capability to control a limited number plasma geometrical parameters chosen among plasma gaps, coil currents, the X-Point location and the plasma centroid. The set of controllable parameters can be varied during a plasma discharge to accommodate the experimental needs.

Eight, independently supplied, poloidal shaping circuits can be used to obtain the desired configuration. However their action is not totally decoupled: normally only 5 shaping parameters can be easily controlled.

Since now the full JET electromagnetic system has never been modeled accurately because of the non-linearity introduced by the iron core. For the purpose of designing an improved shape controller, a new iron model has been developed. Its results have been tested using data collected from dedicated dry runs, where all the 8 available poloidal circuits were excited separately. The results show a much more accurate prediction of the magnetic measurements compared to the old model.

The CREATE-L approach has been used to linearise an actual experimental plasma equilibrium. The plasma shape of several different experiments from standard H mode discharges to ITBs and high β_p experiments were then reconstructed using this linear model. Despite the large variations of $\beta_p + li$ (up to a factor 10) and of the plasma current the reconstructed plasma boundary was in quite good agreement with that obtained using the equilibrium code EFIT. This model has been used as the base for the design of the new shaping controller.

The newly proposed shape controller, while providing the same capabilities of the old one, will try to control the full boundary against variation in the plasma current profile.

The new control strategy has already been benchmarked connected in feedback to a CREATE-L plasma model. These tests have simulated old JET pulses as well as novel configurations. The results will be presented in this paper. Significant improvements in the control of the shape are apparent.

So far no equilibrium code was able to correctly predict the JET plasma vertical growth rate. The new model of the the JET electromagnetic system has been used to simulate a disruption caused by a vertical instability; the obtained results were in good agreement with the experimental data both when the model was run in open loop and in feedback with a simplified controller. It is now therefore possible to study how to improve the present vertical control system: the new model can be used to predict what is the best choice of turns in the radial filed circuit together with the right combination of voltages and currents in the amplifier.

D-25

PLASMA CURRENT AND POSITION FEEDBACK CONTROL IN ADITYA TOKAMAK

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Ohmic heating power supply is used in ADITYA Tokamak for plasma breakdown and current drive. Presently the ohmic coils store a magnetic flux of 0.40 volt-secs and high di/dt is created for the required loop voltage. After the plasma breakdown the plasma current control is made active with the difference of reference plasma current and measured plasma current to drive the PI regulator, which regulates the ohmic falling current. The vertical field current is also made proportional to plasma current by feedback control at appropriate time. The control scheme adopted and the results with the feedback control are described in this paper. We have observed that if the measured plasma current is less than the reference plasma current the rectifier goes to more inverting essentially increasing the loop voltage and the regulator behaves opposite way if the measured is more than the reference current. We could improve the plasma current as well as plasma current flattop by employing the plasma current feed back control. Further to correct the radial motion / position in fast time scale, 4 quadrant transistor switched bridge inverter power supply by plasma position feedback control is proposed. On line +/- Δ X (by position coils) position signal is given to PI control and polarity detector accordingly the current is amplified by the bridge inverter and fed to the 4 set of feedback coils which intern shifts the plasma position on radial directions.

D-26

HL-2A DISCHARGE PROGRAM EDITOR

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A distributed, graphical and interactive discharge program Editor has been developed within Microsoft Excel. The Editor has taken good advantage of powerful communication ability of MS Excel with Siemens WinCC and MatLab, which are our monitor and simulator, and exploited some new-developed ActiveX controls, so its performance has improved amazingly and the reusability of the involved codes has enhanced greatly.

A Tokamak plasma discharge is a series of complex and concerted action conducted by its control system. In order to understand the performance of control system, to study the plasma discharge, and to fulfill the physicists' experiment purpose, an Editor is necessary, which should provide functions to enable physicists to preset the discharge parameters and to display the collected results as well. In order to store the preset parameters and download them to the corresponding subsystems to control the plasma discharge, well-structured data files are necessary which should meet the constantly growing demands on safety and control tasks. Having encapsulated the working logics and rules, the involved data and their methods, some new-developed ActiveX controls have enhanced the performance of control system and eased the pain of code debugging when logics, rules or the involved data structures change. The switch and condition definition parameters are preset in the WorkSheets of the Editor by clicking the radio button, checkbox, combobox or modifying the cells' values. The switch and condition definition information is stored in a data file with .DPF extension; the data file is well structured so that it can be easily extended to meet the continuously growing demands on safety and control task. Online input checks are realized by using the MS Excel built-in WorkSheet_Change event with a Range parameter. The Editor uses MS Excel built-in Chart to edit the waveform graphically. This is the kernel part of the Editor. A new CommandBar and their CommandButtons are added to the MS Excel workbook as the interface for physicist to edit the waveform, to save the temporal data, and display the waveform. MS Excel built-in msoFreeform shape is used as a tool to get the ideal waveform. The curve the msoFreeform shape provides can be edited easily to any ideal shape by adding nodes to the curve and moving the nodes to the right position. The smoothness of the curve is good enough to requirement, so curve fitting is not necessary when you convert the curve to its corresponding data. Security is the most important aspect of this Editor; ordinary precautions have been implemented into the editor.

The Editor will be used to preset the discharge parameters and operational modes before a discharge, and display the collected data after a discharge. The Editor has been debugged within and outside HL-2A group and will be run in future commissioning and operation.

D-27

MODEL DEVELOPMENT AND CONTROL SYSTEM SIMULATION IN HL-2A

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The mathematical models for control of the plasma radial position and its current in HL-2A are described, and the simulation of the feedback control system by the Simulink of Matlab is presented in this paper too. The feedback control system is consisted of magnetic probes, poloidal flux loops, an IPC computer, and phase controlled thyristor converters.

The mutual inductances among the poloidal coils are very small in HL-2A, normally they are considered to be zero. So the plasma current and its radial position can be controlled separately. In HL-2A, plasma current is controlled by the ohmic heating (OH) coils, in the meantime the radial position is controlled by the vertical field coils. These poloidal coils are fed by the 12-phase thyristor converters which are connected to a motor flywheel generator. OH coils and vertical field coils are located outside of the vacuum vessel in HL-2A. A model has been developed for designing a plasma radial position controller. Four equations are used to deduce the transfer function of the plasma radial position control in a thin conducting shell. First equation comes from Grad-Shafranov equation, assuming that the plasma current is constant. The penetration of the vertical field produced by the external control windings outside the shell should be considered, this makes the second equation. The third one is the circuit equation of the vertical field coils. The last one comes from the mathematical model of the controller. Taking the Laplace transformation, the transfer function of the plasma radial position control is obtained. In the plasma current control, the mutual inductances between the poloidal coils and plasma column should be considered. The density of the plasma current is assumed to be constant. The effect of the displacement of the plasma column on the plasma current is neglected. So the transfer function of the plasma current control can be obtained from the circuit equation of the plasma column and OH coils. The plasma current feedback control is implemented by a proportion-integral (PI) controller. In order to test the stability of the control system, the Simulink of Matlab is used. The feedback control system has been simulated in continuous and discrete way.

Simulation result shows the feedback control system is stable. The steady state error of the plasma radial position control is close to zero and the steady state error of its current control is about 0.1% ~ 0.5%. The control system is consisted of plasma radial position control and its current control, and it will be used in near future commissioning and first operation of HL-2A.

D-28

STRUCTURAL CHARACTERISATION OF SiC/SiC_F COMPOSITES EXPOSED TO CHEMICAL INTERACTION WITH Be AT HIGH TEMPERATURE

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Coated and untreated SiC/SiC_f composites were exposed to a long term annealing (550 hours) at 800°C in contact with Be pebbles. The annealing atmosphere was He + 0.1% H_2 simulating fusion reactor conditions. SiC/SiC_f composites and Be pebbles surfaces were characterised using SEM and Ion microprobe in order to study their chemical compatibility. Glancing incidence X-ray diffraction studies are underway in order to identify the crystalline phases on the composite surface.

The SEM results reveal localised attack on the surface of the coated samples. The composites without coat display a generalised attacked surface, more intense in the regions in contact with the Be pebbles. The corroded zones show the presence of oxygen and fluorine. Ion beam results of the same samples also show the presence of oxygen and reveals some carbon depletion in the outer surface.

On the other hand, the SEM studies of the Be pebbles show the formation of pits and an increase of F at the surface even if they are not in contact with the composites which indicates reaction with the flowing gas.

J-1

HEAT TRANSFER IN COMPRESSED BERYLLIUM PEBBLE BEDS

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Beryllium in form of single size pebbles with a diameter of about 1 mm is foreseen as neutron multiplier in the European helium cooled pebble bed (HCPB) blanket for fusion power plants. The pebbles are arranged in beds between flat cooling plates. During operations, the pebble beds and the containing structure can have different expansion. Therefore large compressive stresses could arise causing considerable plastic deformations of the pebbles and, due to the increase in the contact surface between pebbles, relevant changes in the thermal conductivity of the beds. For the proper thermal mechanical design of the blanket, the thermal conductivity of these beds as a function of deformation and temperature as independent parameters must be known.

Measurements of thermal conductivity and heat transfer coefficient to the containing wall of strongly deformed pebble beds are being performed at the Research Centre of Karlsruhe using the test section HECOP. In the experimental activity beryllium pebble beds are heated and uniaxially compressed, such that it is possible to adjust independently temperature and strain in the pebble beds.

Minimisation of uncontrolled heat losses, and reliable measurement of the temperature gradients in the bed guided the design of the HECOP facility, which has the following characteristics:

- Mechanical uniaxial pressure on the bed up to 6 MPa,
- Possibility to define the temperature gradient in the pebble bed,
- Average temperature in the bed up to about 650 °C.

In the present paper, after a short description of the experimental device, the status of the activity and the available results are presented and discussed.

J-2

EXPERIMENTAL TEST ON Li-CERAMIC BREEDERS FOR THE HELIUM COOLED PEBBLE BED (HCPB) BLANKET DESIGN

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The Helium Cooled Pebble Bed (HCPB) Test Blanket Module (TBM) for the DEMO Reactor foresees the utilisation of Lithiate ceramics as breeder in form of pebble beds. The pebbles are organised in several layers alternatively stacked among couples of Cooling Plates (CP). ENEA has launched an experimental programme for the out-of-pile thermo-mechanical testing of mock-ups simulating a portion of the HCPB-TBM. The programme foresees the fabrication and testing of different mock-ups, to be tested in the HE-FUS3 facility at ENEA Brasimone.

The first evaluation of the Lithiate breeder pebble bed thermo-mechanical behavior was performed on a simplified mock-up, called HELICHETTA, reproducing a rectangular portion of a TBM toroidal-radial breeder cell. The mock-up represents two pebble bed cells constituted by two lateral CP's with an intermediate flat electrical heater. The cell frame and the CP were made in austenitic stainless steel AISI 316L. The dimensions of each of the two cells are 100 mm in toroidal width, 485 mm in radial length and 10 mm in poloidal height. The total net volume of the two cells is 970 cm^3. The breeder ceramics tested during these campaigns were Li_4SiO_4 pebbles, produced by Schott-Germany, and Li_2TiO_3, produced by CEREM-CEA-France, both candidate as breeder for the reference helium cooled blanket design. The heat flux of the electrical resistors was regulated by a DC power supply (10 kW-50 V-200 A) up to a maximum value of 0.125 MW/m^2. This heater has a total thickness of 6 mm and was realized by KANTHAL A-1 flat strips in-between two INCONEL 718 cladding sheets, insulated by alumina plasma sprayed layers. Due to the mock-up geometry, the pebble temperature field, imposed by the electrical heater immersed in the pebbles towards the lateral CP, could be considered almost uniaxial in the toroidal-radial plane of the beds. The CP helium flow was controlled in a pressure range of 1.5–2.0 MPa and inlet temperature range of 250–300°C. The temperature distribution were measured by thermocouples positioned in the pebble beds at different locations and on the heater and CP surfaces. The pebble bed cells were closed by an end plug that could be loaded by an elastic cup spring (Schnorr) system. The pressure loads on the pebbles were measured by a load cell located on the plug. The displacements of the plug, its spring system and the CP's were measured by LVDT transducers.

The paper presents the results of several thermo-mechanical test campaigns carried out for measuring the breeder pebble temperatures and calculating their thermal conductivities and thermal expansions. Furthermore, comparisons among different mechanical boundary conditions (breeder cell lateral constraints) or thermal conditions (presence of an helium flow sweeping the pebble cell) are also presented.

J-3

INFLUENCE OF PEBBLE BED DIMENSIONS AND FILLING FACTOR ON MECHANICAL PEBBLE BED PROPERTIES

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For the characterisation of thermomechanical properties of pebble beds uniaxial compression tests (UCTs) are used for the determination of the strain-stress dependence, respectively, the modulus of deformation E_{def}, during load increase and load decrease. It is important that the results of UCTs are characteristic for blanket prototypical conditions.

The HCPB blanket is characterised by shallow beds (small bed heights H: breeder pebble bed heights ≈ 10 mm; beryllium bed heights ≈ 40 mm) compared to the other dimensions (toroidal width ≈ 1m; radial depth ≈ 0.5m). Another important pebble bed quantity is the filling factor γ which depends on the filling geometry (probably through individual tubes at the rear side) and the filling procedure (appropriate vibration of the bed). Presently, the conditions of blanket relevant filling procedures are not sufficiently known.

Detailed UCTs at ambient temperature have been performed using different cylindrical containers and lithium orthosilicate (pebble diameters between 0.25 and 0.6mm) and lithium metatitanate (pebble diameters ≈ 1mm) as granular material. The dependence of E_{def} on the bed height and bed diameter was determined for dense pebble beds (maximum values of γ). Additionally, the dependence of E_{def} on γ for constant bed heights was studied. It shows that E_{def} is strongly dependent on γ and at small bed heights is also significantly dependent on bed height.

In order to investigate pebble bed characteristics for more relevant blanket conditions, a test section was used with a cross section of 100 x 100 mm^2 and variable bed heights between 10 and 30 mm where the pebbles were filled in through a tube of 10mm at one corner. It showed that significant efforts will be required in order to obtain a homogeneous pebble distribution in large shallow beds.

J-4

QUALIFICATION OF TRITIUM PERMEATION BARRIERS IN LIQUID Pb-17Li

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The reduction of tritium permeation from the Pb-17Li, or plasma, into the cooling water is of crucial importance in order to reduce the radiological hazard in the turbine/condenser area and to optimise the tritium balance in the reactor.

The use of aluminium rich coatings, which form Al_2O_3 at their surface, has been selected as reference solution for the WCLL (Water Cooled Lithium Lead) blanket in order to produce reliable tritium permeation barriers (TPB).

TPB qualification activities performed in the past allowed the selection of two reference deposition techniques, the Chemical Vapour Deposition (CVD) process developed on laboratory scale by CEA, and the Hot Dipping (HD) process developed by FZK.

On the basis of the results obtained in the past with the Corelli I-II devices, a new apparatus named Vivaldi was designed to perform comparative tests on two hollow cylindrical specimens in the same operating conditions. The performance of aluminium coating on EUROFER 97 steel has been tested in gas and liquid metal phase. The permeation chamber was filled with gas and/or liquid lithium-lead, to measure the difference of hydrogen/ deuterium permeation rates between two specimens, a reference and a coated one, in gas and in liquid metal phase.

The obtained results in terms of permeated fluxes and permeation reduction factors (PRF) are herein presented and discussed, together with a description of the testing procedure. A post experiment examination of coatings was performed by use of optical and SEM microscopy. Also the composition of the coatings was estimated by mean of EDS analysis to verify the presence of aluminium-lithium interaction.

J-5

MEASUREMENT OF TRITIUM PRODUCTION RATE FOR A MODIFIED LITHIUM-6 ENRICHED BLANKET ASSEMBLY WITH D-T NEUTRON SOURCE

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Lithium-6 enriched ceramics belong to the most promising candidates for tritium breeder to acquire higher tritium breeding ratio of the D-T fusion reactor. The investigation of the tritium production performance of such a breeder is necessary. However there is no experimental data concerned with TPR measurement for blanket assemblies with the enriched lithium-6 breeder and the neutron multiplayer. Therefore experiments to measure TPR with a modified assembly, which consisted of enriched ^{6}Li in beryllium, have been carried out with D-T neutrons at Fusion Neutronics Source (FNS) at JAERI.

The experiments have been carried out using the FNS 80 degree D-T neutron source (1.7×10^{11} n/sec average). The scale of the assembly is about 500 x 500 mm^2 wide with a total thickness of 350 mm Three 12-mm thick ceramics layers consisting of 40% enriched $^{6}Li_2TiO_3$ are set up between 100- and 50-mm thick layers of beryllium in the assembly. The tritium target and the assembly were enclosed in a cylindrical stainless steel (SS-316) reflector and a 100-mm thick beryllium reflector was set up, across from the assembly. Enriched Li_2CO_3 and Li_2TiO_3 pellets, with sizes of ϕ13 and ϕ12 mm respectively and thickness between 0.5–2 mm, were used as the tritium detectors in each $^{6}Li_2TiO_3$ ceramics layers to measure the produced tritium. Nb, In and Au foils were also set up at important positions in the assembly to verfy the neutron field. After D-T neutron irradiation, the tritium in the pellets was measured with liquid scintillation counting methods.

Measured tritium distributions obtained from the Li_2TiO_3 and Li_2CO_3 pellets in the Li_2TiO_3 layers was reflected in the thermalized neutron field in the assembly and the local TPR of the surface area is significantly higher than the bulk and especially, the surface TPR is three five times more than bulk TPR in the third $^{6}Li_2TiO_3$ layer. Also from the comparison the measured value and calculated one with Monte Calro code MCNP-4B and JENDL-3.2 data, the C/E values of average TPR was between 1.2 and 1.4. Especially, the calculated tritium retention of the both pellet at the surface area was overestimated 40% more than the measured value. The overestimation of the TPR at the surface region is considered to be caused by uncertainty of low energy neutron flux in the beryllium, which may be due to the difference of the secondary energy and angular distribution of neutrons emitted through the $^{9}Be(n, 2n)$ as shown in Ref. 1.

Reference

[1] R. Tayama A, T. Tsukiyama, K. Hayashi, U. von Moellendorff, U. Fischer, H. Giese, F. Kappler, Fusion Engineering and Design 55 (2001) 365–372.

J-6

INFLUENCE OF IONISING RADIATION ON THE ELECTRICAL BEHAVIOR OF A Be PEBBLE BED

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The solid breeder blankets with lithium ceramics as breeder and stainless steel as structural material requires beryllium to increase the tritium breeding ratio (TBR) performance. In the event of plasma disruptions the currents induced in the blanket structure have to pass through the Breeding Blanket and their distribution and consequences will depend on the electrical resistivity of the entire module. Since beryllium pebbles occupy a large fraction of the module volume, and pure beryllium has a high electrical conductivity, it is important to know the electrical behaviour of the whole pebble bed. However, the oxide scale present on the beryllium pebbles surface, and the limited contact between the pebbles due to the bed porosity lead to a resistivity which is about two orders of magnitude higher than that of steel. The results so far obtained can be significantly altered in the presence of ionising radiation to which the pebble bed will be exposed during real working conditions.

In this work, the beryllium pebble bed was accommodated in a typical purge gas environment, using He + 0.1%H_2 as foreseen in the blanket design. This reducing atmosphere could have an impact on the oxygen distribution on the Be surface which can influence the resistivity behaviour of the pebble bed at high temperatures. Moreover the presence of a γ field can also increase the conductivity of the insulating Be oxide layer. To study this effect the pebble bed was placed in a γ radiation field ranging from $5x10^5$ Gy/h to $1.3x10^6$ Gy/h at the Portuguese Nuclear Reactor.

The results show a decrease of the resistivity values from 0.061Ω.m to 0.055Ω.m during the exposure in a γ field of $1.3x10^6$ Gy/h. The relatively small influence of the γ radiation on the resistivity is fundamental for its use in the future power plants.

J-7

COMPATIBILITY BETWEEN $Be_{12}Ti$ AND Li_2TiO_3

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Beryllides have good high temperature properties (mechanical property, chemical stability with steam, etc.) and are one of the candidate materials as the advanced neutron multiplier for DEMO-Reactor that requires good performance in the temperature from 400 to 800°C. On the other hand, the new blanket design with mixed pebble packing of the neutron multiplier and the tritium breeder has been proposed to improve TBR (tritium breeding ratio). Therefore, it is expected for beryllides to have good compatibility with tritium breeder to realize this new concept of blanket design. $Be_{12}Ti$ and Li_2TiO_3 are the most promising candidate material for neutron multiplier and tritium breeder, respectively. In this study, the compatibility test between $Be_{12}Ti$ and Li_2TiO_3 was carried out.

$Be_{12}Ti$ disk ($\phi 8 x^t 2$mm) specimens were fabricated by HIP process after mixing beryllium and titanium powder. Then, these were polished like mirror on the surface. Li_2TiO_3 disk ($\phi 8 x^t 2$mm, 83%T.D.) was fabricated by sintering process. Diffusion couple for this test consisted of the $Be_{12}Ti$ disk and Li_2TiO_3 disk. And these were inserted into Zry-2 container and set by tungsten spring in the Zry-2 container. High purity helium gas (6N) was sealed into the container by TIG welding. The compatibility test was carried out at 600, 700, 800°C by annealing in the vacuum furnace. Annealing periods were 100, 300 and 1000h. Same tests with Be were carried out to compare with $Be_{12}Ti$. After annealing, the interaction of the diffusion couple was investigated.

The amount of Li diffused in $Be_{12}Ti$ or Be in contact with Li_2TiO_3 was identified using ion micro analyzer and the thickness of reaction layer was measured. From these results, the thickness of the reaction layer in Be at 800°C for 300h and 1000h were about 200μm and 500μm, respectively. On the other hand, the reaction layer in $Be_{12}Ti$ at 800°C for 300h and 1000h were completely not observed. From these results, it was clear that the compatibility of $Be_{12}Ti$ with Li_2TiO_3 was superior to that of Be. A bright prospect to realize mixed pebble packing of the neutron multiplier ($Be_{12}Ti$) and the tritium breeder (Li_2TiO_3) was obtained.

In this symposium, the results of the structure analysis by the X-ray diffraction, the observation by scanning electron microscopy, etc. will be also reported.

J-8

EVALUATION OF EFFECTIVE THERMAL DIFFUSIVITY OF Li_2TiO_3 PEBBLE BED UNDER NEUTRON IRRADIATION

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Li_2TiO_3 is one of the candidate materials for the tritium breeder in the fusion blanket design. The effects of various parameters, i.e. irradiation temperature, sweep gas flow rate, etc. to the tritium release behavior were evaluated on the Li_2TiO_3 pebble bed. Additionally, the thermal characteristics of Li_2TiO_3 pebble bed are necessary to design the fusion blanket. In this study, the evaluation of effective thermal diffusivity of Li_2TiO_3 pebble bed under the neutron irradiation carried out.

The dimension of pebble bed container in in-pile mockup is $\phi 20mm^{ID} \times 260mm^{L}$. The total weight of binary packed pebble bed is about 170 g. Each diameter of pebbles is 0.3 mm and 2 mm. The theoretical density and the packing fraction of pebble bed were about 83%T.D. and 81%, respectively. This irradiation test was performed at Japan Materials Testing Reactor (JMTR). The effective thermal diffusivity was studied up to ~$9x10^{23}n/m^3$ (thermal neutron fluence) before each reactor startup and under neutron irradiation. The irradiation temperature of the Li_2TiO_3 pebble bed was controlled by the electrical heaters and the temperature of pebble bed was measured by the multi-paired thermocouples installed in this mockup. Each multi-paired thermocouples (T/Cs) were set at the distance of 0, 6 and 9 mm horizontally from the center of the Li_2TiO_3 pebble bed. The effective thermal diffusivity of pebble bed was measured under constant heating rate (2, 3 or 4°C/min). The center temperature of Li_2TiO_3 pebble bed was changed from 25˚C to 400˚C before each reactor startup.

From the results, it was obvious that the effective thermal diffusivity of Li_2TiO_3 pebble bed decreased with increasing the irradiation temperature or the thermal neutron fluence. However, difference (ΔT) of measuring temperatures was almost constant at 400°C under neutron irradiation. Therefore, it was also obvious that the effective thermal conductivity of Li_2TiO_3 pebble bed was not changed up to $6x10^{23}n/m^3$ (thermal neutron fluence) at least. Additionally, as to this in-pile mockup, the effective thermal diffusivity of Li_2TiO_3 pebble bed was constant even though the sweep gas flow rate was changed between 0 and $600cm^3/min$.

J-9

PRELIMINARY NEUTRONIC ESTIMATION FOR DEMO BLANKET WITH BELLYLIDE

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In Japan, the study of bellylide ($Be_{12}Ti$, etc) has been conducted actively for the development of neutron multiplier. Then, good properties, i.e. low tritium inventory, good compatibility with tritium ceramic breeder, etc have been obtained. Therefore, two kinds of preliminary neutronic estimation were conducted as to the fusion blanket in which bellylide was packed.

First purpose of this estimation is to make clear the effect of neutron multiplier materials (i.e. Be, $Be_{12}Ti$, $Be_{12}V$ and $Be_{12}W$) on Tritium Breeding Ratio(i.e. TBR). Second purpose of this estimation is to make clear the effect of pebble packing condition of tritium ceramic breeder and neutron multiplier (separated type or mixed type) on TBR. The neutronic calculations were performed using the two-dimensional discrete ordinates code DOT3.5 with multi-group neutron and gamma cross-section set FUSION40 that was preceded from JENDL 3.2. The design terms are shown as follows: Neutron wall loading at the first wall is $5MW/m^2$, Packing fraction of pebbles is 80% (binary packing), Structural material is ferritic steel (F82H), Coolant is pressurized water (15Mpa/320°C). Li_2TiO_3 was selected as tritium ceramic breeder.

From these estimations, it was obvious that TBR on mixed type blanket is ~1.2 times of that on separated type blanket. TBR on mixed type blanket with $Be_{12}Ti$ or $Be_{12}V$ were ~1.2 and 90% of the TBR on mixed type blanket with Be, Then, it is considered that $Be_{12}Ti$ or $Be_{12}V$ are one of most useful neutron multipliers from a point of TBR. In this symposium, the relationship between beryllium atom density and TBR will be presented.

J-10

KINETICS OF TRITIUM RELEASE FROM IRRADIATED Li_2TiO_3 PEBBLES BY OUT-OF-PILE TPD TESTS

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Lithium metatitanate pebbles are being developed for tritium-breeding in several fusion reactor blanket projects. Their tritium release property as detected by fast and cheap post irradiation tests is considered a useful screening tool for the materials R&D, although not relevant for engineering design. In this work tritium release from irradiated Li_2TiO_3 specimens, mainly obtained through the "Temperature Programmed Desorption" (TPD) technique, is investigated. Observed TPD signals are generally characterized by a main broad peak. A versatile peak de-convolution software has been developed to retrieve the kinetic parameters of the overlapping tritium release steps. Results from single heating ramp tests are validated against independent TPD-diagnostic criteria, which are usually based on tests performed at various heating rates β. Moreover, validation against isothermal annealing desorption tests is also carried out. The following reaction scheme is considered to describe the tritium removal from irradiated pebbles:

$$H_2 + O^*_{ads} \rightleftarrows H_2O_{(gas)} \text{ and/or } H_2 + O_{(lattice)} \rightleftarrows OH_{ads} + H_{ads} \rightarrow H_2O_{(gas)} + V_O \text{ (reduction)} \quad (1)$$

$$H_2 + OT_{ads} \rightleftarrows HT_{(gas)} + OH_{ads} \text{ (isotope exchange)} \quad (2)$$

$$OH_{ads} + OT_{ads} \underset{k_{-2}}{\overset{k_2}{\Longleftrightarrow}} HTO^*_{ads} + O^*_{ads} \xrightarrow{k_d} HTO_{(gas)} + O^*_{ads} \text{ (desorption)} \quad (3)$$

where the constant k_{-2} characterizes the tritiated water dissociation rate, k_2 its recombination rate and k_d its desorption rate. The constant k_2 may also be interpreted as a mobility rate constant preceding the hydroxides recombination into activated HTO* complex, therefore determining tritiated water desorption under diffusion control. The main assumptions of our analysis are: i) we fix the composition (He + 1000 vpm H_2) of the purge gas, whose flow rate and very low moisture level affects steps (1) and (2) always in the same way; ii) we assume that the isotopic exchange (2) is so fast as to assure equilibrium at the pebbles surface; iii) we assume that the rate-determining step (3) is of pseudo first order type, and that the kinetic constants are of Arrhenius type ($k = A \exp(-E/RT)$ [s^{-1}]).

The chemical interaction (1) between the O-atoms of Li_2TiO_3 with the purge gas (due to its H_2 content) is found to affect the tritium release rate (3) only above 600°C. Within these limits, the TPD data-spectra elaboration provides the parameters A and E which determine the kinetic constants for the main sites or modes characterizing each pebble batch/specimen. These depend on texture, microstructure and/or thermal annealing and irradiation history of the specimens. The results are found to be self-consistent with the time decay constants of tritium release rate ($\tau \sim 1/k$ [s]) as determined by isothermal annealing step tests performed both ex-situ and/or in-situ on the same type of irradiated pebbles. The presented fast and single-run out-of pile tritium release TPD tests might also be useful for a quantitative evaluation of Li_2TiO_3 pebble batches main functional properties.

J-12

LAY-OUT OF THE He-COOLED SOLID BREEDER MODEL B IN THE EUROPEAN POWER PLANT. CONCEPTUAL STUDY

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The European Helium Cooled Pebble Bed (HCPB) blanket concept is the basis for one of two limited-extrapolation plant models that are being elaborated within the European Power Plant Conceptual Study (PPCS). In addition to addressing the case for fusion safety and environmental compatibility, following earlier studies like SEAFP or SEAL, this reactor study puts emphasis on plant availability and economic viability, which are closely related to specific plant models and require a detailed lay-out of the fusion power core and a consideration of the overall plant (Balance of Plant).

Within the development of in-vessel components for the plant model, the major tasks to be carried out were: (i) adaption of the HCPB concept – featuring separate pebble beds of ceramic breeder and Beryllium neutron multiplier, and reduced-activation ferritic-martensitic steel EUROFER as structural material – to the large module segmentation chosen for reasons of plant availability in part II of the PPCS; (ii) proposal of a concept for a Helium cooled divertor compatible with a maximum of 10 MW/m^2 heat flux to satisfy the requirements of reasonably extrapolated plasma physics; (iii) lay-out of the major plant model components and integration into the in-vessel dimensions found from system code calculations for a power plant of 1500 MW electrical output and iterated data on the plant model performance.

The paper defines all major in-vessel components of plant model B, as it is called in the PPCS, namely (i) the unit of FW, blanket and high temperature shield that is to be replaced regularly; (ii) the low temperature shield that is laid out as a lifetime component of the reactor; (iii) the divertor; and, (iv) the in-vessel manifolding. Results are presented for the thermal-hydraulic performance of the components, and for the thermal-mechanical behaviour of the blanket and the divertor target plate. These results suggest, together with results from the wider exploration of the plant model within the PPCS, that the He cooled solid breeder blanket is a credible concept. They stress the positive role that He cooling can play in economically attractive fusion power plants.

J-13

THERMOHYDROMECHANICAL EVALUATIONS OF Li-BASED BREEDER PERCOLATION RATES THROUGH SELECTED SiC-FIBRE STRUCTURES

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A multipurpose Finite Element is being developed to compute multi-phase/multi-species Liquid Metal (LM) flow rates through coupling with porous materials thermo-mechanics (TM).

For a large variety of reactive and non-reactive porous materials, fluid characteristics and operational conditions the code is being tested. The general program solves gas/liquid and linear momentum mass balance equations together with the consistent TM material constitutive equilibrium conditions. Thus, its 3D state variables are: *solid displacements*; *liquid pressure*, *gas pressures* and *temperatures*.

Darcy's law for gas/liquid phases under hybrid viscous drag flow/capillary regimes is the theoretical basis for percolation. Among others, the dissolution of bred gases in the liquid phase (Henry's constant scheme) and Fick's gas diffusion in the liquid/solid phases are the gas transport mechanisms considered. The allowed by the code constitutive mechanic characteristics are the most general thermo-elasto-plastic (TEP) model. Phase equilibrium between LM and its pressure vapour is assumed. Balance of momentum for the medium as a whole is reduced to the equation of stress equilibrium together with a mechanical constitutive model to relate stresses and strains. Strains are defined in terms of displacements and assumed as small for solid deformation. Eulerian time derivatives approximate solid displacements being the volumetric strains properly considered.

The detailed code features are presented. The tool advance large capabilities for the analysis of tritium/helium problems related to the design of fusion technology LM units as LM/SiC_f-SiC FW concepts and divertors, self-cooled LM SiC_f-SiC blankets, etc.

J-14

INFLUENCE OF BOUNDARY CONDITIONS ON PENETRATION OF ISOTOPES OF HYDROGEN THROUGH STRUCTURAL MATERIALS

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Investigations of the hydrogen penetration through structural materials under simulation of thermonuclear reactor working conditions are very important in connection with the problems of safety and solving the material science problems of controlled thermonuclear fusion.

In the given work results of research of influence of boundary conditions and of some structural factors on penetration of hydrogen through structural materials during bombardment by ions of isotopes of hydrogen are presented. The experiments were carried out in the temperature 100–300^{o}C range. The energy of these light ions used for irradiation was 30 keV, while the irradiation doses varied from 1×10^{21} m^{-2} to 1×10^{23} m^{-2}. It was revealed, that hydrogen penetration through materials strongly depends on boundary conditions (penetration from a gas phase, from plasma of the glow discharge, while bombardment by hydrogen ions). Preliminary implantation of interstitial atoms (O, N, C) into a superficial layer of a sample significantly effects on penetration of hydrogen through materials under study.

Results of the study of influence of diamond like coverings of different thickness on hydrogen permeability of structural materials during their bombardment by hydrogen isotopes ions are presented. It is revealed, that penetration of isotopes of hydrogen through similar materials strongly depends on a ratio of thickness of a film and length of run of bombarding ion in the being studied material.

Results are presented of investigation of hydrogen penetration through structural materials with the given structural characteristics during bombardment by the accelerated hydrogen ions. It is established, that preliminary cold deformation as well as reduction of the size of a grain raises permeability of hydrogen through investigated materials during their bombardment by hydrogen ions.

J-15

ANALYSIS ON STRESS FIELD OF THE DUAL-COOLED WASTE TRANSMUTATION BLANKET FOR THE FDS

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The Fusion-Driven Sub-critical System (FDS) is one of the Chinese concepts to be further developed for fusion application. It will provide a feasible, safe, economic, and highly efficient potential of disposing High Level Waste (HLW). Its Dual-cooled Waste Transmutation Blanket (DWTB) cooled by helium and liquid metal take the features of safety, tritium self-sustaining, high efficiency and feasibility as one of the most important part of the FDS.

This paper is mainly on the stress field numerical calculations and analysis. Taking into account the electromagnetic and neutronic loads on different conditions, DWTB structure needs to have capability to remove tremendous thermal power. In order to deal with the thermal-stress of the dual cooling system (helium and liquid metal) for the inboard and outboard of the blanket, it is necessary to calculate the DWTB structure stress field.

The following aspects seriously account in the DWTB structure stress field calculations and analyses:

1) Setting up finite elements model for DTMB structure.
2) Selecting coolant and transmutation materials (helium, Pb-17Li and nuclear wastes) and deciding proper characteristic parameter.
3) Taking into account the electromagnetic and nuclear power distribution loads, which were applied to a finite elements model of the DTMB structure.
4) Ascertaining the boundary conditions.
5) Application of the finite element computer code ANSYS and FLUENT are used to determine the stress field of the DWTB.

Besides the loads assumed by the design specifications (pressure, temperature, etc), electromechanical and thermal loads have been evaluated. The thermal loads have been evaluated considering the heat source from the plasma and from the fission in the blanket.

Finally, the 3-Dimension numerical stress field of the DTMB has been presented. Its structure has been optimized according to the above calculations and analysis.

J-16

NUMERICAL SIMULATION OF TRANSIENT PROCESSES IN ITER AS 3D COUPLED PROBLEMS

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To ascertain a design concept and optimize key components of a tokamak device a comprehensive study is required which involves electromagnetic, thermal - hydraulic and structural analyses. As a first step, a global numerical model is developed for an EM analysis to investigate transient behaviour of inductively coupled components. Calculated distributions of ponderomotive forces, thermal loads, eddy currents, and magnetic fields are used to develop models for subsequent thermal - hydraulic and structural analyses. A detailed full-scale 3D solid body model for the EM analysis would require excessive computation resources. As a compromise, a full-scale finite-element model described with thin conducting shells arbitrarily located in space is recommended at this stage. The problem is formulated with respect to a single unknown, the normal component of electric vector potential. The EM analysis of basic ITER machine components was made with the use of the TYPHOON code developed at the Efremov Institute as a part of ITER activities. Transient behaviour and loads were calculated for the vacuum vessel, thermal shields, divertor, port-limiter and other structures. Transient behaviour of the EM system can be analysed for various operation conditions, including nominal operation and plasma disruptions. The effects of plasma filament movement, halo currents, variations of the toroidal magnetic flux and plasma and coils currents are considered. Different physical models are combined on the basis of a common reference finite-element mesh adjustable to a specific model. Special procedures are proposed to adequately transfer physical values onto a finite-element meshes. This enables the use of different software products for the global analysis. Transferring of the EM force density means a correct calculation of $\int_{V_e} N_i \vec{f} dV_e$, where $\vec{f}$ is the force, N_i is the shape function for finite elements, V_e is the element volume, and generation of a system of equivalent nodal forces in the ANSYS format suitable for the structural analysis. Similar procedures are developed to transfer thermal loads for the thermal-hydraulic analysis. A shell model is applicable for the analysis of a temperature field inside the vacuum vessel. A VV wall is modelled as a thin shell, and an average temperature distribution through the vessel is calculated with a 5% accuracy. The boundary condition of the third kind in the shell model is described with equivalent heat sources. The heat transfer coefficient is replaced with its effective value. Illustrative calculations of the ITER vacuum vessel are presented. The developed self-consistent calculation technique enabled the comprehensive simulation of transient processes in the ITER machine. The results were included in the final Design Description Document.

J-17

EM ISSUES ON THE ITER IN-VESSEL COMPONENTS DURING PLASMA DISRUPTIONS

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Introduction - The EM loads driven by plasma disruptions are one of the most concerning issue for the in-Vessel engineering. For design purposes six reference disruption events have been defined. Considerable design, analysis and R&D effort has been spent to obtain in-vessel components capable of withstanding the EM loads induced by these events. In support of such efforts both extensive and very detailed EM analyses have been needed. To be effective in this support, contrasting features for the analyses were required: very accurate component modeling, precision in describing the EM transient and very short computing time. These competing objectives have been achieved using a zooming procedure developed at ENEA that has allowed to run the large number of cases needed to select the most performing design option for each component. The aim of the work is to present an almost complete picture of the main EM issues for the in vessel assembly and to make evident which components should still require some further R&D effort in relation to the EM problems.

Work Content - The analyses have been performed in presence of the most dangerous plasma disruptions that can be predicted for the ITER machine. The input excitations coming from simulations performed using time evolving MHD equilibrium codes have been critically examined. The EM loads induced by the ITER reference plasma disruptions have been evaluated for the following in Vessel components: the divertor, the ICH assembly, the equatorial port Limiter assembly, the shielding blanket modules, the Test Blanket Modules. For each component the analyses were aimed, together to a precise assessment of the EM loads, to an investigations of the effectiveness of their geometrical features in reducing these loads. Using the zooming approach only a restricted region including the component under investigation has been analyzed allowing a good model detail without loosing accuracy in describing the EM transient. The field inaccuracy in the region of interest did not exceed 5% in the worst case. The 3-D EM code EMAS that allows the treatment of ferromagnetic materials, that are present in the TBM, has been used for the analyses. The running time of the most complex EM model do not exceed 15’ of CPU time on a normal desktop computer. It should be noticed that an equivalent 3-D EM model including the whole machine sector involved by the EM transient would require some days of CPU time on a performing workstation.

Conclusions – The most critical in-vessel components are indicated; some design variants compatible with the other main design philosophy of the component are suggested, to reduce these loads. A modification of the last current quench phase according to most of experimental evidence from the main today working tokamaks is suggested that could lead to less pessimistic and more realistic estimate of the loads. The effect of such modifications is shown in some cases of interest.

J-18

FULL SCALE ELECTRICAL INSULATION COATING DEVELOPMENT

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A de-mountable mechanical system has been designed for the attachment and the remote handling of the blanket modules. The modules are attached to the supporting structure by bolting using four flexible radial supports (cartridges). Mechanical keys are used to withstand the electromagnetic forces. The modules are electrically insulated from the adjacent structures, except in the central region of their rear side, where the hydraulic connections are located and electrical straps are used to provide an electrical path for halo currents. Coating the contact surfaces between the attachment fixtures with alumina provides the electrical insulation.

The flexible attachment system consists essentially in a bolt screwed in a stainless steel structure with the interposition of an insulated cartridge and a collar. The insulations are applied in the interfaces between the stainless steel base, the cartridge and the collar.

The objective of this work was to test the adequacy and performance of the electrical insulation coatings specifically for the ITER blanket mechanical attachment design. The testing campaign has required the manufacture of mock-ups and the application of alumina coating by plasma spray technique. The mock-ups had to be mechanically and thermally cycled under conditions applicable in ITER. Once preloaded, the flexible attachment system has been mechanically cycled with a load of ±500kN in room conditions. Another preloaded assembly has been thermally cycled between 150ºC and 300ºC for 1000 times in inert gas atmosphere and successively has been mechanically cycled for 1000 cycles. After testing all the components and the coatings have been checked and investigated. The activity included also a mechanical test of two wedged keys, to verify the behaviour of the alumina coating under an impact load of 500kN applied for 1000 times.

The testing results show a good mechanical and thermal behaviour of the alumina coatings. No failures and damages occurred on the coatings which preserved a good insulating performance after testing confirming the adequacy of the electrical insulation coating for the ITER blanket system.

J-19

MHD PRESSURE DROP IN FERRITIC PIPES OF FUSION BLANKETS

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Magnetohydrodynamic (MHD) flows in pipes made of ferromagnetic material is an important issue for liquid metal blanket concepts using MANET as wall material. The MANET steel has a relative magnetic permeability up to 400 and considerably large magnetic saturation of $M_s \approx 1.65$ T, which is of similar order of magnitude than the externally applied magnetic field B_0=6-8 T required for plasma confinement. This magnetic field is responsible for high pressure drop due to the electromagnetic flow-braking in regions of the blanket where the field's transverse component to the flow direction is high.

MHD flow in ferritic pipes is studied analytically and experimentally. The analytical model suggests that the pressure drop of such MHD flows confined by massive ferritic structures follows classical laws if the effective field in the pipe is known. More precisely the pressure drop in such configurations is reduced by a factor α^2□, where α stands for the ratio between the internal field that is reduced by magnetic shielding and the externally applied field. The theoretical predictions are in good agreement with recent experimental data performed in the nonlinear magnetization regime.

For fusion relevant conditions, an extrapolation of the present data suggests a reduction of pressure drop by magnetic shielding of about 10%. The pressure drop could further be decreased by reducing the induced current density in the fluid. In fact those induced currents short-circuit through the well conducting thick walls resulting in high MHD damping of the flow. Electrical decoupling of the liquid metal from the conducting wall minimizes currents and associated pressure drop. In real conditions the insulating layer (Al_2O_3 for instance) has to be protected from direct corrosive attack by the liquid metal by a thin sheet of steel. Since this sheet is relatively thin it yields high Ohmic resistance and does not increase currents drastically. Such a test section was fabricated by spaying a few micron insulation layer of Al_2O_3 on a thin walled pipe. The latter one was inserted in a thick block of ferritic MANET. Pressure drop measurements performed between the inlet and outlet confirm the effective contribution of the electrical decoupling.

J-20

A BREEDING BLANKET IN ITER FEAT

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ITER FEAT at beginning of life will have only a shielding blanket, however many advantages can be envisaged in having, at least in a later phase of the operating life, a breeding blanket in the machine. The presence of a breeding blanket would indeed give useful information on the behaviour of solid breeder blankets in a future advanced machine and, moreover, would permit to produce directly tritium, eliminating the problems and the costs related to its transport.

The studied breeding blanket concept, developed from previous designs of breeding blanket for ITER, is water-cooled and both breeder and multiplier are present in form of single sized pebble beds. These beds are used in order to transfer thermal energy from the breeder (where the most part of energy is generated) to the coolant. The lithium ceramic breeder (e.g. Li_2ZrO_3) is in cylindrical tubes and is surrounded by the multiplier (beryllium), which is cooled by water flowing in perforated plates in strict contact with the pebble bed.

Since several years, studies are being devoted to the heat transfer in pebble beds and, in particular, the influence on heat transfer of the compaction of the pebble beds and of the dimension of the contact area between pebbles is being investigated. . As the experimental activities aiming to obtain the relationships between thermal parameters, temperature and stresses in the pebble beds in relevant temperature range have not been yet completed, a sensitivity analysis (635 %) has been performed in order to verify the variations of breeder and multiplier temperatures in the breeding blanket assuming different thermal conductivity for the beryllium pebble bed. The performed thermal analyses show that a breeding blanket in ITER-FEAT can be operated keeping the minimum breeder temperature at such a level that the generated tritium is released and can be purged in a continuous operation. On the other side, the maximum multiplier temperature is kept at a value that does not lead to dangerous chemical reactions with the coolant in case of accident.

The neutronic analysis indicates that a local TBR of 0.84 is achieved. The extrapolated 3-D value is 0.47 when only the outboard coverage of the blanket is available. An increase of 15% could be obtained if a further tube row containing the breeder would be added to the blanket. Nevertheless the tritium self-sufficiency is not a requirement for the breeding blanket in ITER-FEAT.

The obtained results indicate the possibility of having a breeder blanket with a significant TBR in ITER-FEAT. Such a blanket could allow producing tritium during the ITER operating life, while satisfying the safety requirements of the reactor.

J-21

DESIGN AND R&D PROGRESS OF BLANKET ATTACHMENTS

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The attachment of the ITER blanket modules to the vacuum vessel wall includes the mechanical support system, the electrical strap, and the hydraulic connection to the coolant manifolds. All these features constrain the design of the modules and the interface of the vacuum vessel. A significant effort has been devoted to their design to improve the size, to include modifications from the test results, to simplify the manufacturing or to satisfy higher loads, coming from additional plasma scenarios.

The tightening and release procedure for the M45 Inconel 718 bolts of the four flexible supports has been established experimentally by thermal expansion and fixed rotation. The 650 kN pre-load is controlled by the stem extension through the ultrasonic measurement of the bolt length. The anti-rotation keys of the inboard blanket have been rearranged in between the modules to achieve a wider span, thus a higher strength for 1 MNm torque. Their number has been reduced from 36 to 18 and their fabrication is envisaged as block welded on a collar forged from the vessel plate to reduce the weld amount and the cost.

The electrical straps which ground the module on the vessel and discharge up to 280 kA halo currents has been tested with a full scale prototype inside a large solenoid at 7T. No instabilities or irregular distortions of the CuCrZr cold-formed 2.5 mm sheet appeared. The fatigue strength for 10.000 displacements of 1mm in 3D has also been assessed. Increased heating appeared under the single M20 closure bolt due to variation in contact resistance. It was improved by a flexible bar ensuring an even compression in the joint.

Welding, cutting and re-welding of the coaxial hydraulic connection with YAG laser has been developed for module installation and replacement. A toroidal cavity sealed by a metal gasket has been introduced outside the weld for leak checking. Inside, a telescopic sleeve separates the inlet from the outlet stream.

The paper reports the analyses in support of the design progress and describes the experimental activities required for the development of the blanket attachments and for the assessment of its efficiency.

J-22

DEVELOPMENT OF INTERIM STRUCTURAL DESIGN CRITERIA FOR ITER MULTI-LAYER COMPONENTS

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In-vessel ITER components, being in direct contact with the plasma, have to withstand concentrated heat fluxes and thermal shocks, dynamic loading and high radiation influence. ITER designers have to apply heterogeneous multi-layer structures in order to distribute operating requirements among the layers and provide necessary strength and life-time requirements by means of appropriate combination of materials such as beryllium, copper (bronze), stainless austenitic steel.

In accordance with well-established international practice, strength design elaboration of ITER reactor components shall be performed in two stages: firstly, basic dimensions selection and checking strength calculations; secondary, checking fracture analysis.

Multi-layer application is not typical design and engineering solution in the reactor designing. Experience in using such materials is quite limited. Strength analysis methodology for multi-layer ITER structures has been developed and presented in design-intended document named as "ITER Interim Structural Design Criteria”. This document contains necessary design basis as applied to the bulk defect-free materials, formed heterogeneous components. Further development of this document has been focused on rules (guidelines), which describe fracture mechanics applications. The following results on development of temporary criteria for designing of in-vessel high heat flux ITER multilayer components (IMC) were reached.

The basic cases of singularities for standard zones of IMC were analyzed and the analytical data on values of singularity parameter were obtained. The parameter is a function of geometry of interfaced materials and their mechanical properties. The procedure of stress-strain state calculation for ITER multilayer components were developed. The requirements to the sizes, location and singularity parameters for defects postulated during an analysis of the standard zones were determined. The criteria approaches to the analysis of multilayers on resistance to the brittle fracture were formulated. The methodology for calculation of ITER multilayer components on crack resistance were developed, the methodology were implemented to design calculations of ITER limiter. The section in ISDC on criteria for designing of multilayer components with discontinuities and singularities were developed.

Thus, it is possible to consider, that to the present time basically the methodology for the analysis of integrity and operationability of IMC with use of linear fracture mechanics were developed.

J-23

DESIGN OF PARALLEL INTERSECTOR WELD/CUT ROBOT FOR MACHINING PROCESSES IN ITER VACUUM VESSEL

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The international thermonuclear experimental Reactor (ITER) is the multinational project for next step fusion demonstration plant. It is part of a research program aimed at demonstrating the technologies essential to a fusion power reactor and testing of critical components required to enable fusion energy to be utilized for future power production.

ITER sectors are very large component with some complex geometrical features. Plasma physics and internal component assembly require more stringent tolerances than normally expected for the size of structure involved. Overall assembly tolerances are expected to be within ±5 mm in the whole vacuum vessel. The outer walls of ITER sector are made of 60 mm thick stainless steel. They are jointed together through efficiency structural and leak tight welds. In addition to the initial vacuum vessel assembly, sectors may have to be replaced for repair. Since the commercially available machines are too heavy, they are difficult to be used as hose machines to carry out machining and welding inside the vacuum vessel. Flexible, light-weight and mobile robotic machines should be considered.

The traditional industrial robots have been used as general purpose positioning devices and are open chain mechanisms that generally have the links actuated in series. These kinds of manipulators usually have a long reach and large workspace. However, they are inherently not very rigid and have poor dynamic performance at high speed and high dynamic loading operating conditions. Compared with open chain manipulators the parallel mechanisms have high stiffness, high accuracy and high force /torque capacity, and have been found many applications in manufacturing system. Despite this, there are no commercial solutions applicable in ITER environment available in the markets. The required force density can be achieved only by using hydraulic drives, therefore a special design is required.

This paper presents a new parallel robot which has five-degree freedom driven by the hydraulic cylinders. The proposed mechanism has large singularity-free workspace and high stiffness. As the flexible functional property of parallel robot, it can carry out both the machining and welding processes inside the ITER vacuum vessel. The kinematic structure of parallel robot particularly suitable for ITER environment is presented. The analysis of machining process for the ITER , such as machining method and machining force are given. The kinematic analysis, such as workspace, singularity and accuracy are discussed. Finally the experimental demonstrator model is presented.

J-24

BUCKLING ANALYSIS OF THE COMPLETE W7-X PLASMA VESSEL

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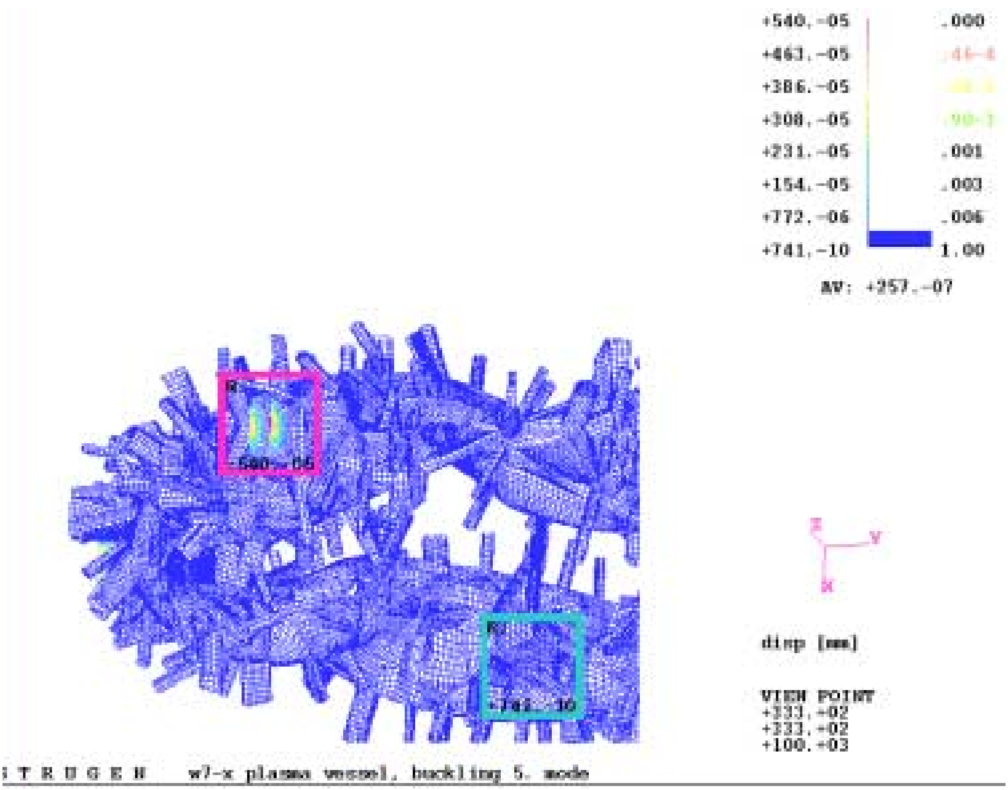
A buckling analysis of the complete W7-X plasma vessel (inner vacuum vessel) has been performed. The magnet of the W7-X stellarator consists of five modules each containing five types of nonplanar and two types of additional coils [1]. Two vessel systems outside and inside the toroidal assembly of the different coils confine the vacuum for cryogenic operation. The shape of the plasma vessel is defined by the geometry between the surface of the plasma and the inside surface of the nonplanar coils. Subject of the analysis is the buckling behavior of the complete plasma vessel including the individual nozzles for diagnostics. Different operating cases are shown together with special applications to get informations with more detail. A zoomed result of the undisplaced 5th mode of the plasma vessel is shown in the figure below representative for other calculations.



W7-X plasma vessel, loading case 3, buckling 5th mode, displacements

[1] Jaksic, N. et al. Mechanical stresses of the W7-X coil set with reduced structural weight, 18th Symposium on Fusion Technology, Karlsruhe, Germany, 1994, Elsevier, Amsterdam, 1995, pp 937 – 940.

THE VACUUM VESSEL THERMAL SHIELD OF THE KSTAR TOKAMAK

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KSTAR has an all-superconductor magnet system and inevitably needs a thermal shield to cut off 300K radiations from the other components of the room temperature. The vacuum vessel thermal shield (VVTS) is placed in the narrow gap of minimum 80 mm between 4 K TF magnets and the 300 K vacuum vessel. The VVTS is designed to be divided into 16 assembly units of 22.5° sector, each unit has an electrical insulation along the center line in the toroidal direction and 4 insulations in the poloidal direction to reduce eddy currents induced during plasma operations. All connections are bolted. The VVTS becomes consequently a rigid torus composed of 64 electrically-insulated pieces.

A key point of designing the VVTS is that supports of the VVTS are to be flexible enough to resist thermal constriction during cooling down by 70 K from room temperature as well as sufficiently strong to withstand electromagnetic forces exerted on the VVTS during plasma disruptions. Main supports of the VVTS which are installed on the equatorial plane at the rate of three per sector and adopt the leaf spring type to cope with thermal constriction, stand up to mainly self-weight. As a whole, the cooling panels of the VVTS bear most mechanical forces. Some additional semi-flexible supports may be attached for convenience in assembly.

The cooling panel is of quilted plate type whose total thickness is 12 mm, and the period of the lattice pattern is 60 mm. The connection of adjacent panels will be done through a stainless-steel step welded on a panel. Prototype cooling panels were manufactured for simulating the fabricated process of curved quilted plate.

The stress analysis is carried out with a 67.5° model consisting of three sectors, and the stress of the support number change of mid-sector and each VVTS is used for evaluation and comparison.

J-26

MANUFACTURE OF CRYOSTAT COMPONENTS FOR WENDELSTEIN 7-X

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WENDELSTEIN 7-X (W7-X) is a follow-up of the successful stellarator W7-AS and is presently being built at the Greifswald branch institute of IPP. One of the main parts of the stellarator is the cryostat which provides the thermal protection of the coil system. Construction of the cryostat of W7-X is based on the experience gained during design, construction and test of the DEMO-cryostat. The design of the main components of the cryostat, in particular the plasma vessel, the outer vessel, the ports and the support structure is finished and manufacture by European industry has started.

The plasma vessel and the outer vessel are being manufactured by the German Deggendorfer Werft und Eisenbau GmbH (DWE). The plasma vessel is constructed by welding steel stripes which have been bent to the required curvature. Most of the plate stripes will be formed by chamfering the segments. To achieve the complicated free-form curves of the plasma vessel sectors, DWE had to realise free form compression-moulding. The challenge during manufacture is to keep the small tolerances while considering shrinkage of the welds. In preliminary tests DWE successfully assembled several ring segments and thus demonstrated the feasibility to reproduce the complicated 3-dimensional shaped plasma vessel. The outer vessel has a large number of openings for ports, manholes and feedthroughs. The design of weldseams and the dimensions of flanges is supported by structural calculations.

Manufacture of the 309 ports is done by the Swiss company Romabau. After design has been completed and some material and component tests have been successfully performed production has started end of 2001.

The plasma-vessel rests on 15 adjustable vertical supports. Horizontal fixation is provided by 5 special supports, which allow for the thermal contraction of the plasma-vessel during heating while ensuring selfcentring of the structure. Furthermore the supports will allow to fine adjust 7-X.

The poster will summarise the design and give short descriptions of the status of construction of the main cryostat components. In particular the construction of the plasma vessel will be shown in more detail.

J-27

ELECTROMAGNETIC STUDY OF THE ITER VACUUM VESSEL

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As primary functions the ITER Vacuum Vessel (VV) provides a high quality vacuum for the plasma, as well as the first confinement barrier of radioactive materials, and a second barrier (after the cryostat) for the separation of air from potential sources of in-vessel hydrogen generation.

Vacuum vessel is a torus-shaped, double wall structure with shielding blocks and cooling channels between walls. The blanket modules are internal to the VV and attached to the VV inner shell. The VV has upper, equatorial and lower port structures. Particular attention in this study has been focused on these components. Two separate models of the VV according to different types of equatorial ports (regular or heating/diagnostic neutral beam (H/DNB)) have been studied. The main task was to determine the 3-D load pattern in these structures. The NB ports provide access for the neutral beams for the plasma heating and current drive, and the diagnostics. The three NB ports are located side by side at three equatorial port locations. The calculation model includes the double-wall VV with different port structures (regular and H/DNB) and stiffening ribs, regular and unique (placed on the outboard area of VV shell by the H/DNB port side) blanket modules, plasma, Poloidal Field (PF), and Toroidal Field (TF) coils. The effect of the TF ripple is also taken into account. In this electromagnetic study, two different models of the VV upper and central port plugs (with and without taking into consideration the electrical break between each front and back plug portions) have been considered. Electromagnetic loads, applied to the VV due to different scenarios of plasma behavior such as the fast Central Disruption 27ms, fast upward/downward Vertical Displacement Event (VDE) were calculated for every calculational model. Two different transient electromagnetic processes were modeled separately: one process caused by a variation of the toroidal plasma current, shape and position, and the another one caused by flow of the Halo currents and a variation of the toroidal magnetic flux created by the plasma. Afterward they were superimposed in automatic mode to obtain the local and total electromagnetic loads applied to the VV components. The calculations have been carried out using the TYPHOON code, developed at the Efremov Institute. The TYPHOON code solves problems related to quasy-stationary transient electromagnetic processes in thin multi-connected conducting shells, arbitrary located in 3-D space, using an electrical and magnetic vector potentials and is suitable for structures with the skin time rather small compared with the characteristic time values of the external field variations. TYPHOON code is based on the finite element method to solve 3-D problems.
The obtained data were used in subsequent thermal-hydraulic and dynamic structure analyses. These results were included in the Design Description Document of the ITER tokamak and used to define the design loads of the ITER Vacuum Vessel and Blanket.

J-28

THERMAL ANALYSIS OF ITER VACUUM VESSEL THERMAL SHIELD (VVTS) COOLDOWN PROCESS

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There is a wide range of problems related to the cooling (heating) of massive structures with the complex geometry and cooled by gaseous or liquid coolant flowing inside the long tubes. As a rule, the cooldown process in this kind of problems is characterized by the relatively small ratio of the coolant total heat capacity to the heat capacity of the massive structure. The latter results in the coolant fast heating up close to the structure temperature. It means that only the small part of the cooling channel operates effectively and this part expands with time. Thus, the transient thermal analysis of the stated problem requires a simultaneous account of 3-D geometry of the cooled body and its thermal inertia together with the coolant transient behavior. To resolve this coupled problem the Fortran code has been developed. It comprises (1) a 1-D finite-difference model simulating a coolant motion inside the cooling channel and a heat exchange between the coolant and channel wall and (2) a 3-D finite-element model reflecting the complex geometry and material properties of the solid body. The developed code has been applied to perform the comprehensive thermal analysis of the VVTS cooldown process.

The ITER VVTS (consisting of the **inner** and **outer** massive panels) is aimed to protect the super-conducting magnet coils against the heat flux radiated from the ITER vacuum vessel. The radiated heat is intercepted by the VVTS cooling system comprising the long tubes welded to the thermal shield panels. The VVTS **outer panel** shown in Fig. 1 was chosen for the transient thermal analysis as a representative part of the entire VVTS.

The VVTS cooldown process (from the room temperature down to the operating temperature - 80 K) has several restrictions related to the process duration, to the cooldown rate and to the temperature non-uniformity in the VVTS structure.

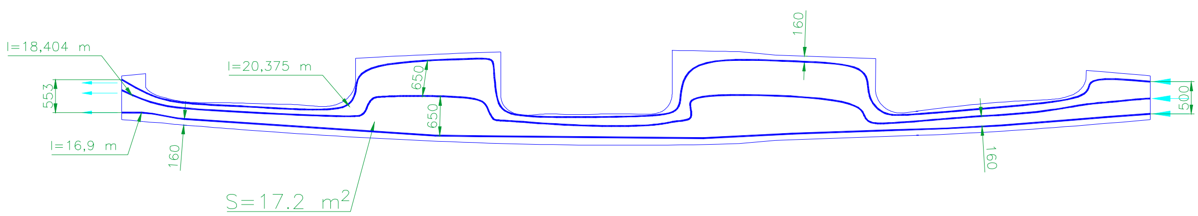

Fig. 1. Scheme of VVTS outer panel.

The obtained results accounted the cooldown process all peculiarities and allowed us to propose the cooldown scenario met the specified process restrictions.

J-29

T-15M TOKAMAK VACUUM VESSEL. MAIN RESULTS OF ENGINEERING DESIGN STAGE

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The Engineering Design Stage for the T-15M tokamak (upgrading of the existing T-15 tokamak) has been completed. T-15M is a reduced prototype (by a number of parameters) of the prospective ITER intended to solve a number of research problems. The vacuum vessel (VV) is to ensure formation, heating, confinement of the plasma column (major radius – 1.55 m, minor radius – 0.5 m, single-null divertor) and plasma diagnostics. The main characteristics of the VV are as follows: diameter – 4,95 m; height – 3,15 m; volume – 30 m^3; ultimate background pressure - 3×10^{-6} Pa; number of full-scale plasma discharges - 5×10^4; number of plasma current disruptions - 2×10^3 (at $B_T \le 2.5$ T, $I_p \le 1.7$ MA); gap between the plasma and VV first wall - 50 mm in the equatorial plane on the VV inboard and 150 mm on the VV outboard. The VV are characterized by a constant operating temperature of the shell amounting to 220°C and a large number of ports in all gaps between EMS windings (20 equatorial ports 360×950 mm in size for introduction of plasma heating power up to 20 MW and solution of the diagnostic problems), 20 inclined ports 320×420 mm at the VV top and bottom, vertical ports 360 mm in length - 20 ports at the top and 10 at the bottom of the vacuum vessel).

The VV comprises the all-welded VV shell with 4 field joints for assembly with EMS and installed inside the latter. The austenite stainless steel shell is single-walled with D-shaped cross-section, the wall is 4 mm thick on the inner cylindrical part (∅1900 mm) and 8 mm on the 20-faceted VV outboard shell. The VV comprises also 10 flexible VV supports; plasma facing graphite components; divertor; elements of the degassing warm-up system; set of passive stabilization coils; electromagnetic diagnostics components; movable limiter for operation with the round plasma and for formation of the plasma break-down boundary.

Stress analysis of vacuum vessel under design loads was performed using 3D finite element model of the whole vessel. Three cases of the vessel electromagnetic loading were analyzed: loading due to the eddy currents in the vessel during the central and vertical plasma disruption events, and loading due to the halo current during the slow downward vertical disruption event. Stress analysis of vacuum vessel has shown that stresses are within allowable limits for all analyzed design cases. The structure deadweight, atmospheric pressure and normal operating temperature of 220°C were taken into account. Nonlinear buckling analysis of the vacuum vessel has shown that the safety margin of the structure is more than 3. The first three natural frequencies of the vessel fall into the range from 28 to 60 Hz and correspond to the vessel movement as a whole structure on flexible supports. The evaluation of the dynamic amplification factor (DAF) for the most serious electromagnetic loads has shown that the DAF with respect to stress in supports is 0.35, and that the DAF with respect to stress in vessel walls is 0.9.

J-30

DESIGN AND ANALYSIS OF THE ITER VACUUM VESSEL AND PORTS

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The ITER vacuum vessel (VV) is a torus-shaped double-wall structure reinforced by poloidal and toroidal ribs. The VV main function is to provide the high-vacuum and primary safety confinement boundary. The vessel also supports the blanket and the divertor components. To provide access inside the vessel for plasma heating, diagnostics, vacuum pumping and other needs, the VV is equipped with upper, equatorial, and lower port structures.

Towards the construction phase, the VV design has been developed in more detail with the focus on simplified manufacture and reduced cost of the vessel sub-components. Options of the general scheme of the VV fabrication have been considered in cooperation with the industrial companies and the design has been updated in conformity with the main manufacture requirements/recommendations. To simplify the design, the inboard triangular supports of the blanket modules were eliminated and the module configuration was modified in this area. The outboard triangular supports are kept but their component design was also simplified.

For the port design, a single-wall construction will be used at a certain distance from the main vessel, where the neutron load is reduced. This approach will simplify the port manufacture and decrease the cost. For the upper and equatorial ports, the in-port space is occupied with an integrated subassembly - the port plug, which, apart from its functional purposes, provides the vacuum/pressure boundary for the in-vessel volume. Special attention was paid to the design of the supporting and sealing components between the plug and the port with the focus on simplified maintenance. For further cost reduction, the number of large lower ports is halved, with the ports between every other TF coils. Only pipe feedthroughs and local small penetrations will be used between the remaining TF coils.

The VV sectors are welded to each other on site with field weld. The pressure boundaries for these field joints were re-arranged to provide independent water flow from that of the VV sectors. This arrangement allows a simplification of the leak testing procedures during initial assembly and operation. The general principles of the main vessel cooling are unchanged but the port plugs will be cooled in series with the end portions of the ports – an approach, which simplifies the cooling pipe arrangement.

The VV must withstand the loads both directly affecting the vessel and transmitted from the in-vessel components. Additional analyses have been performed to demonstrate that the safety margins are sufficient. Details of the current VV design and the related analyses are reported in this paper.

J-31

DESIGN OF THE ITER VV SUPPORT

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The ITER vacuum vessel (VV) is a large torus that forms the first confinement barrier of the tokamak and provides the support for the in-vessel components. The VV supports must withstand a large number of loading conditions including electromagnetic, seismic, operational and upset conditions, and support the large VV and in-vessel component masses that are required to provide radiation shielding to the cryogenic superconducting coils. The supports have also to accommodate the thermal expansions (VV operating temperature is 100°C) and limit the VV displacements in operation and in fault conditions to allow small clearances between the VV and the surrounding structures and to simplify the design of the structures connected to the VV that penetrate though the cryostat (i.e. port bellows, in-vessel component cooling pipes, etc.).

The thermal shield and the cryogenic magnets with the reinforcing inter-coil structures surrounding the VV complicate the VV support design, as there is very little space available. Also the VV is located inside the cryostat in vacuum condition. This poses an additional difficulty to the design as friction is enhanced by vacuum. Furthermore the nuclear reactions from the plasma and the structure activation impose a limitation in the choice of the material based on the behaviour under radiation.

Several design options have been considered and analysed. The criteria that mainly influence the selection of the final design are the structural safety margin, the impact of each option on the design of the full ITER machine, the access for inspections, the possibility to repair the component in case of failure, and the impact on the machine and support assembly procedure. The design and analysis are performed following the requirements set by the ASME III code Subsection NF.

This paper describes the selected option of the ITER VV support design and the structural analyses performed to demonstrate the structural integrity under the worst load case conditions.

J-32

HIGH POWER Nd:YAG -LASER WELDING IN MANUFACTURING OF VACUUM VESSEL OF FUSION REACTOR

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Laser welding has shown many advantages over traditional welding methods in numerous applications for example in the field of transportation industry. The advantages are mainly based on very precise and powerful heat source of laser light which change the phenomena of welding process when comparing to traditional welding methods, for example GMAW. According to the phenomena of the laser welding, penetration is deeper and thus welding speed higher. Typical welding speeds, with few millimeters thickness of parts to be welded, are metres per minute. Because of the precise power source and high welding speed, the heat input to the workpiece is small and distortions are minimal respectively. Also the shape of laser weld is less critical for distortions than traditional welds.

In the case of welding thick sections usability of lasers is not so practical than with thin sheets, because with power levels of present Nd:YAG -lasers depth of penetration is limited up to about 10 mm by single pass welding. One way to overcome this limitation is to use multipass laser welding, in which narrow gap and filler wire is applied. By this process thick sections can be welded with smaller heat input and then smaller distortions and process seems to be very effective comparing "traditional" welding methods not only according to the narrower gap. Another way to increase penetration and fill the groove is using so called hybrid process, in which laser and GMAW are combined.

VTT Industrial Systems has been involved several in EU-ITER tasks in which suitable joining method has been considered for manufacturing of vacuum vessel of fusion reactor. Although the walls of vacuum vessel are made from 60 mm stainless steel, Nd:YAG -laser welding with some additions has shown great potential to be the joining method. One reason for using Nd:YAG -laser is the enormous size and weight of the vacuum vessel. So vessel will be constructed at the place of it's final placement. This together with the geometry of vessel set the positional welding requirements for the processes used and Nd:YAG -laser seems to be usable, because of the flexible transmit of laser ligth via optical fibre. In this paper results of several testruns made with Nd:YAG -laser are reported. Filler wire feeding as well as arc welding has been used as an addition with Nd:YAG -laser itself.

J-33

THE HT-7U SUPERCONDUCTING TOKAMAK VACUUM VESSEL DESIGN

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The vacuum vessel of the HT-7U superconducting tokamak will be fully welded structure with double wall. And space between double wall will be filled with borated water for neutron shielding. Non-circular cross-section of the vessel is used for plasma elongating. Sixteen horizontal ports, sixteen upper vertical ports and sixteen lower vertical ports are used for diagnosing, vacuum pumping, plasma heating and plasma current driving, etc. The vacuum vessel consists of 16 segments. It will be baked out at 250°C to get a cleaning wall. Low stiffness supports are employed to support vacuum vessel and to accommodate the thermal deformation. Bellows on ports is to avoid welding at the interaction of vessel and ports damaged. COSMOS2.0/M and ANSYS5.7 are accepted to analysis stress and thermal analysis. The stress caused by thermal deformation and electromagnetic (EM) is serious, but it is under allowable stress of material 316L stainless steel. R&D of the support and bellows had been done. The result of measured is good accord with analysis result. One section of the vacuum vessel fabricate is under going.

J-34

THERMAL HYDRAULIC ANALYSIS OF THE ITER VACUUM VESSEL COOLING SYSTEM

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The Vacuum Vessel (VV) of the ITER is a torus-shaped double wall structure with shielding and cooling water between the shells. The Vacuum Vessel operates under the high neutron loads, which are responsible for the vessel structure heating. To remove heat absorbed by the VV wall and shielding plates the water forced flow is used. The Vacuum Vessel cooling system consists of two independent cooling circuits serving alternate sectors. Computation of the VV temperature state and its cooling system parameters under the normal and off-normal machine operations are reported in this paper.

The possible impact on the VV cooling system capability the heat loading conditions together with the cooling system parameters has been studied. For this purpose (1) the theoretical global model of the VV sectors with design details including the rib layout and port structures has been created and (2) the thermal-hydraulic analyses has been done for normal and off-normal operations in order to assess the actual flow distribution inside the VV structure and the global temperature state of the VV with taking into account natural convection and non-uniform heat distribution in the VV.

The velocity range being studied for both normal and off-normal operations incorporates all flow regimes: laminar, transition-to-turbulent, developed turbulent. Heat load on the VV due to neutron heat flux from the plasma was taken into analysis with actual non-uniformity in poloidal and toroidal directions and as a decay curve in the radial distribution of volumetric heat in the vessel body. Based on the performed study the completing correlations for description of heat transfer at forced, natural and mixed convections have been recommended.

In this paper the developed VV transient global model is described and the results of thermal and hydraulic analyses for off-normal condition for one of two cooling loops, when other is affected by LOCA condition, are presented. The obtained results made possible to assess cooling system capacity to provide the required temperature regime for the VV shell during LOCA condition developing.

J-35

SPRING-BACK SIMULATION OF SHEET METAL FORMING FOR THE HT-7U VACUUM VESSEL

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The HT-7U superconducting tokomak, which is approved by Chinese government, is an advanced steady-state plasma physics experimental device to be built at the Institute of Plasma Physics, the Chinese Academy of Sciences (ASIPP). Its mission is to develop the scientific basis and technologic basis for the future Tokamak fusion reactors and to study physical issues on the substainment of a non-burning plasma scenario for the steady-state operation.

As one of the key components for the device, the vacuum vessel of the HT-7U is an all-metal welded double wall toroidal structure, which can provide ultra-high vacuum and cleanly location for the operation of plasma. The design base pressure is $1.3x10^{-5}$ Pa. 316L stainless steel was chosen as its material. The whole HT-7U vacuum vessel consists of 16 sectors. During final assembly, they are all field welded together to form toroid. The vessel outermost radius is 2.7m and height is 2.58m. But the thickness of all walls is only 8mm. The shell thickness is smaller compared with its circumflexion radius. So it has characteristics of ultra-high vacuum and thin shell. The cross-section of vessel is noncircular, but D shape, which has five different dimensions along its circumference. Each piece of arc wall also has two curvatures in the space of three dimensions. Now all kind of the R&D programs is in progress. The primary die and punch of HT-7U vacuum vessel has designed. During this process, the sheet metal panel undergoes plastic deformation and develops residual stresses. Some of problems will be found, such as localized buckling, undesirable local deformation at the head of panel and excessive spring back of the end of panel. In order to find optimal shapes of technical surface on the die and punch a kind of software DYNAFORM was used in the design and fabrication. In this paper a kind of dynamic simulation for sheet metal forming are presented. According to the simulation analysis results the optimum shape of the die and punch surface was finally determined.

J-36

LEAKAGE RATE OF COMBINED HEMISPHERICAL SHELLS WITH D-T NEUTRONS

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A experimental assembly made of the three hemispherical shells had been established. The geometrical sizes of beryllium, stainless steel and simulation material shells with densities of 1.82 g/cm^3, 7.88 g/cm^3 and 1.41 g/cm^3 are ϕ 11.4 cm - 19.4 cm, ϕ 19.4 cm - 22.0 cm and ϕ 28.0 cm - 32.0 cm, respectively. There is vacancy in the space between the stainless and simulation material shells. The chemical composition of the beryllium shell is 98.9% beryllium with a small amount of impurities, e.g., Fe, Al, Mn, Cr, Ni, and so on. The chemical elements of the stainless steel shell include Fe, Cr, Ni, Mn, Si, Ti, and so on. The their mass percentages are 69.1, 18.0, 9.0, 2.0, 1.0, and 0.7, respectively. The element compositions of C, H, N, O, F and Cl in the simulation material shell are 3.07, 6.77, 2.23, 1.48, 0.06, and 0.01, respectively.

The titanium-tritium target was installed in the D^+ beam drift tube made of aluminum, 16mm in diameter and 1mm in thickness. The detector of α-particles emerging in D-T reaction was installed in the angle of 178.6 degrees to D^+ beam direction. A ϕ 50.8 $\times$ 50.8 mm NE-213 spectrometer was used to measure leakage neutron spectrum in the range of 0.75–14MeV. The O5S Code are used to calculate the mono-energy neutron response function of the scintillator. The surrounding background was measured using a shadow technique. The overall experimental error was 6%–8%.

The assembly was vertically placed on a wood table. The top of the shell externality and the sphere center was in the line of D^+ beam. D-T neutrons were incident from the top of the shell externality. The NE-213 detector placed on D^+ beam direction was used to measure leakage neutron spectram. When the distance between the center of the titanium-tritium target and the top of the shell externality was chose as 50 cm, the leakage neutron spectram were measured at the four position of 0, 150, 200 and 250 cm from the front end of the detector to the shell plane. With the distance from the front end of the detector to the shell plane kept 200cm, the leakage neutron spectram varying with the distance between D-T target and the assembly were also measured. The leakage rates (1/neutron.cm^2) with the threshold energy of 1, 10, 12 and 14MeV were obtained from the measured spectram.

The leakage rates of the experimental assembly was also calculated using MCNP/4A Code with ENDF/B-V library data. The neutron angle flux distribution was considered. The neutron energy was supposed 14.5 MeV. The experimental results are 15%–20% larger than the calculated. To resolve the problem further experiment and calculation are necessary.

J-37

MECHANICAL CHARACTERIZATION OF EUROFER97 IRRADIATED (0.32 dpa, 300°C)

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The reduced-activation ferritic martensitic steel EUROFER97 (9CrWVTa) is presently considered for the Breeding Blankets Modules of DEMO and for the Test Blankets Modules developed in the EU. So far, the mechanical properties of the as-received material have been investigated by other european associations. In this work, the first results on neutron irradiated EUROFER97 are reported.

Mechanical specimens extracted from a forged bar (heat n° E83699) have been irradiated in the BR2 reactor at 300°C up to a fluence of 2.14 X 10^{20} n/cm² (E > 1 MeV), in the frame of the IRFUMA-I experiment. Tensile, impact and fracture toughness test results have been obtained from the irradiated samples, allowing to assess the consequences of neutron exposure on the mechanical strength (hardening) and toughness properties (embrittlement) of the material, after comparison with data available for the unirradiated condition.

The following has been observed:
irradiation produces hardening, which can be quantified in the range RT to 300°C as an increase of 20% for the yield strength and 35% ÷ 55% for the ultimate tensile strength;
ductility remains satisfactory, with no significant loss in uniform elongation or reduction of area and a slight degradation of total elongation (3% ÷ 4.5%);
the shift of the DBTT measured from standard Charpy tests is moderate (14 ÷ 16°C), while no statement can be made with respect to the variation of Upper Shelf Energy;
the shift of the reference temperature (T_o) measured from PCCv toughness tests is higher than for the Charpy DBTT (43°C), indicating that actual fracture toughness tests have to be performed if a reliable assessment has to be made of the influence of irradiation on the material's toughness.

In spite of the limited amount of relevant data available in the open literature, the effects of irradiation on EUROFER97 have been compared with those observed on other RAFM steels (F82H in particular), under comparable irradiation conditions. Irradiation hardening was found to be higher than for other RAFM steels (including F82H) in terms of mechanical strength increase, whereas ductility is practically unaffected. As far as irradiation embrittlement (shift of Charpy-based DBTT) is concerned, EUROFER97 appears to perform slightly better than the most recent RAFM steels (specifically, F82H) after 0.32 dpa neutron irradiation at 300°C.

J-38

MEASUREMENT AND ANALYSIS OF NEUTRON AND GAMMA-RAY FLUX SPECTRA IN SiC

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Silicon carbide composites are candidates for advanced structural materials in future fusion power plants, mainly due to their low decay heat and activation properties. Integral experiments are required to benchmark the nuclear data involved in design calculations, both with regard to the activation and the neutron-photon transport behaviour.

As part of the European Fusion Technology Programme, a benchmark experiment with SiC was carried out by a collaboration of ENEA Frascati, TU Dresden, FZ Karlsruhe and JSI Ljubljana. A block of sintered SiC (457 mm x 457 mm x 711 mm) was irradiated at the Frascati Neutron Generator with 14 MeV neutrons and several responses were measured inside the assembly. In this paper, the results of neutron and gamma-ray flux spectra measurements and their analysis are presented.

Neutron and gamma-ray flux spectra were measured simultaneously, using a NE 213 liquid-scintillation spectrometer at four positions (central axis of the block; 127, 279.4, 431.8 and 584.2 mm distance to the front plane) inside the SiC assembly.

The computational analysis of the spectra was performed with the Monte Carlo code MCNP-4C using a full 3D geometry model of the assembly, the neutron generator and the experimental hall. Nuclear data were taken from the Fusion Evaluated Nuclear Data Library FENDL/MC-2.0, except for ^{28}Si, for which the new data evaluation of the European Fusion File EFF-3.0 was used.

Measured and calculated flux spectra are compared and ratios of calculated-to-experimental values are derived for various energy ranges. The fast neutron flux in the thick assembly can be reproduced by the MCNP calculations within 20%. The gamma-ray flux for E > 0.4 MeV is underestimated by up to 24%. Differences between experimental and calculated spectra can be attributed to some resolved peaks or groups of peaks in the distributions.

J-39

TENSILE AND IMPACT PROPERTIES OF F82H STEEL APPLIED TO HIP JOINED FUSION BLANKET STRUCTURES

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In JAERI, a water-cooled solid breeder blanket has been selected as the primary goal for a DEMO blanket [1]. In the design, a low activation material like reduced activation ferritic/ martensitic steel, e.g. F82H is regarded as the first candidate structural material. The first wall of the blanket being developed by JAERI has a complex structure and is fabricated by using solid Hot Isostatic pressing (HIP) joining method for two plates and a stack of rectangular channels. A partial mock-up of such a blanket structure has been successfully fabricated in the previous study. However, it was revealed that grain coarsening was found at the corner of HIP-joined rectangular cooling channels through the metallurgical observation. Furthermore, a machining process and several heat treatment processes were applied to fabricate the mock-up structure. It is very important to verify that these treatments do not affect the mechanical properties of the first wall materials including the joined region. In this study, test pieces cut out from the fabricated mock-up were examined through tensile, charpy impact tests and hardness measurement. Tensile specimens were sampled from the HIP-joined cooling pipes (8 x 8 mm^2 in cross-section, 1.5-mm wall thickness) and from the plate in the first wall. These sub-sized specimens, which are in a plate-like-shape, have dimensions of 25.4 mm in total length and 0.76-mm-thick. The tests were performed in the temperature range from RT to 773 K with a strain rate of 5 x 10^{-4}/sec. Impact specimens were cut out in longitudinal direction from the plate with the dimensions of 20 mm in length and 3.3-mm-square. The tests were performed in the temperature range from 153 K to 287 K. Vickers hardness measurements were carried out for the tensile specimens with a testing load of 9.8 N. Hardness of specimen including the HIP interface region was measured in the direction perpendicular to the interface line. In addition, a region very close to the HIP interface was measured by using an Ultra Micro Indentation System with a testing load of 15 mN. Metallurgical observation in the coarsening region and the fracture observation in these specimens were conducted by using an optical microscope and a SEM, respectively. The microstructures including the HIP interface were observed by using a field emission TEM with 200 kV. The tensile tests for the HIP joint resulted in an increase of strength by about 50 MPa, and a decrease of the elongation by about 4% in comparison with the results of an IEA standard alloy, at all of the test temperature regions. It was observed that the HIP joint was fractured at the center of the specimen where the grain coarsening was located. Vickers hardness was almost constant (225 ~ 230 HV) in the coarsening region including the HIP interface, and the hardness at the HIP interface was 227 HV. The hardness in the coarsening region increased by about 5% in total in comparison with that of the standard alloy. Microstructural observation showed that relevant martensite lath structure was maintained in the coarsened prior-austenite grains. Therefore it is judged from the present study that the grain coarsening had no effect to the increase of the hardness and strength in direct, and also that the coarsening did not drastically affect the basic tensile property of the HIP joint. On the other hand, ductile-brittle transition temperature in the plate was increased by about 40 K though the tensile property was nearly equal to that of the HIP joint. It can be understood that this is the effect of the heat treatments in the fabrication process.

[1] M. Enoeda, et al. Design and Technology Development of Solid Breeder Blanket Cooled by Supercritical Water in Japan. To be published in Fusion Energy 2002.

J-40

DISPERSION STRENGTHENED EUROFER 97 STEEL FOR FUSION REACTORS

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Efforts are made world-wide to increase the high temperature (>550°C) properties of reduced activation ferritic martensitic (RA-FM) steels by dispersion strengthening. This kind of steels is aimed to become the supporting structure of future fusion reactors.

Dispersion strengthening, a method where small (typically 5–25 nm) particles are distributed into a matrix can be used to introduce such high temperature properties. One way to bring in the dispersoid into a matrix is by milling powder of the dispersoid and the matrix (mechanical alloying, MA) followed by extrusion and/or Hot Isostatic Pressing HIP. Materials produced by MA with yttrium oxide as a dispersoid have shown excellent properties, but it is very demanding and expensive to optimise such an unisotropic material in a complex component. Complex components can be produced by powder HIP with good economy to a near net shape (NNS). A condition of that is using atomised spherical powder. During MA the shape of the atomised powder particles change from spheres to flakes and the powder does not fit for use in the production of NNS components anymore.

Initial tests with an alternative route for production of dispersion strengthened (DS) RA-FM steel for NNS components by powder HIP were therefore undertaken. The route comprises conventional atomisation of a pre-alloyed melt to powder followed by heat treatment in a special atmosphere. The powder, still with spherical powder particles, will be used for production of NNS components by powder HIP. Initial experiments where a conventionally produced bar was re-melted and atomised showed that dispersoid particles of the aimed size (8–25 nm) were produced by this route in the RA-FM steel Eurofer 97. The dispersoid particles consist mainly of Ti (C, N). The small increase in the nitrogen (N) content required to form the particles is not wanted in a reduced activation material because C14 (5730 y) is produced via the N14(n,p)C14 reaction. However, as N is present in small amounts it is really a balance between the benefits of dispersion hardening, production costs and costs due to environmental effects. The production cost for components by this route will be low compared to that using MA. Results from the production of DS Eurofer 97 steel from virgin raw materials and results of the creep testing of the HIPed DS material are presented.

J-41

CORROSION STUDY OF AN AUSTENITIC STEEL IN Pb-17Li UNDER A MAGNETIC FIELD AND ROTATING FLOW

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The liquid alloy Pb-17Li has been proposed as a breeder material in the water-cooled liquid blanket for fusion reactors. During operation, structural materials such as austenitic or martensitic steels exposed to Pb-17Li will be subject to corrosion. Many parameters like temperature, thermal gradient or hydrodynamics may influence the corrosion rate of such materials in contact with flowing Pb-17Li. The strong magnetic fields expected in fusion reactors must also be considered. In fact, under a high magnetic field, a liquid alloy flow is modified and characterized by the presence of a core velocity in the central region and various boundary layers in the vicinity of the channel-walls which exhibit strong velocity gradient. It is therefore anticipated that corrosion and mass transfer processes may be modified in the presence of a magnetic field due to their sensitivity to the liquid hydrodynamics and especially the velocity gradient at the wall.

In order to study the effect of a magnetic field on the corrosion of steels in contact with liquid Pb-17Li, it has been built an experimental device to perform corrosion tests under and without a magnetic field in flow driven by a rotating disk in a cylindrical crucible. An analytical (numerical) model has been developed to predict the velocity distribution in the cylindrical crucible with the flow driven by a disk (which is the corrosion specimen) at the top of the crucible. The velocity profiles predicted numerically without and under a magnetic field were validated by experimental measurements carried out with mercury and water at room temperature instead of the liquid alloy Pb-17Li in a Plexiglas container equipped with an ultrasonic sensor. The calculations and the experimental results, which are in good agreement, are presented here.

Corrosion tests have then been performed in liquid Pb-17Li at 480°C up to 500 h without and under a magnetic field with a flow generated in a cylindrical crucible by an austenitic rotating disk and with the following hydrodynamic characteristics: $\Omega = 2.1$ rad.s^{-1}, $R_e = 21000$, $H_a = 0$ and 130. The test results are presented here. They highlight the sensitivity of the corrosion process to the hydrodynamic of the flow.

J-43

DEVELOPMENT OF IN-SITU CaO INSULATOR COATINGS ON VANADIUM ALLOYS

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The self-cooled lithium blanket concept with a vanadium alloy structure offers a potential for a high performance fusion system with attractive safety and environmental features. However, a key issue for this concept is the development of an electrically insulating coating on the coolant channel walls to mitigate the magneto-hydrodynamic pressure drop of the flowing lithium in a high magnetic field. Based on the diverse requirements for the coating, an in-situ formed CaO coating has been identified as a leading candidate for the insulator coating for the self-cooled vanadium/lithium system.

A systematic investigation has been undertaken to define the parameters for development of an in-situ-formed CaO coating with a self-healing capability on a vanadium alloy substrate exposed to lithium with controlled chemistry. The process involves interactions of oxygen from the vanadium alloy with calcium added to the lithium to form CaO as a reaction product at the solid-liquid interface. The effort involves analysis and experimental evaluation of the thermodynamics and kinetics of oxygen interactions in the vanadium alloy/lithium-calcium system as a function of temperature in the range 500–700^{o}C. For the current series of tests, oxygen is quantitatively precharged into the surface regions of the vanadium alloy by exposure to low-pressure oxygen gas with subsequent annealing at 750^{o}C to reduce any surface vanadium oxide to oxygen in solution in the alloy plus internal oxidation of the titanium. Various analyses, SEM/EDS, X-ray diffraction, microhardness, etc., have been used to determine the distribution of oxygen in the system. Exposure of the oxygenated vanadium alloy to a Li-2.8 at. % Ca alloy at 600 and 700^{o}C has resulted in the formation of reaction films ~10–30 microns in thickness. Analyses indicate that oxygen in the vanadium alloy is reduced and that the reaction film is primarily CaO. Quantitative results from this series of experiments will be presented.

J-44

CORROSION BEHAVIOR OF AlN FOR SELF-COOLED Li/V BLANKET APPLICATION

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A self-cooled lithium / Vanadium-alloy (V-Cr-Ti) blanket is one of the most attractive blanket concepts because of its reduced activation, high tritium breeding ratio, accommodation of high heat loads and so on. The primary issue associated with this blanket concept is potentially high MHD pressure drop induced by the liquid Li flowing through high magnetic fields. In order to reduce this pressure drop to acceptable levels, fabrication of thin insulator coating on an inner side of V alloy tubing has been considered. As one of coating material candidates, AlN had been explored, mainly because it thermodynamically has a chance to escape from severe chemical corrosion by liquid Li. However, only purified AlN bulk specimen has a possibility to realize the high chemical compatibility, because oxygen, which is always the main impurity in AlN, dissolve easily into Li causes fragile structures or low electrical resistivety of AlN. In this paper, we investigate corrosion mechanisms of AlN bulk specimen to contribute to a coolant system concept in the blanket from a viewpoint of chemical stability of the materials.

2 types of high purity AlN bulk samples, containing 0.9 wt% and 0.2 wt% of oxygen, were sintered in 5–20 cc of liquid lithium dissolved in Mo or V alloy pot at 673–1073 K for max. 1400 h. The samples with higher oxygen concentration reduced 0.2% of their weights and electrical resistiveties to $\sim 10^8$ ohm.m after the sintering tests at 773 K for 1400 h, while, below 873 K, no changes on both were observed after the tests in cases of those with lower oxygen concentration. This indicates, below 873 K, corrosion mechanism is dissolution of oxygen impurity into the Li. At 1073 K, a significant dependence of weight loss on the pot materials (~15% in cases of V alloy pot and less than 0.1 % in cases of Mo alloy pot) were observed, while no differences were observed below 873 K. Regardless of no direct contact between AlN and V alloy, Ti in the V alloy is an only element, which can absorb a large amount of N from AlN, by thermodynamic consideration. At 1073 K, the main corrosion mechanism is considered to be N dissolution into the V alloy through liquid Li. This means N diffusions in liquid Li and V alloy are high enough to transfer a large amount of N in spite of a very small equiliblium N concentration in the Li. At 973 K, dependences of the weight loss on both O concentration and pot material are considered to be caused by the effects of these two mechanisms.

One of the attractive points on Li / V blanket is high temperature operation. However, at Li / AlN / V blanket system at 973 K, AlN will be corroded severely as far as Li is contact with both AlN and V alloy. To solve this problem, all the inner sides of every V alloy tubing have to be covered by AlN coatings to avoid the contact of Li with V alloy. In this concept, coolant tubing outside the blanket should not be built by V alloy, or a large area of inside of heat exchanger has to be covered by the coatings.

J-45

IRRADIATION OF SiC_f/SiC COMPOSITES, FIBERS AND MATRICES

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This paper reports on the irradiation of SiC_f/SiC materials as a contribution to the EFDA Technology Programme on advanced materials. SiC_f/SiC composites are envisaged as candidate structural materials for future fusion reactors because of the low activation of SiC, the good corrosion resistance and the excellent high temperature performance of those composites. However, the impact of neutron irradiation on the mechanical behaviour, swelling and thermal conductivity of the SiC_f/SiC composites at a high temperature is to be assessed, and the application window has to be determined.

A high temperature irradiation of SiC materials in the High Flux Reactor in Petten, denoted as 'SICCROWD' is a 3 dpa target dose irradiation and has two temperature levels of 600–650ºC and 900–950ºC respectively. The SiC_f/SiC materials included in the irradiation are supplied by the EU partners. Further materials are included in the frame of an IEA collaborative effort, by Japanese and USA partners. CVD monolithic SiC serves as a reference material and should enable close comparison with other, e.g. complementary, irradiation data. The specimen types are bars, for four-point bend tests, and discs for flash diffusivity measurements. Both types will be used to measure the dimensional stability of the SiC_f/SiC composites.

Some specimens have been mounted to allow for in-situ determination of thermal conductivity degradation by neutron irradiation. The first results of these in-situ measurements are presented and discussed.

The irradiation matrix comprises a variety of fibers and matrix materials, as well as composites with 1-D, 2-D or 3-D architectures. The pre-irradiation properties like thermal diffusivity and flexural modulus will be presented, demonstrating that the irradiation results will be relevant for further development of SiC_f/SiC composites.

J-46

ELECTROMAGNETIC STUDY OF THE ITER THERMAL SHIELD

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The ITER thermal shield is designed to set upper limits for both total and local thermal loads applied to the surface of components operated at 4.5K, by convection, conduction and thermal radiation. The thermal shield serves as an optically opaque reflector between all warm (room temperature and above) surfaces emitting thermal radiation to the liquid helium temperature magnet structures. The overall thermal shield system consists of four sub-systems, corresponding to the parts of the warm components, which they shield. These four systems are the vacuum vessel thermal shield (VVTS), the cryostat thermal shield (CTS), the transition thermal shield (TTS) and the support thermal shield (STS).

The electromagnetic loads are some of the most important loads for the thermal shield design. For their estimation, a detailed electromagnetic analysis of the thermal shield has been carried out. 3-D shell models of the thermal shield sub-systems have been developed. Depending on destination, the thermal shield has been simulated with a 3-D single-layer shell for the VVTS, and a set of 3-D single-layer plates, with reinforcement ribs for the CTS/TTS/STS. Two different transient electromagnetic processes were modeled separately and superimposed afterward in automatic mode: the first process caused by a variation of the toroidal plasma current, shape and position, and another the second one is due to the Halo currents and a variation of the toroidal magnetic flux created by the plasma. To assess the reliability of the thermal shield at different operating conditions dependent on different scenarios of plasma behavior such as the fast Central Disruption 27 ms, fast/slow upward/downward Vertical Displacement Event and Toroidal Field Coil Fast Discharge, the most critical condition from a standpoint of EM loads, was determined. The calculations have been carried out using the TYPHOON code, developed at the Efremov Institute. The TYPHOON code solves problems related to quasy-stationary transient electromagnetic processes in thin multi-connected conducting shells, arbitrary located in 3-D space, using electrical and magnetic vector potentials and it is suitable for processes with a skin time rather small compared with the characteristic time of external field variations. The TYPHOON code is based on the method of finite elements to carry out the 3-D problem discretization. As a result of the numerical simulation of transient electromagnetic processes in the shells described above, the distributed and total EM loads, caused by different external sources (either plasma toroidal current or Halo current and toroidal magnetic flux) and AC losses energy and power, dissipated in different parts of the thermal shield were calculated. The data obtained were used as initial data in subsequent thermal-hydraulic, static and dynamic and structural analyses of the ITER machine thermal shield. They influence not only the design of thermal shield components, but also the choice between different thermal shield design options.

J-47

THE SSC RIAR HIGH-FLUX RESEARCH REACTORS: EXPERIENCE AND POSSIBILITIES OF TESTING OF MATERIALS AND MOCK-UPS FOR FUSION

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One of the most important problems of the development of a fusion reactor is validation of radiation stability of its structural materials and high energy loaded structure elements of high heating rate. Operational parameters of these units (reception divertor plate, first wall of the vacuum chamber, blanket, etc.) are determined by a complex effect of powerful fields of plasma radiation, high temperature, cyclic mechanical loads, thermal changes, etc.

A considerable part of the above parameters can be reproduced in fission reactors, namely in the channels of high-flux research reactors with a neutron flux density of $\sim 10^{18} \div 10^{19} s^{-1} m^{-2}$ ($E \geq 0,1 MeV$) and γ-quanta $\sim 10^{19} s^{-1} m^{-2}$ ($E \geq 0,01 MeV$).

There are six research reactors operated at the SSC RF RIAR:
- three pool-type reactors RBT of different power;
- loop-type water-cooled reactor MIR;
- fast sodium reactor BOR -60;
- high-flux reactor SM with an intermediate neutron spectrum.

For investigations of materials and units of fusion reactors the reactors SM and BOR-60 are of the most interest, since their channels provide sufficient spectrum hardness and neutron flux intensity. Besides, a set of irradiation cells available in SM allows, in a wide range, the variation of the temperature conditions as well as accumulation rate of helium and other transmutants in the materials under irradiation. This makes it possible to study the effect of damaging factors individually or as their combination.

At the SSC RF RIAR essential operational experience was gained on the performance of in-pile experiments for candidate materials of fusion reactors such as copper alloys, graphite, refractory metals and beryllium. Data were obtained on radiation damage of these materials, their soldered and welded joints and also on serviceability of the reception divertor plate mock-up under cyclic load conditions. A brief overview of investigation methods is presented.

J-48

INTEGRAL BENCHMARK EXPERIMENTS WITH 14-MeV NEUTRONS FOR TESTING THE NUCLEAR DATA OF VANADIUM

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The perspective of using the vanadium in fusion reactor leads to high requirements to accuracy of its nuclear data, which, in turn, is followed by the necessity of testing the corresponding files of recommended evaluated data library (FENDL-2) in integral experiments. The experimental data on the 14-MeV neutrons interaction with the vanadium samples with thickness up to 25 cm were investigated (IPPE-FZK, JAERI), though there is a need in experiments as with higher sample thickness as at lower neutron energies.

To meet these demands, the ISTC project #910 "Execution of the complex of benchmark-experiments for testing the nuclear data of vanadium – main component of low-activation structural materials for advance nuclear energy" was proposed by the Russian Research Centre “Kurchatov Institute” (RRC “KI”), Moscow as a Contracting Institute with the two Participating Institutes: the Russian Federal Nuclear Center VNII of Experimental Physics (VNIIEF), Sarov and the State Scientific Centre of the RF – Institute of Physics and Power Engineering (IPPE), Obninsk.

In the project a set of benchmark-experiments was performed with a 14-MeV neutron source with vanadium spherical samples of different diameters (10, 24 and 34 cm). The largest spherical sample from vanadium of 34 cm in diameter was manufactured in the project, in addition to the existing vanadium samples used in previous joint IPPE-FZK experiments.

The neutron leakage spectra from spheres were measured by three methods: proportional gas counter and TOF technique with stilben and polystyrene detectors. A total neutron leakage multiplication in spheres was measured with the Total Absorption Method (in a boron tank) with an uncertainty on a level of 3–4%. The technique of gamma leakage measurements is based on the use of a single-crystal scintillation spectrometer with a large crystal NaI(Tl) and the TOF method.

Neutronics analysis of the experiments is performed with available neutron data files of vanadium, as well as with the new evaluated data files for vanadium prepared in frame of the project activities. A comparison of experimental and calculated results for the three vanadium spheres is given.

J-49

CURRENT STATUS OF THE PEBBLE BED ASSEMBLIES IN-PILE EXPERIMENT FOR THE HELIUM COOLED PEBBLE BED CONCEPT

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In the framework of the European blanket programme an in-pile experiment is developed for the Helium Cooled Pebble Bed concept. This experiment is designed to test the thermo-mechanic behaviour of lithium ceramic pebble beds under neutron irradiation in the High Flux Reactor in Petten. This test accommodates four separate pebble bed assemblies. A pebble bed assembly is made of a Eurofer steel cylinder containing a horizontal bed of ceramic breeder pebbles sandwiched between two horizontal beryllium beds. Floating Eurofer steel plates separate the pebble beds. A thermal barrier at the breeder bed circumference forces the heat flow through the beryllium beds and provide a radial temperature distribution in the ceramic breeder pebble-bed as flat as reasonably possible. A second stainless steel cylinder surrounds the pebble-bed assembly, which leaves a gap between both cylinders for temperature control by gas mixture technique.

Detailed nuclear and thermo mechanical analyses and calculations have been completed, which resulted in a definitive course of the manufacturing of the pebble bed assemblies and a plan for in-pile operation. All four Eurofer cylinders are filled with the ceramic and beryllium pebble beds, then compressed to 3 MPa at ambient temperature and 300–350^{o}C. After that the Eurofer cylinders were seal welded. The gas gap dimensions were defined by rigorous thermo-mechanical calculations, which included also the experimentally defined creep laws of the ceramic and beryllium pebble beds. The calculated gap dimensions are manufactured in the second stainless steel cylinder with a 5-µm tolerance to ensure the in-pile temperatures to be well defined.

The in-pile temperature field is to be measured by 18 thermocouples in each pebble bed assembly, which yields information about (changed) thermal and thermo-mechanical properties of the pebble beds. Also neutron radiographs are to be taken in between the irradiation in the HFR for mechanical analysis of the pebbles beds like swelling and creep compaction. The Pebble bed assemblies are scheduled for in-pile operation by the end of June 2002 and will stay in the reactor for about twelve months.

This paper describes the current state of the manufactured and assembled Pebble bed assembly in-pile experiment and gives a summary of all experimental inputs used for nuclear and thermo- mechanical calculations and design. Also the first results of irradiation performance may be presented. The first hours of in-pile operation are to a large extent determining the achievement of the key objectives.

J-50

THERMOMECHANICAL BEHAVIOUR OF CERAMIC BREEDER PEBBLE STACKS FOR HICU

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As a part of the European programme for the development of the Helium Cooled Pebble Bed blanket concept a high fluence irradiation, HICU, in the High Flux Reactor is under development. The HICU project concerns the investigation of the impact of neutron spectrum and the influence of constraint conditions on the thermo-mechanical behavior of ceramic breeder pebble-beds. The prelimary designs of the pebble-stacks shows that they are too small to apply the presently available tools for thermo-mechanical analysis of pebble-beds. In particular the ratio of pebble size to stack diameter is large, and the larger pebble-wall contact areas increase wall friction. This prevents a straightforward analysis of stresses during irradiation, and the determination of pebble swelling and the effects on heat transfer. This requires pre-testing of specimens representing key features of the irradiation specimens.

This paper reports on thermomechanical behaviour of miniaturised pebble-beds or so-called pebble stacks as a pre-irradiation test. Nimonic steel tubes with an innerdiameter of 4, 8 and 12 mm are used for uniaxial compression tests of the pebble stacks.

The tests are performed at room temperature and at 800°C i.e. the maximum irradiation temperature. The results of the tests show the thermal creep of the breeder pebbles at high temperatures. Also the influence of the stack-size because of the friction between the pebbles and the steel tube is studied. The findings are compared with the thermo-mechanical data obtained from Uniaxial Compression Tests, that apply for larger pebble-beds. The results are applied to the detailed design of the constrained pebble-beds in the high fluence irradiation.
Also the compatibility between the ceramic breeder materials and the containment is studied. It has been shown that platinum has the best compatibility at 800°C for both the lithiumtitanate and orthosilicate ceramics. Platinum foil is used as inner cladding for the pre-tests as well as the irradiation specimens.

The results of the pre-tests for the design of the high dose irradiation in the HRF will be presented. The design of irradiation specimens for HICU will be presented.

J-51

MHD SIMULATION OF LIQUID METAL IN DUAL-COOLED WASTE TRANSMUTATION BLANKET FOR FDS

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The liquid metal (LM) He-cooled waste transmutation blanket is one of promising candidate blanket for the excellent potential of disposing High Level Waste (HLW) in a Fusion-Driven Sub-critical System (FDS). The outer blanket consists of MA-Zone for transmutation of the minor actinides (MA: 237Np, 241Am, 243Am, 244Cm), U-Zones for breeding of 239Pu, FP-Zones for transmutation (FP: 129I, 99Tc, 135Cs), graphite neutron reflector and steel structure. The fuel (MA or U) are manufactured into particles with diameter ~0.5mm and LM Pb-17Li flows carrying the particles to complete a close cycle.

MHD simulation in LM zone offers the reference for the design of the blanket module. Using general CFD code of PHOENICS, the LM flow in MA-zone and U-zone is simulated, at the same time some proximate postulation is done, because LM contain a quantity of tiny particles. In MA-zone of high power density and the, high Hartman number influence on side layer thermal transfer has been analyzed. In U-zone, velocity profile and temperature field have been simulated in U-turn bottom of stagnant influence. For MHD pressure drop analytical model, because of the difficulties associated with modelling the complex blanket configuration, simplified assumptions have been made in considering the pressure drop of LM with suspending particles. An order of magnitude determination, compared with the packed beds pressure drop and that of LM uninvolved heavy fuel, has been made by starting from basic principles using the momentum equation. Meanwhile, the pressure drop of liquid-solid two phases in the LM with nuclear fuel particles slurry has been cautiously considered in magnetic field.

The LM MHD simulation results in MA-zone and U-zone are offered by PHOENICS, and the interface wall temperature of the U-turn or LM exit duct can be ensured to be less than 400• through analysis of thermal transfer. The preliminary MHD pressure drop analytical model of LM with suspending nuclear fuel particles in FDS blanket has been presented. Valuable reference is provided to improve the design of the blanket module.

J-52

SENSITIVITY AND UNCERTAINTY ANALYSES OF 14 MeV NEUTRON BENCHMARK EXPERIMENT ON SILICON CARBIDE

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Silicon carbide composite is a candidate structural material for advanced fusion power plants due to its good activation and thermal mechanical properties. The shielding performance of silicon carbide is, however, low necessitating additional shielding to sufficiently protect lifetime components such as the vacuum vessel and the super-conducting field coils. A qualified assessment of the shielding performance including uncertainty assessments is required to ensure reliable shielding calculations for SiC/SiC based fusion power plants. This can be achieved through benchmark experiments and their computational analyses. A suitable 14 MeV neutron experiment on silicon carbide has been recently conducted at the Frascati Neutron Generator (FNG) in a collaboration of ENEA Frascati, TU Dresden, FZ Karlsruhe and JSI Ljubljana with the objective to provide the experimental data base for the required benchmark analyses. This work addresses the computational analyses of the experiment including measurements of neutron flux spectra and reaction rates at different locations in the SiC assembly block. The analyses comprise transport, sensitivity and uncertainty calculations using both deterministic and probabilistic computational methods.

In the first stage of the experiment the reaction rates including Nb-93(n,2n), Al-27(n,α), Ni-58(n,p), and Au-197(n,γ) were measured by ENEA Frascati at various positions in the assembly. The computational analyses were performed with the discrete ordinates codes DORT and TWODANT as well as the Monte Carlo code MCNP-4C using cross sections from the European Fusion File (EFF-2.4 and -3.0) and the Fusion Evaluated Nuclear Data Library FENDL-2. The related sensitivity and uncertainty analyses were based on the deterministic approach using the SUSD3D code and showed that the key nuclear cross sections for the measured reaction rates were the elastic, inelastic, (n,p), (n,α), (n,np) and (n,nα) reactions on Si and C. The second stage of the experiment was devoted to measurements of the neutron and photon flux spectra by TU Dresden. The computational transport, sensitivity and uncertainty analyses were this time based on the Monte Carlo technique. Calculations were performed with the MCNP4C code and nuclear cross-section data from the EFF-2.4, -3 and FENDL-1, -2 data files. The Monte Carlo sensitivity calculations were performed with a local extension to the MCNP4C code that enables the calculation of point detector sensitivities. The benchmark analyses using EFF-3 data revealed a general trend for slightly overestimating the neutron flux across the silicon carbide assembly. With FENDL data, on the other hand, the fast neutron flux is underestimated (grossly by FENDL-1, up to 20 % by FENDL-2). The uncertainty estimations based on EFF-3 cross-section covariance data are in the range between ~3 and 12%. They are in general greater than the experimental uncertainties of typically 2 to 5% and are by ~30–50% lower than uncertainty estimations based on ENDF/B-VI co-variance data.

J-53

REDUCTION EFFECT ON TiO_2 CONTENT BETWEEN Li_2TiO_3 AND HYDROGEN IN SWEPT GAS

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Recently, lithium titanate (Li_2TiO_3) has attracted the attention of many researchers. As the shape of Li_2TiO_3, a small pebble was selected for both Japanese and European Blankets based on ceramic tritium breeding materials. In this work the TiO_2 doping of Li_2TiO_3 was examined about its effects on (i) storage in air, and (ii) reduction under flowing Ar + H_2 (0.1%) sweep gas simulating the "reference" purge for the actual fusion blanket designs. Various TiO_2 doped and undoped Li_2TiO_3 pellets (8 mm diameter by 2 mm thickness) were developed with near the same microstructure (density, grain size in the region of 0.2–2 μm, porosity) and high purity (99.99%) by Soekawa Chemical Co. Ltd. The Li_2TiO_3 and 5–30wt%TiO_2-doped Li_2TiO_3 pellets basic properties were examined by crystal measurement and structure observation. The weight loss and specific BET-Surface Area (SA) of each pellet were evaluated after heating ramps from R.T. to 800°C (at the rate β = 10 K/min, test run A), this last temperature being held constant for 1 hour under Ar gas flowing. The pellets reduction rates were evaluated by the ramp-annealing tests (test run B) performed under Ar+H_2 (0.1%) as before but at a slower heating rate (β = 5 K/min). Main results are reassumed in the following table:

Pellet type	Test run A: H_2O+CO_2 release rate peak temperatures (°C)	Tot.weight loss (%)	BET SA (m^2/g)	Test run B: Red. rate peak temp. (°C)	H_2 reacted (μmol/g)
Li_2TiO_3	230, 450, 670	~ 2.5	0.85	700 (broad)	11
TiO_2-doped- Li_2TiO_3	230, 450	0.5–0.8	0.76–81	higher than 800	9.2

The TiO_2 doping (from 5 to 30%) was found to improve at near the same level the Li_2TiO_3 pellets resistance to the environmental attack for very long exposure times in air at r.t. Grain size (from SA values) was confirmed to be near the same for doped and non-doped pellets (1.4±2 μm by SEM observation). The reduction rate of these Li-titanates as exposed to Ar + H_2(0.1%) purge gas was found to decrease with TiO_2 doping. X-ray diffraction pattern demonstrated the formation of $Li_4Ti_5O_{12}$ phase in the 10–30wt%TiO_2-doped Li_2TiO_3 which were sintered at 960°C x (1–4) hours.

J-54

STUDY OF DENSITY IMPROVEMENT FOR Li_2TiO_3 PEBBLES FABRICATION BY DIRECT WET PROCESS

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The application of Li_2TiO_3 pebbles (diameter: 0.2–2 mm) was proposed in the Japanese and European blanket design of a fusion reactor. The wet process and direct sol-gel methods are most advantageous from the viewpoint of low cost mass fabrication, reprocessing lithium-bearing solution, etc. Preliminary fabrication test of Li_2TiO_3 pebbles by the direct wet process with solvent exchange was carried out in a previous paper. However, density and grain size of the pebbles were less than 50%T.D. and 5μm, respectively. The targets of those are 80–85%T.D, less than 5 μm, respectively. Therefore, the improvement tests of Li_2TiO_3 pebbles by the direct wet process were examined about its effects on (i) dissolving process of Li_2TiO_3, (ii) degassing process from the solution, (iii) concentration and preparing processes of the solution and (iv) drying and sintering processes of the droplet for increasing density of Li_2TiO_3 pebbles in this study.

The improvement tests were performed to survey the parameter of each process and the results were as follows:

1) Heating condition was critical for the solution of Li_2TiO_3. Li_2TiO_3 powder was perfectly dissolved in 30%-H_2O_2 without any additives that occurred gas by heating at 60˚C with stirring.
2) The concentration of the Li_2TiO_3 solution was effective on the density. In these survey, the solution with more than 20wt% Li_2TiO_3 content showed the better density more than 80%T.D after sintering.
3) Degassing by heating at the temperature of 90–100˚C was effective to eliminate gas (oxygen or hydrogen peroxide) dissolved in the Li_2TiO_3 solution.

Based on these results, trial fabrication was performed with best condition. Obtained pebbles were characterized by scanning electron microscope (SEM), X-ray diffractometer (XRD), etc. The surface had no crack and the density was 85%T.D. Only Li_2TiO_3 was identified as a component, and the grain size was less than 5μm.

It was clear that the density was improved by the modification of the solution procedure, concentration of Li_2TiO_3 in the solution and degassing. In future plan, the mass fabrication of Li_2TiO_3 pebbles with controlled particle size and high sphericity will be developed to establish the direct wet process as mass fabrication process.

J-55

RECENT IMPROVEMENTS IN THE MODELLING OF IRRADIATED BERYLLIUM: THE ANFIBE CODE VERSION 1

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Beryllium is foreseen as a neutron multiplier in the Helium Cooled Pebble Bed tritium breeding blanket for fusion reactors. One of the key issues of the concept is the behaviour of beryllium under irradiation, since large quantities of helium and sensible amounts of tritium are generated (about 32000 appm ^{4}He and 150 appm ^{3}H peak production at the End Of Life of a Fusion Power Reactor). Swelling due to helium bubbles and tritium retention have to be predicted in the whole range of operating and accidental conditions of the HCPB blanket. The code ANFIBE (ANalysis of Fusion Irradiated BEryllium) is being developed to meet this need. The main issue in the development and validation of the code is related to the total lack of experimental data in the range of neutron fluences and temperatures typical of the EOL, as a consequence a far extrapolation of the models outside their validation range is necessary. Since the effective gas diffusivity depends on the microstructure of the sample, most part of the available experimental data are not directly relevant because the kind of material is significantly different from the one that is presently considered as a reference for the HCPB blanket, i.e. 1 mm pebbles, produced by Rotating Electrode Method. The need for extrapolation imposes a general, detailed description of the gas kinetics, as well as a particularly careful validation procedure of all sub-models of the code.

This paper presents the recent developments of the modelling of irradiated beryllium, which have been implemented in the version 1 of the ANFIBE code, with the aim to increase the confidence in its extrapolation. In comparison to the previous version:

1. Helium and tritium atomic diffusion coefficients in irradiated beryllium pebbles from the BERYLLIUM irradiation experiment (2 mm diameter, 480 appm ^{4}He, 780 K irradiation temperature) have been measured, on the basis of gas release experiments and Transmission Electron Microscopy observations of bubble precipitation. The measured values have been implemented in the code.
2. A new validation procedure, the so-called *microscopic validation*, based on a microstructure characterisation of irradiated beryllium samples, has been defined: this consists in studying, by optical microscopy and TEM, the size and distribution of gas bubbles and in comparing them with the predictions of the code.
3. The microscopic validation procedure has been applied to the samples from the BERYLLIUM irradiation experiment, at different temperature conditions. The experimental and theoretical analysis coves features evolving in a size range of five orders of magnitude.

J-56

AN ADVANCED THERMAL-MECHANICAL MODEL FOR FUSION PEBBLE BEDS

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Within the framework of the European Fusion Technology Programme, the use of Lithium ceramics and Beryllium in form of packed pebble beds as, respectively, neutron multiplier and tritium breeder, is foreseen for the Helium Cooled Pebble Bed (HCPB) breeding blanket of the fusion power reactor operating with a D-T plasma.

The packed pebble beds are granular systems composed of quasi-spherical pebbles having thermal and mechanical behaviours quite complex and strictly interconnected, whose knowledge is a pivotal key in the design and analysis of the HCPB breeding blanket modules.

The Nuclear Engineering Department of the University of Palermo is, at the present, involved in the development and validation of constitutive models suitable to realistically describe the thermal-mechanical behaviour of the fusion relevant pebble beds and apt to be easily implemented in the most quoted finite element codes.

This paper is focused on the investigation of the potential use of an advanced constitutive model for the description of the Lithium Orthosilicate and Beryllium pebble bed thermal-mechanical behaviour in blanket-typical configurations.

The proposed model is based on the assumption that the pebble bed can be considered as an homogeneous and isotropic medium. The bed thermal behaviour is described by a thermal conductivity depending on both the bed temperature and mechanical volumetric strain. The mechanical behaviour is described by adopting a Porous Elasticity model, with a pressure depending logarithmic bulk modulus, and a modified Drucker Prager model, suitable to describe, respectively, the bed non-linear elastic behaviour and its irreversible consolidation process. Moreover, a simple thermal creep model has been taken into account.

The model has been used to numerically simulate the SCATOLA benchmark. A detailed 2D axisymmetric finite element model of the SCATOLA mock-up has been set-up, taking into account a thermal-mechanical contact model for the interface between the bed and the steel-containing wall.

Coupled thermal-mechanical analyses have been performed by means of ABAQUS code to numerically reproduce the SCATOLA tests and the obtained results seem to agree quite well with the experimental ones, encouraging to a further development of the proposed model.

J-57

CHARACTERISATION OF DEUTERIUM TRANSPORT IN THE FIBRES AND MATRIX OF A 3D-SiC_f/SiC COMPOSITE

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A non-stationary heterogeneous transport model has been used to simulate a series of deuterium absorption-desorption tests carried out with two sets of 3D-SiC_f/SiC composite specimens over a temperature range of 675 to 1029 K and driving pressures ranging from 13 to 101 kPa.

The deuterium transport parameters corresponding to each phase of the composite have been derived showing the following Arrhenius tendencies for the diffusivity and solubility of deuterium in the matrix : $D_m\,(m^2s^{-1}) = 9.032\cdot10^{-8}\exp(-39.0(kJ/mol)/RT)$, $S_m(mol\ m^{-3}) = 1.857\cdot10^5\exp(-70.7(kJ/mol)/RT)$, and the fibre : $D_f\,(m^2s^{-1}) = 1.636\cdot10^{-11}\exp(-91.0\,(kJ/mol)/RT)$, $S_f\,(mol\ m^{-3}) = 2.128\cdot10^1\exp(-28.5(kJ/mol)/RT)$. The temperature dependency of D_m leads to much higher values than the corresponding ones reported for monolithic SiC in the same experimental range (more than 5 orders of magnitude); the accelerated migration through the extensive porosity of the matrix being a suitable reason for such a dissimilar behaviour. Because this phenomenon is less probable in a compact fibre, D_f approximates the values of monolithic SiC. The activation energy of diffusion through fibres D_f reported here, 91.0 kJ/mol, is the lowest in comparison with the range of values for SiC but still much higher than the reference values for metals and alloys. Regarding solubility, it is worth noting that the fibres are an exothermic absorbent according to the general behaviour detected in different types of SiC with a similar activation energy for dissolution, -28.5 kJ/mol. Conversely, the matrix does not show such behavior, indicating a different nature of the dissolution mechanism.

Two different manufacturing procedures were followed in the densification of the matrix of the two types of specimens. The different CVI/PIP time distribution in the preparation of the specimens has not provoked a noticeable variation of the deuterium transport characteristics of the matrix or the fibres and, consequently, a unique set of Arrhenius tendencies has been obtained for the two types of specimens.

The Sieverts' law has been checked for the deuterium solubilities obtained for both phases. From the heterogeneous analysis of several isothermal (871 K) absorption – desorption tests it has been observed that the solubility in the matrix shows a clear departure from the 0.5 power relationship with the loading pressure whereas the fibre phase approaches to the foreseen trend. Not having observed any influence of surface effects on the gas transport, other interaction mechanisms have been proposed as possible sources of the deviation.

J-58

MECHANICAL PROPERTIES OF HIPPED RAFM ODS STEELS FOR FUSION APPLICATION

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The efficiency of future fusion power plants is strongly dependent on the operating temperature. Conventional reduced activation ferritic martensitic (RAFM) steels presently considered for structural applications limit the operating temperature to around 550°C. Oxide dispersion strengthened (ODS) materials would allow an increase in the operating temperature of about 100 K. Reduced activation ODS steels are not available commercially at present. As a first step in developing a RAFM ODS Steel an existing 9%CrWVTa-RAFM steel, called EUROFER 97 was chosen as base material for the production of two variants with different Y_2O_3 contents (0.3 and 0.5 wt%) that have been produced by experienced industrial manufacturers. The production process included inert gas atomisation of EUROFER (H. C. Starck) and subsequent mechanical alloying in industrial ball mills by Plansee. Hot isostatic pressing (HIP) was chosen as the appropriate consolidation process for the production of bars with 60 mm in diameter and 300 mm in length. The hipping process is regarded as the most promising production route for near net-shape structures for future fusion reactors.

Miniaturised specimens with 2mm diameter and 7.6 mm gauge length were fabricated from the as-received bars and subjected to tensile tests in the temperature range between RT and 750°C. Yield strength and ultimate tensile strength of the ODS-EUROFER are raised by 50% and more, compared to the non-ODS RAFM steels like EUROFER 97 and F82H mod. This gain in strength still persisted at elevated temperatures. The uniform elongation of the ODS material is superior to that of common RAFM steels, whereas the total elongation above 400°C is lower, what is not unusual for this type of alloys.

Impact bending tests performed on sub-size Charpy specimens of KLST type showed that the ductile-to-brittle transition temperature (DBTT) is shifted from -100°C for forged non-ODS EUROFER to values between 70 and 100°C for ODS-EUROFER. The upper shelf energy (USE) i.e. the maximum absorbed energy of the ODS alloys is reduced by approximately 40% compared to the values of EUROFER 97.

Creep tests in the temperature range from 600 to 700°C up to test times of 10,000 hours confirm the superior creep behaviour of ODS-EUROFER. It exhibits a similar creep and creep strength as RAFM steels but at temperatures about 100°C higher. At 650°C and an applied stress of 100 MPa the extrapolation gives a rupture time of about 50,000 hours for the ODS steel (0.5% Yttria). Compared with EUROFER under the same test conditions the time to rupture has increased by more than two orders of magnitude.

Isothermal strain controlled low-cycle fatigue tests ($\Delta\varepsilon_t$=0.5–1.2%,T=250, 550, 650°C) on the same type of specimens as used for the tensile tests show the good LCF-behaviour of ODS-EUROFER. Compared to RAFM steels lower cyclic softening, a lower plastic deformation, substantially higher stress amplitudes at given strain and a higher lifetime can be observed.

J-59

ON THE NUCLEAR RESPONSE OF THE WATER-COOLED Pb-17Li TEST BLANKET MODULE FOR ITER-FEAT

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Within the European Fusion Technology Programme, the Water-Cooled Lithium Lead (WCLL) DEMO breeding blanket line was selected in 1995 as one of the two EU lines to be developed in the next decade, in particular with the aim of manufacturing a Test Blanket Module (TBM) to be implemented in ITER. This specific goal has been maintained also in ITER-FEAT programme even if the general design parameters of the TBM have reported some changes.

The Department of Nuclear Engineering of the University of Palermo has been involved for a long time now in the study of the nuclear response of the WCLL-TBM paying a particular attention to its power deposition, Tritium production and radiation damage features.

The present paper is focused on the continuation of the study of the WCLL-TBM nuclear response in ITER-FEAT, being specifically oriented to the study of the effect of the typical C-shaped tubes on the WCLL-TBM nuclear response and, in particular, on its top and bottom caps one. The impact of the structural material grade on the aforementioned response has been investigated too together with the coolant one.

A 3D heterogeneous model of the most recent design of the WCLL-TBM has been set-up simulating realistically its new lay out and taking into account 9% Cr martensitic steel as reference structural material. A particular attention has been paid to the simulation of the characteristic C shape of the breeder zone double walled tubes to study its influence on the module nuclear response. The WCLL-TBM model has been inserted into an existing 3D semi-heterogeneous ITER-FEAT model accounting for a proper D-T neutron source.

Analyses have been performed by means of MCNP-4C code running on a cluster of four workstations through the implementation of the PVM software. A large number of histories (10.000.000) have been simulated for each analysis and the obtained results are affected by statistical uncertainties lower than 3%. MANET and pressurized Helium have also been considered, respectively, as alternative structural material and coolant.

The main nuclear responses of the TBM have been determined, such as detailed power deposition density, material damage through DPA and He and H gas production rate, radial distribution of tritium production rate and daily tritium production in the module. The obtained results have shown as the presence of the C-shaped tubes in the breeder produces an increase in the power deposition density in the caps proximity, which could play a pivotal role in the thermo-mechanical design of the WCLL-TBM structure. At the same time, the influence of the structural material grade and of the coolant have been investigated, observing that the former is quite negligible while the second is pivotal, mainly as far as the TBM Tritium production capability is concerned.

J-60

NEUTRON TRANSPORT BENCHMARK ON IRON USING A WHITE HIGH-ENERGY NEUTRON FIELD

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The International Fusion Material Irradiation Facility (IFMIF) will provide an accelerator based intense neutron source for high fluence irradiation of fusion reactor candidate materials. A white high- energy neutron field (extending up to ≈ 55 MeV neutron energy) is produced by 40 MeV deuterons impinging on a thick lithium target. Dedicated computational tools and data are required to enable neutronic design calculations for IFMIF so as to provide the neutron and photon flux spectrum distributions as well as nuclear responses of interest. These tools must be capable of simulating the transport of neutrons generated through the d-Li reactions in the Lithium target. Neutron cross-section data for neutron transport calculations must be provided above 20 MeV, which is the upper energy limit of the standard nuclear data libraries. Such cross-section data have been evaluated in a collaboration of Forschungszentrum Karlsruhe (FZK) and the Institute of Nuclear Power Engineering (INPE), Obninsk, up to 50 MeV incident neutron energy for a number of selected important nuclides including the major iron and chromium isotopes. In addition, a series of cross-section data evaluations up to 150 MeV became available with the APT project from the Los Alamos National Laboratory (LANL). Validation of these data evaluations is required to ensure they provide reliable results when applied in neutronics design calculations. This can be achieved through integral benchmark experiments and their computational analyses.

Such a benchmark experiment has been conducted on iron at the U-120M cyclotron of the Nuclear Physics Institute (NPI), Řež. The $D_2O(^3He,xn)$ reaction was employed with 40 MeV helium ions to produce an IFMIF-like white neutron source spectrum extending up to 35 MeV neutron energy. A cylindrical iron slab with a thickness of 20 cm was placed 30 cm away from the neutron source in beam direction. Neutron spectra were measured with a NE213 scintillation detector 340 cm from the neutron source at 0 and 20 degree, respectively. The energy spectra from 3.5 to 35 MeV were obtained by using the pulse height unfolding technique.

The computational analysis of the experiment was performed with the MCNP4C Monte Carlo code and iron cross-section data from the FZK/INPE and the LANL evaluations. A detailed 3D geometry model was devised to accurately represent the experimental set up including neutron source, iron assembly, detector shield and experimental hall. The neutron source distribution was modelled on the basis of differential angle-energy neutron yields obtained through measurements of the bare $D_2O(^3He,xn)$ source. This approach was checked by comparing the simulated neutron source spectrum to the background neutron spectrum measured with a shadow bar located between source and detector. The comparison of calculated and experimental neutron transmission spectra has indicated good agreement for the FZK/INPE iron data and overestimations up to 50% for the LANL data.

J-61

VALIDATION OF ACTIVATION CALCULATIONS USING THE INTERMEDIATE ENERGY ACTIVATION FILE IEAF-2001

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The International Fusion Material Irradiation Facility (IFMIF) will provide an intense high-energy neutron field extending up to ≈ 55 MeV neutron energy. To enable activation calculations in the high-energy domain, a suitable activation data library, the Intermediate Energy Activation File IEAF-2001, has been recently developed by a collaboration of Forschungszentrum Karlsruhe and the Institute of Nuclear Power Engineering, Obninsk. The IEAF-2001 activation library comprises target nuclides from Z = 1 (hydrogen) to 83 (bismuth) up to 150 MeV neutron incidence energy. The activation code ALARA (Analytical and Laplacian Adaptive Radioactivity Analysis), developed at the University of Wisconsin-Madison, has the ability to handle the numerous activation reaction channels considered in the IEAF-2001 data library. IEAF-2001 application tests have been previously performed to demonstrate its applicability and suitability for IFMIF activation analyses.

This paper is devoted to validation analyses of the IEAF-2001 data library on the basis of activation calculations with the ALARA code. The validation procedure comprises several steps including various benchmark calculations and computational analyses of integral activation experiments. In the first step, ALARA activation calculations for the low activation steel Eurofer were benchmarked against FISPACT activation calculations. Since FISPACT is not capable of using IEAF-2001 cross-section data, an irradiation in the first wall neutron spectrum of a fusion Demo reactor was considered so as to enable the use of the European Activation File EAF-99 with FISPACT. The same EAF-99 data have also been adopted in the IEAF-2001 library for the energy range below 20 MeV. The second step was devoted to the testing of the IEAF-2001 data against integral activation measurements performed previously at FZK on samples of SS-316 and F82H steel, pure vanadium and V-Ti-Cr alloys using a d-Li neutron source with a white energy spectrum extending up to 55 MeV. The final step addressed the issue if the activation and transmutation of Eurofer in a fusion Demo reactor first wall can be appropriately simulated in the IFMIF high flux test module.

The computation benchmark revealed good agreement for the nuclide and activity inventories, the afterheat and the dose rate as calculated by ALARA and FISPACT. Measured activities can be predicted satisfactorily by ALARA/IEAF-2001 calculations for the majority of the considered radio-nuclides. Several discrepant results indicate, however, the need for improvements of specific IEAF-2001 cross sections. Activation and transmutation rates of Eurofer were shown to largely agree under irradiations in the IFMIF high flux test module and a fusion Demo reactor first wall.

J-62

ELEMENTARY DEVELOPMENT FOR BERYLLIDE PEBBLE FABRICATION BY ROTATING ELECTRODE METHOD

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Beryllium pebble is applied in the blanket as a neutron multiplier. However, it may not be applicable to the DEMO blanket that requires higher temperature (~900˚C) and neutron dose (~20,000 appmHe, ~50dpa) because of large swelling, hydrogen generation by the reaction with steam in case of accident and large tritium inventory. Therefore, beryllides ($Be_{12}Ti$, $Be_{12}V$, etc), which have high melting point and good chemical stability, have been expected as promising candidates for advanced neutron multipliers. But, it has been not easy to fabricate beryllide pebbles by rotating electrode method that had been reference fabrication process as to beryllium pebble, because of the brittleness of electrode rod. Previously $Be_{12}Ti$ rotating electrode was fabricated by HIP process (Hot Isostatic Pressing) and trial pebble fabrication was performed with the rotating electrode method. But, it was not successful because electrode broke by thermal shock during the arc heating because of the brittleness of the electrode. In this study, the new procedure for pebble fabrication was developed with $Be_{12}Ti$ that is one of most promising candidate materials.

In order to clarify the cause of the brittleness i.e., porosity due to the HIP process, large grain size, original brittleness, etc., the relationship between the microstructure and the ductility was studied with arc melting method. Then, it became clear that 1) the structure with melting process had fewer porosity than that with HIP process, 2) it was difficult to improve brittleness by heat treatment for stoicheometric composition.

In order to improve the ductility by structure control, the specimens with 5at%Ti, 7.7at%Ti (stoicheometric), 9at%Ti and 15at%Ti were fabricated, then microstructure observation and hardness test were performed. Hardness was 650, 1100, 1160 and 1230 respectively. The microstructure showed that Be-5at%Ti had finer structure with $Be_{12}Ti$ phase and α-Be phase. Next, the small electrode of Be-5at%Ti was fabricated and the rotating electrode method was performed as thermal shock test. The electrode withstood against the thermal shock from arc heating and some pebbles were obtained. The pebbles were observed and it was clear that pebbles were dense and had fine structure consisted of $Be_{12}Ti$ phase and α-Be phase. From these results, bright prospect for beryllide pebble fabrication was obtained.

In this symposium, the results of the chemical composition optimization and the results for Be-V system will be also presented.

J-63

THERMAL CONDUCTIVITY OF NEUTRON IRRADIATED $Be_{12}Ti$

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$Be_{12}Ti$ which is one of beryllides is expected as the neutron multiplier for fusion DEMO blanket that requires low swelling at high temperature, low hydrogen generation by the reaction with steam in case of accident and low tritium inventory. Therefore, beryllides that have high melting point and good chemical stability have been expected as promising candidates for advanced neutron multipliers. Thermal property is one of the important properties to design the blanket. Therefore, thermal conductivity of neutron irradiated $Be_{12}Ti$ were measured to evaluate thermal property of the fusion blanket in this study.

$Be_{12}Ti$ specimens (ϕ8 mm $\times$ ^{t}2 mm) were fabricated by HIP (Hot Isostatic Pressing) process from beryllium and titanium powder, and were irradiated by JMTR (Japan Material Testing Reactor) with a total fast neutron fluence (E > 1MeV) of 4 X 10^{20} n/cm^2 at 330, 400 and 500˚C. Thermal diffusivity and specific heat of un-irradiated and irradiated one were measured up to 1000˚C by laser flash method, and thermal conductivity was calculated.

Thermal conductivity of un-irradiated $Be_{12}Ti$ was constant up to 1000˚C. It was quarter of that of beryllium at R.T. and was half of that of beryllium at 1000˚C since thermal conductivity of beryllium decreased with heating temperature increasing. Thermal conductivity of irradiated $Be_{12}Ti$ was quarter of that of un-irradiated one and increased with temperature increasing. Finally the value was same as un-irradiated one at about 800˚C. After re-heating, the measurement was performed for the same specimens, and the value was same as un-irradiated one up to 1000˚C. It was considered that irradiation defect decreased thermal conductivity and the defect was recovered by annealing effect. In order to discuss the feasibility as neutron multiplier for the blanket, the effective thermal conductivity in the pebble bed was calculated with several models. The effective thermal conductivity in the pebble bed with $Be_{12}Ti$ and beryllium metal was calculated as binary packing (packing fraction 80%) with ϕ1mm and ϕ0.2mm pebbles for the un-irradiated and the irradiated using the measured data in this test. The values for the un-irradiated $Be_{12}Ti$ and the irradiated $Be_{12}Ti$ were close to those of beryllium at higher temperature.

In this study, thermal conductivity for un-irradiated and irradiated $Be_{12}Ti$ was measured. The effective thermal conductivity in the pebble bed was estimated with several models. It was clear that thermal property of $Be_{12}Ti$ was close to that of beryllium metal in the pebble bed especially at higher temperature. It was expected that $Be_{12}Ti$ could be used as neutron multiplier of fusion DEMO blanket.

J-64

MICROSTRUCTURAL ANALYSIS OF BERYLLIUM SAMPLES IRRADIATED AT HIGH TEMPERATURE

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Among the presently available low-Z materials beryllium represents one of the most promising candidate materials to be used as protection of the first wall and as neutron multiplier in the blanket of a next-step fusion reactor. Both sintered-product blocks and pebbles have been considered and research and evaluations are underway to study the characteristics of several material grades associated with safety, tritium release, heat transfer, thermal-mechanical and irradiation stability.

The knowledge of the tritium and helium release kinetics as a function of neutron fluence and temperature is relevant for beryllium fusion applications. The tritium and helium behaviour in neutron-irradiated beryllium is a complex function of the irradiation history (e.g. flux, irradiation temperature, time at temperature, etc.). Release kinetics is, therefore, expected to be dependent on the particular trapping mechanisms and, in particular, to be hindered both by structural sinks (physical trapping) and by beryllium oxide impurities (chemical trapping).

Several types of beryllium samples were irradiated in the BR2 reactor at different temperatures but with the same irradiation damage. In this paper the results of a series of microstructural analyses followig out-of-pile annealing tests up to 1000°C aiming at investigating both tritium and helium release kinetics from the S-65C beryllium grade produced by Brush Wellman are presented and discussed. The effects of crack formation eventually leading to a concurrent release of tritium and helium are also discussed.

J-65

THERMOHYDRAULIC CALCULATION OF THE HIGH FLUX TEST MODULE COOLED WITH HELIUM

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A reference design of the High Flux Test Module (HFTM) is basically a vessel with a number of irradiation rigs. The irradiation rigs have a rectangular cross section and contain the capsules with material specimens to be irradiated. The test section structure is heated, first of all, due to nuclear heating. Additionally, it is foreseen that at normal operation 15–25% of the total heat deposition in the specimens should be provided by ohmic heating. The rigs are cooled with helium. The thermohydraulic simulation of the HFTM is carried out with the STAR-CD code. The HFTM itself is considered to consist of three parts: the irradiated, the lower and the top sections. The irradiated section is simulated as four parallel boxes, each containing four rigs with samples. The rigs are positioned with equal gaps between them. The lower and the top sections are identical, consisting of four solids located symmetrically relative to the rigs. A standard k-ε high Reynolds number turbulence model is used for the simulation. Thermohydraulic calculations are carried out taking into account the flow compressibility and dependency of the helium thermal conductivity on the temperature. The heat deposit distribution is taken from the nuclear calculations and is applied to the model simulated. The results of the calculation show that the lower part of the test section smoothes effectively the velocity distribution at the input to the irradiated section. The pressure loss in the test section can be reduced with the help of an appropriate choice of geometry at the input and output of the test section. A hydraulic model of the small channel was developed for the study of thermal-hydraulic processes in He-cooled gaps. The results of the calculation will be validated by the experiments with the ITHEX Test section. The Re number for the flat duct flows with the width of 1–0.8 mm is in the range 5.000 < Re < 10.000. There is a not fully developed turbulent flow, where the viscose layer near the wall and the transition zone can take a large part of a channel cross-section and remarkably influence the heat transfer. The choice of an appropriate turbulence model is very important for the calculation of the near-wall region and to account for the low-Re-number effects in the flow. The test results show that the standard k-ε high-Re-number model and two-layer model are no valid for the flow calculations in small channels and cannot be applied in our case. The low-Re-number model, in which the general transport equations for k and ε are solved everywhere, including the near-wall regions, and equation for the dissipation rate has an additional term to account for low-Re-number effects, is suitable for this case. The inlet turbulence conditions influence considerably the results of the flow calculation in the small and short channels. Therefore, the following approach is used for the model under consideration. The high-Reynolds number turbulence model is used for the thermohydraulic calculations of the HFTM on the whole. The distribution of the main parameters obtained with the high-Reynolds number model is applied as input data for the low-Reynolds number model for more detailed simulation of the test section itself.

J-66

THERMAL MECHANICAL ANALYSIS OF A SOLID BREEDING BLANKET FOR ITER FEAT FUSION REACTOR

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The ITER-FEAT Breeding Blanket (B.B.) will be quite different from the design defined by the released version named "ITER-98" from the size as well as power point of view. In fact ITER-FEAT global dimensions and output power have been approximately reduced by a factor 3. In the ITER-FEAT foreseen program, the B.B. tests have been postponed in the final phase, that is in the last 10 years of tokamak operation (Enhanced Performance Phase – E.P.P.). During the initial operation phase, the B.B. will be substituted by means of a Shielding Blanket (S.B.) which doesn't have any tritium breeding aims. In the time period (from now to the beginning of the E.P.P.) research groups are called to develop their innovative designs which would be tested during the E.P.P. in some ports of the tokamak equatorial plane. This paper deals with a thermal mechanical analysis of a solid breeding blanket which could be used in the ITER-FEAT reactor. The breeding blanket taken into exam is made of:

- tubes of pellets of a Lithium compound as breeder,
- Be pebble bed as neutronic multiplier,
- cooling plates (using pressure water or Helium as coolant),
- stainless steel as structural material.

The use of pellets as breeder and multiplier has been demonstrated to be the best solution in order to limit the damage of the material mechanical properties, due to the radiation. Nevertheless the thermal-mechanical pebble bed behaviour is not well known yet. Some experimental tests have been performed, considering simplified geometries, in order to investigate the thermal-mechanical behaviour of pebble beds. In this paper a theoretical model of a single pellet under compressive static and dynamic loads as well as thermal loads is reported. The model considers the elastic-perfectly plastic behaviour of one pellet under compressive loads determining in such a way its stiffness. The pebble bed global stiffness is obtained considering the theoretical configurations of the pellets in lattices, characterised by a packing factor. The values of the packing factor (γ) ranges between 0.52 and 0.74 depending on the pellet configuration. Theoretically, the pellets can be assembled in three different configurations: a simple cubic arrangement ($\gamma = 0.52$) or a centred body cubic arrangement ($\gamma = 0.68$) or a face centred cubic arrangement ($\gamma = 0.74$). The lattice stiffness is obtained considering the assembly of the pellets as non linear springs in a serial or parallel configuration. In same manner the pebble bed stiffness is obtained considering the lattices as non linear springs in a serial or parallel configuration. The comparison between the experimental and theoretical results, obtained in a static test as well as a cyclic and creep test, shows a good agreement. In order to analyse the complex geometries of the breeding blanket, this model has been implemented in a FEM code. The preliminary results of the numerical simulations seem to confirm the capability of the model to analyse the complex behaviour the pebble beds.

J-67

PHYSICOCHEMICAL PROCESSES IN BLANKET CERAMIC MATERIALS

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The helium-cooled pebble bed (HCPB) will be one of alternative solutions of a blanket zone for the ITER and DEMO reactors. Ceramic Li_4SiO_4 or Li_2TiO_3 pebbles will be an operating material in the HCPB. Because the material will operate under hard conditions (temperature up to 950 °C, the neutron flux up to 2 $MW \cdot m^{-2}$, the magnetic field up to 10 T, etc.), possible changes of the composition and structure of the ceramics and their effect on the tritium release have to be predicted. Radiolysis, the effect of magnetic field on the radiolysis and the tritium release of the Li_4SiO_4 (FZK) and Li_2TiO_3 (CEA) pebbles selected by the EFDA were investigated. Electrical properties of the ceramics were investigated as well.

Radiation-induced defects (RD) form initially in radiolysis of the ceramic pebbles. RD aggregate into products of radiolysis (RP) – colloid lithium, molecular oxygen. The colloid lithium can form thermally stable LiT causing the tritium retention in the ceramics up to 900 °C. The efficiency of radiolysis decreases with increase of the temperature at irradiation. Processes of recombination prevail at 600 °C. The optimum conditions for the formation of colloid lithium in the Li_4SiO_4 pebbles are at the dose 10–20 MGy and 150–200°C. No colloid lithium was observed in Li_2TiO_3. The magnetic field facilitates the radiolysis (the efficiency of radiolysis increases by 20–25% at 2 T). The Li_2TiO_3 pebbles have the high radiation stability – the degree of decomposition is 10^{-3}% at 500 MGy. Their stability is much greater than that of Li_4SiO_4 pebbles.

The intense magnetic field retards the tritium release from the ceramic pebbles by changing the path of volume diffusion of T^+. The effect depends on the grain size of ceramics. The grain size can increase from 1–5 μm to 50–70 μm as a result of the combined action of temperature and radiation at blanket operation. The critical grain size, at which the complete tritium retention in an inner part of a grain can take place, is estimated to be 30–40 μm at 10 T.

The ceramic materials investigated (Li_4SiO_4 and Li_2TiO_3) have ionic conductivity. Their conductivity is 10^{-2}–10^{-1} $S \cdot cm^{-1}$ at 800–900°C. Because the volume diffusion of tritium is proportional to the conductivity, the conductivity can be used to predict the tritium release at the different degrees of lithium burn-up in ceramics. $Li_2Ti_5O_{12}$, which corresponds to 60% lithium burn-up, has the conductivity $2.8 \cdot 10^{-4}$ $S \cdot cm^{-1}$ at 300°C, whereas Li_2TiO_3 has the conductivity $(0.06–4) \cdot 10^{-6}$ $S \cdot cm^{-1}$ at 300 °C. $Li_{1.9}Ti_{0.9}Nb_{0.1}O_3$ ceramic samples have the conductivity at 500°C approximately 10 times greater than that of undoped Li_2TiO_3 samples synthesised under the same conditions.

The results obtained enable to predict the properties of Li_4SiO_4 and Li_2TiO_3 ceramics during long-term operation of the HCPB.

J-68

ON AN OPPORTUNITY OF RADIOCHEMICAL PROCESSING OF VANADIUM-CHROMIUM-TITANIUM ALLOYS TO BE USED AS STRUCTURAL MATERIALS IN FUSION REACTORS

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The vanadium-chromium-titanium (VCrTi) alloys are candidate structural materials for the demonstration fusion reactor DEMO. In particular, V-5Cr-5Ti alloy doped with 5 wt.% Cr and 5 wt.% Ti is considered in the Russian DEMO design. Such alloys may have high level of induced activity due to impurities. After the irradiation of the VCrTi alloy in the first wall of DEMO to fluence of 35 $MW \cdot a/m^2$, the contact dose rate from it will be about 10 kSv/h. 10–50 years after the irradiation completion, the main contributor to the dose rate will be ^{60}Co produced by Co, Ni, Cu and Fe impurities. Contribution of Eu, Sm, Nb, Mo, Ag, Al and Tb into the dose rate in 20 years after irradiation completion is also essential. About 80 years of cooling is required to bring the dose rate from irradiated VCrTi structures to 10 mSv/h enabling their remote handling. Therefore, radiochemical cleaning of the alloy components from the activation products is required to bring these materials back into production.

The radiochemical refining of the VCrTi alloy is basically similar to parting of fission-products from U and Pu. It also may involve the use of HNO_3 solutions in liquid-liquid extraction technology. In principle, H_2SO_4 and HCl solutions are also acceptable, but since the V-extraction from HNO_3 had been given earlier less attention than that from H2SO4, the top priority in the experiments was given to HNO_3 solutions. Tributil phosphate, the main extractant employed in the uranium radiochemical industry, is not suitable for VCrTi refining. Therefore, the solution of di-2-ethyl-hexyl-phosphoric acid (D2EHPA) in a hydrocarbon solvent (dodecane) was opted as an extractant in the experiments. Radioactive isotopes ^{152}Eu, ^{60}Co, ^{110}Ag, ^{99}Mo and ^{55}Fe were introduced in the solutions to determine the V-purification efficiency. The study was focused on the extraction of impurities from V, the main component of the VCrTi alloy. The extraction was studied depending on the medium, solution acidity and V-concentration. Changes in the impurity content were monitored radiometrically.

The experiments have shown that titanium can be extracted at the solution acidity ≥ 1 mole/l, vanadium – in the pH range from 2 to 2.5, and chromium – at pH equal to 4.0–4.5. The functional diagram of vanadium, chromium and titanium separation and removal, based on different conditions of their extraction (reextraction) is given. On the basis of the element distribution factors obtained in the experiments, segregation factors describing the efficiency of the VCrTi alloy component cleaning of the basic activation products (^{60}Co, ^{157}Eu, ^{99}Mo, ^{110}Ag,^{55}Fe) were estimated. The segregation factors are: For V/Mo ~ $2 \cdot 10^3$, for V/Co ~ 102–103, for V/Ag ~ 10, for V/Fe ~10, for V/Eu ~ 8, for Ti/Eu ~ 103, for Ti/Co ~ 105, for Ti/Ag ~ 104, for Ti/Fe ~ 200, for Cr/Mo ~ 104, for Cr/Eu ~ 103, for Cr/Co ~ 103. For majority of elements the separation occurs at the extraction stage. However, V-cleaning of Eu and Ti-purification of Mo take place at the reextraction. The data obtained in the study show that V, Cr and Ti after the radiochemical reprocessing of the alloy can be refabricated in the conditions of normal metallurgical process without additional biological shield.

J-69

WATER LARGE LEAKS INTO LIQUID Pb-17Li: FIRST EXPERIMENTAL RESULTS ON LIFUS 5 FACILITY

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Since several years ENEA is involved in experimental activities concerning the interaction between molten lithium lead alloy, in eutectic composition, and pressurised water in order to understand the behaviour of a WCLL blanket module in case of rupture of a cooling tube.

LIFUS 5 apparatus, designed and constructed to carry out the experimental campaign, after two tests performed to qualify all mechanical components and the data acquisition system, was operated at the ENEA C.R. Brasimone for the test n. 3-4-5, which were performed in thermal-hydraulic conditions similar to those foreseen in DEMO. Water was injected into the reaction tank at a pressure of 155 bar with different values of sub-cooling and different free volumes in the expansion vessel. The initial liquid metal temperature was fixed to 330°C.

The first pressure peak due to the water vaporisation and jet expansion was clearly recognized together with the subsequent pressure increase due to further water injection and hydrogen generation. The kinetics of pressure evolution was found to strongly depend on the compressibility of the system. As a matter of fact, this parameter seems to be much more relevant than the water enthalpy in determining the pressure evolution in the blanket module. In the experiments so far carried out, the maximum pressure peak, as detected in different points of the reaction and expansion vessels, never overcame the value of the injected water. Moreover in these three tests a significant temperature increase in the reaction vessel occurred, depending on the amount of the injected water.

K-1

VALIDATION OF INTRA AGAINST THE UPGRADED INTEGRATED INGRESS-OF-COOLANT EVENT TESTS

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The code INTRA (IN-vessel TRansient Analysis), is one of the reference codes specified by the ITER Joint Central Team and the European Community as a reference code for safety analyses of Tokamak type fusion reactors. INTRA has been developed to analyse integrated behaviours such as pressurisation, chemical reactions and temperature transients inside the plasma chamber and adjacent rooms, following postulated accidents, e.g. ingress of coolant water or air.

The code is presently involved in a validation program against the integrated ICE (Ingress-of-Coolant Event) tests performed at Naka, JAERI, Japan. The integrated ICE test facility is planned to investigate some ITER safety design parameters, and to provide experimental data for validation of safety analysis codes, such as INTRA, and methodologies that will be used for regulatory quality analyses. The tests are focusing on the Category IV multiple first wall pipe break events. The analyses provide pressure rise characteristics during the ICE events, two-phase flow characteristics and pressure drop through the divertor, condensation effects in the suppression tank and functional performance of the pressure suppression system. The construction of the integrated ICE test facility was completed in December 1999 and three different test series have been performed until now. Other codes involved in this validation programme are ISAS, MELCOR, PAX, CONSEN, TRAC-BF1 and CATHARE.

The present work addresses the INTRA model of the new ICE facility and the calculation results of two tests from the third series. The new ICE test facility consists of boiler, plasma chamber, simulated divertor, simplified vacuum vessel, relief pipe, pressure suppression tank and drain tank. During a test, water from the boiler is injected into the plasma chamber. As a result, steam is generated and the relief pipes lead the steam to the pressure suppression tank and eventually condenses while the rest water is discharged to the vacuum vessel through the simulated divertor.

The INTRA model consists of plasma chamber, divertor, vacuum vessel, pressure suppression tank and drain tank. The validation effort includes assessment of the modelling assumptions, comparison of the test result to the pre- and post-test calculation results and analyses of the results obtained. Both the pre- and post-test INTRA calculation results have shown good agreement with the test data.

K-2

MELCOR MODEL OF DIVERTOR COOLING LOOP AND DIVERTOR EX-VESSEL LOCA ANALYSIS FOR THE ITER-FEAT PLANT

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A MELCOR input model of the divertor components and associated heat transfer system has been developed in order to analyse the divertor ex-vessel LOCA that potentially could result in radioactive releases to the environment.

The divertor MELCOR input model is a detailed description of the divertor cooling system. The model includes the complete divertor primary heat transfer system with pumps, heat exchanger, pressurizer, divertor components, and associated pipings. The plasma heat loads on divertor plasma facing components are imposed by specified surface heat fluxes while the neutron heating is given as volumetric power in certain parts of the divertor components.

In order to analyse the divertor ex-vessel LOCA, the MELCOR divertor cooling loop model was joint with an other MELCOR model which includes First Wall/ Blanket cooling system, Tokamak Cooling Water System (TCWS) vault, Vacuum Vessel Pressure Suppression System (VVPSS), Drain System, Heating, Ventilating and Air Conditioning (HVAC), Standby Vent Detritiation System (S-VDS), and Emergency Detritiation System (EDS).

The MELCOR divertor cooling loop model has been used for thermal-hydraulic analyses of two accidents:

- Large ex-vessel pipe rupture in the divertor cooling loop during ITER-FEAT normal plasma operation
- Divertor ex-vessel coolant leak during the baking phase

The analyses provide the resulting coolant flow, pressures, and temperatures within the divertor cooling system, the TCWS vault and inside the vacuum vessel. The most important objectives for analysing the event are to estimate the pressure transient of the vacuum vessel and the TCWS vault, to show that post accident cooling is established to remove decay heat, and to evaluate the amount of in-vessel hydrogen generation and radioactive releases.

K-3

MITIGATION OF HYDROGEN RISK IN FUSION REACTORS: EXPERIMENTAL RESULTS

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One of the major accidents in fusion reactor is the "Loss Of Coolant Accident", which occurs in case of a breach of the cooling system first wall. It results in a water vapour ingress in the torus, which reacts with the Beryllium of the wall. This oxidation of Beryllium leads to the production of Hydrogen, with the associated risk of explosion in case of Oxygen presence. Thus, it is necessary to develop a mitigation strategy to handle this risk. The absence of Oxygen in the torus during normal operation forbids the usual methods based on O_2/H_2 reaction, using catalytic recombiner. Previous studies showed that it was possible to oxidize Hydrogen with metallic oxides, such as MnO_2 / Ag_2O mixture. The study presented here is devoted to the determination of Hydrogen elimination kinetics based on this process.

Ideally, the reaction should be fast enough to keep Hydrogen concentration below 4% (Lower Flammability Limit in dry air) at any time. In order to achieve this goal, it is necessary to get the knowledge of the process mechanisms and associated kinetics. Kinetics parameters are studied both through experimental and theoretical approaches.

Experiments are performed in the CIGNE facility. The reacting oxides (MnO_2/Ag_2O) are pressed and then crushed to obtain pellets, which are gathered as a fixed bed. The reacting gas, usually a mixture of H_2 and N_2, flows through the fixed bed where Hydrogen is converted into water by reduction of metallic oxides. The reaction progress is monitored through gas and solid temperature evolution, as well as outlet gas composition. The various parameters studied include flow rate, inlet gas temperature, water content in the inlet gas, …

In parallel to these experiments, a model is developed, based on the heterogeneous catalysis models. Two regimes are then discussed : chemical regime, where the reaction rate is limited by the solid/gas reaction, and diffusion regime, where the limitation is due to the diffusion of gas in the particle pores.

The reaction kinetics data, i.e. reaction order (n), activation energy (E) and kinetics constant (k_0) are then calculated fitting CIGNE experimental results with the model. Another parameter necessary to model the diffusion regime is the diffusivity (D_i) of gas in the solid. Diffusivity is derived from semi-empirical correlations, based on gas and support properties. The model will be used for designing of a Hydrogen elimination system, adapted to ITER, with regards to the quantity and shape of oxide and the reactor design. In this paper, we introduce the strategy and present experimental results.

K-4

MATERIALS ACTIVATION IN LONG TERM BLANKET CONCEPTS

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One of the main issues in fusion reactor studies are the ones related to the level of activity induced in the materials used, especially in the blanket which is the most exposed component. That issue has a great resonance to the public opinion for the acceptance of fusion as a competitive source of energy compared to the other nuclear and not nuclear sources.

From the activation results many considerations relevant to the operation of the reactor and its costs are drawn: the time to reach hands on level of key components for maintenance, safety analyses in case of accident or waste disposal.

As different concepts have been proposed for the long term concepts, designers and fusion budget technical committee would like to have a comparison between them based on common scenario, otherwise numbers could be misleading.

In the frame of an EFDA contract a neutronic study has been performed on two blanket concepts proposed for the Power Plant reactor using materials developed in the frame of the long term strategy: the Tauro and the Dual Coolant Liquid Lead (DCLL). A 3-D simple model of the two concepts have been set-up, assuming same main parameters of the machine (e.g.: major and minor radius, elongation …). The radial layout of the first wall and blanket has been reproduced avoiding when possible material mixtures or homogenisation. Neutron fluxes have been calculated in four topical selected zones by means of MCNP 4C code. Activation calculation has been performed by means of EASY 2001 code. An irradiation cycle for a total 5 years at full power at 3 GW has been assumed. Dose rate and activity together to other physical quantities related to the activation have been calculated. The materials for which activation is calculated have three different levels of impurities: real material, with achievable impurity reduction and without impurities. Information of the main isotopes giving the highest contribution for each quantity is given as sensitivity analysis to the composition.

The present paper will give a summary of the calculation performed useful to be used as a reference handbook by researchers working in the material programme.

K-5

EX-VESSEL BREAK IN ITER DIVERTOR COOLING LOOP ANALYSIS WITH THE ECART CODE

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In the frame of the Generic Site Safety Report GSSR for the ITER experimental plant, several accident analyses have been carried out to quantify in detail the radiological risk linked with the possible releases. In this context a hypothetical double-ended pipe rupture of the largest pipe was analysed to bound all possible leaks in the ex-vessel section of a divertor primary heat transfer system during pulse operation. That analysis was performed using a thermal-hydraulic system code (Athena), a containment code (Intra) and an aerosol transportation code (Naua). The present paper shows the results obtained by using, for the same accident, only the nuclear source term ECART code. A comparison with the results given in GSSR is also presented and discussed.

The ECART code is a computer tool that permits the simulation of chemical reactions and transport of radioactive gases and aerosols under two-phase flow transients in generic flow systems, using a built-in thermal-hydraulic model. It was originally designed and validated for traditional nuclear power plant safety analyses, and has been internationally recognized as a relevant nuclear source term codes for nuclear fission plants. In parallel with classic nuclear safety analyses, this code is presently employed in the evaluation of advanced nuclear designs, and as a support to risk assessment studies for process industry. The present work represents a significant step in the verification of ECART capability to treat a typical accident sequence of ITER. After the implementation of specific fusion reactors models for the oxidation reactions of beryllium, graphite and tungsten in air and steam, the code is used to manage a complete accident analysis using the representative conditions and logics of the ITER plant design, by coupling the new chemical reaction models with thermal-hydraulics and aerosols transport.

The whole ITER plant has been schematised utilizing 14 control nodes and 28 junctions, also having flow characteristics function of time or of the up-stream pressure. For this particular ex-vessel sequence the focus has been on the main thermal-hydraulic path for the transport of dust particles outside the plant, formed by the vacuum vessel VV (including its engineering safeguards like the drain and the suppression tanks), the divertor cooling loop pipes and the vault. About the main thermal-hydraulic parameters, the first comparison of the two different methodologies (an integrated code vs. a codes chain) gives a quite good qualitative and quantitative agreement in the results; i.e. the mass flow-rates from the different breaks are comparable, the VV total pressure peak is about 150 kPa and the long term equilibrium pressure shows a very similar trend.

The outcomes of the comparison between ECART and GSSR results are encouraging for the fine-tuning of the code in the fusion safety context. Important parameters as maximum pressure in the vacuum vessel, for example, are quite similar in the two analyses. For others the different physical model adopted or the different nodalisation leads up to some discrepancies, discussed in the paper. The adaptation of the ECART code to the safety analyses typical in the nuclear fusion plants will continue in the future.

K-6

CONDITIONING METHODS FOR BERYLLIUM WASTE FROM FUSION REACTORS

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Future fusion reactors will generate large quantities of irradiated beryllium, which is a candidate material for the first wall armour shield and the neutron multiplier in breeder blanket designs. Although recycling is the primary option for this material, considerable amounts of beryllium will require permanent geological disposal after an interim storage period of 50 to 100 years. For this waste, appropriate conditioning methods need to be developed.

For technical beryllium grades irradiated in a fusion reactor, and after a decay time of 100 years, the specific activity, γ-dose rate and inhalation dose are dominated by transmutation products from impurities present in the metal. When detritiation of the beryllium shortly after leaving the fusion reactor is assumed, the most important radionuclides contributing to the γ-dose rate and inhalation dose are ^{60}Co, which is an activation product of the impurities Co, Fe, and Ni, and actinides, originating from traces of U in the beryllium (in the ppm range).

In this paper, four conditioning methods are discussed: cementation, bituminisation, virtification, and phosphatisation. The selection of a conditioning technique must be based on a number of criteria connected to both the conditioning process itself (complexity, cost…) and the characteristics of the obtained waste form after conditioning (compatibility, waste loading, compressive strength, impact resistance, fire resistance, radiation stability, retention of radionuclides…). As a reference point, the option of no-conditioning or disposal of beryllium in its metallic form is also presented. In the comparison between the different conditioning methods, emphasis is placed on the long-term behaviour of beryllium in the chemical environment presented by the immobilisation matrix. From this exercise, vitrification resulted as a viable method to immobilise the beryllium waste.

K-7

SEISMIC ANALYSIS OF ITER TOKAMAK INCLUDING INTERACTION WITH SOIL AND BUILDING

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A seismic event is, in many cases, the most demanding loading condition for the ITER tokamak. The goal of the investigation, reported here, has been to develop the procedure for coupled seismic analysis of the full tokamak machine together with the building and soil-structure interaction. This interaction creates a channel for the exchange of vibration energy between the tokamak and the building, as they share on a common basemat, and the frequency-range of the building frequencies is the same as for the tokamak (2.8 Hz tokamak first frequency, and about 2.3 Hz the building first frequency for average soil conditions).

A global 3-D finite-element simplified model of the ITER tokamak was used for a coupled seismic spectrum analysis. The model includes the main tokamak components - toroidal field magnet, poloidal field coils and central solenoid, gravity supports, vacuum vessel, thermal shield, and cryostat. Soil-structure interaction and interaction with the building are included in the model. Soil-structure interaction is modeled in a traditional way with 6 linear springs and viscous dampers. Coupled "translation/rocking" of the building basemat is taken into account.

A new procedure for interaction analysis is proposed, based on the modal effective masses and heights. The detailed model of the building is replaced with an equivalent simulator, which consists of the equivalent parallel set of one-degree-of-freedom simple oscillators: effective masses located at their effective heights, with the springs providing the same partial frequencies. The simulator creates exactly the same shearing force and overturning moment acting upon the base during any arbitrary transient seismic motion. To obtain these partial frequencies, effective masses and heights, it is thus enough to perform a modal analysis of the building only once. After this the simulator is used for different boundary and soil conditions together with the tokamak finite-element model finally obtaining the complete tokamak response. A full analytical explanation is also provided in this paper.

Using this new procedure the response of the ITER tokamak is found versus different parameters of the soil and versus uncertainties of stiffness. For some particular soil conditions the natural frequency of the building is very close to that of the tokamak, and for that reason critical resonance effects may take place.

K-8

EVALUATION AND OPTIMISATION OF OCCUPATIONAL EXPOSURES IN THE HALL OF EXPERIMENTS OF THE "LASER INTEGRATION LINE" AT CESTA

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In the framework of research on inertial fusion, experiments on a reduced-scale prototype of the future Laser Mega Joule (LMJ) are planned in the hall of experiments of the "Laser Integration Line" (HE-LIL) at CEA/DAM/CESTA. By 2005, part of these experiments will involve Deuterium-Tritium targets which fusion reaction produces radiations – neutron flux and photon radiation from activated materials mainly – that constitute a source of radiological exposure at workplace.

At the conception stage of this experiment hall, CEPN has performed, together with all entities involved in the realisation of the HE-LIL project – project manager, construction manager, experiment managers, radiation protection group – an evaluation of the radiological exposures of the staffs involved in the experiments and maintenance, followed by an optimisation study for reducing, when possible, by application of the ALARA principle, individual and collective exposures.

Results from the evaluation of individual and collective occupational doses show that doubling the upkeep staffs and delaying their intervention into the experiment hall may lead to a reduction of the annual individual dose (about 1 $mSv.y^{-1}$) for this category of workers by a factor 4 approximately, without increasing the total collective dose (6.7 man-Sv). Furthermore, a modification of the planning for intervention into the experiment hall for the technical staffs involved in the handling of "diagnoses" (detecting equipments) – the most penalising operations in terms of radiological exposures – may decrease the annual individual dose (about 0.35 $mSv.y^{-1}$) from 2 to 50%, reducing as well the total collective dose by 25% approximately. An option which would consist in modifying the conception of the "diagnoses" was also studied, leading to a reduction in the range of 15 to 30% of individual doses of the staffs concerned and approximately a 10% reduction of total collective dose.

This study is a good illustration of how the ALARA principle can be implemented at the conception stage of an installation. Due to the relatively low individual and collective exposures associated with operation and maintenance of the HE-LIL, the different options to reduce doses are more dealing with staffs and planning re-organisations. These options are characterised by a rather low cost and limited constraints of implementation.

K-9

TRENDS IN RADIATION PROTECTION: POSSIBLE EFFECTS ON FUSION POWER PLANT DESIGN

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Commercial fusion power is prognosticated to become available by the mid of this century. Since the design involves long-term issues, ranging over several decades, it is useful to try to foresee under what kind of regulations the first fusion plants are to be operated. Hence the design can emphasise such points, where future challenges are believed to be. In this paper, trends in the radiation protection regulatory system, and some views affecting it are discussed.

The system of radiological protection in practice consists of justification, optimisation (with respect to collective dose), and individual risk limitation principles. The basis for this lies in the existing scientific evidence of the effects of radiation. Even though it is believed that there is enough knowledge of them on a macroscopic level to be able to use radiation safely, there are several factors, which need further research relating to the actual mechanisms of the interaction between radiation and organism. The effects of radiation at cellular level e.g. due to possible correction mechanisms as well as interactions with other contributing factors are not completely understood at present. The next 50 years may also enhance our knowledge e.g. on individual risks to radiation and on the other hand on treatment of the health effects.

Lately a new concept, "controllable dose" has been introduced to radiation protection. The protection philosophy of controllable dose is based on the individual instead of collective. This would be a major change in the rationale of radiation protection and would certainly affect the regulational environment in which the future fusion power plants would be operating. On the other hand, in some applications the development is moving from optimization to minimization of radiation exposure, which would probably have a different effect.

Application of present regulations to a fusion power plant concept are considered, and expected doses from the future evolutionary fission plants, which are supposed to operate during this century, are discussed to give a perspective of the future occupational doses at nuclear power plants. Comparison of the collective doses from fission and fusion plants is not very straightforward, due to the differences of the design levels as well as levels of uncertainty. The existing collective dose estimates for future fusion power plants are in general slightly higher than the collective doses at modern well-operating fission plants, but the estimation of occupational doses in the fusion power reactor is subject to numerous uncertainties. Even though the current design phase of fusion power plants motivates the top-down dose assessment, it is crucial to aim at carrying out bottom-up assessments, where the spatial dose rate conditions as well as occupational attendance can be modelled in more detail taking into account the special features of a fusion plant. Such an approach would provide better tools to analyse the different alternatives, e.g. different remote control options, and achieve radiation doses which are as low as reasonably achievable (ALARA).

Since several issues relating both to our knowledge on radiation as well as to the practice of radiation protection may change drastically during the coming decades, it is crucial to continuously follow the development in this field also in the further design of fusion power.

K-10

SAFETY ISSUES ON LASER MEGAJOULE FACILITY

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Laser Megajoule (LMJ) is the major project in France for Inertial Confinement Fusion research. 240 laser beams will generate 1.8 MJ energy required to reach ignition. High 14 MeV neutrons amounts (up to 1×10^{19} neutrons) will be produced with a few hundred micrograms of deuterium tritium mixture. Construction phase should begin in the course of year 2003 and operation is planned to start at the end of the decade (around 2008–2010).

Neutron generation and tritium use in the targets will induce radioactive hazards such as:

- high neutron and gamma radiation during a shot,
- gamma radiation between shots due to activation of materials
- contamination of all in vessel equipments

Radioactive issues are not different from magnetic confinement fusion facilities, only neutron and contamination levels are slightly different.

Due to the high neutron levels expected (about 4 decades higher than in existing laser facilities), activation is an important issue for LMJ. It still remains an experimental facility and access inside the experimental bay is needed for maintenance of diagnostics and several sub-systems (laser, vacuum…).

Preliminary calculations have been performed to estimate dose rate expected for maintenance operators. Activation may be reduced by a suitable choice of materials used for all equipments that will be introduced in the target chamber area. Firsts results and ways of design optimisation based on ALARA principle will be given.

Increase of tritium amounts in next steps facilities induces a higher contamination issue. Decontamination processes have to be studied to reduce tritium level and make easier maintenance of internal equipments and diagnostics. Several technologies (laser, foams…) have to be considered. Phebus target chamber (CEA laser facility decommissioned in 1999) has been used to test a physico-chemical process based on acidic foams to characterise contamination level fixed in the alloy and decontamination efficiency. Results will be shown.

Activation and contamination issues have to be included in maintenance prediction to reduce personnel exposure, define equipments handling (hands-on or remote handled) and reduce handling time.

Preliminary information on wastes generation and environmental impact will also be given.

K-11

EX-VESSEL AND IN-VESSEL LOCAS FOR THE PPCS FUSION REACTOR

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The PPCS (Power Plant Conceptual Study) is a European-wide initiative to highlight the state of understanding of fusion research, both physics and technology, through a conceptual design of a fusion power plant. The work started in 2000, and is based on the findings of the earlier programs SEAFP, SEAL, SEAFP-2 and SEAFP99. In the present work safety analyses are performed for the Model A of the PPCS fusion reactor, including a water cooled lithium-lead blanket. The thermal-hydraulic code MELCOR was used for the calculations.

Two accident analyses have been performed. One is an ex-vessel LOCA (Loss Of Coolant Accident) in the FW/BL (First Wall/BLanket) cooling loop. A double ended pipe break is assumed to occur at the inlet to the HX (Heat eXchanger) resulting in a discharge of water into the vault. Due to the fast discharge of water from the cooling pipes, vibrations are assumed to damage 90 out of 6000 HX tubes, resulting in a discharge of steam from the secondary side of the steam generator into the vault. The pipe break is assumed to occur at plasma burn. The FPSS (Fast Plasma Shutdown System) intervenes in 3 seconds. The resulting plasma disruption causes a high energy deposition on the PFC's (Plasma Facing Component). The pipe break affects only one loop and the cooling continues in the other loops. The transient is run for 24 hours. Parameters studied are the coolant flow rate into the vault, the pressure in the vault, the temperatures of PFCs and the vault and releases of activation products and tritium to the environment. The analysis also includes optimisation of the dimensions of the ST (Suppression Tank) and DT (Drain Tank).

The second analysis performed is initially identical to the ex-vessel LOCA, but here the FPSS is assumed not to intervene, resulting in an increase in FW temperature until the melting point of the FW material causes an in-vessel LOCA. A break in the breeder blanket box is also assumed causing lithium-lead to enter the VV (Vacuum Vessel), and to react with the water and steam from the broken coolant pipes. When the coolant pipes are emptied a connection is established between the VV and the vault. Additional parameters studied in this transient are the coolant rate into the VV and the hydrogen production. An optimisation of the dimensions of the ST and the DT is also performed.

The results will help to provide information in the selection of an adequate blanket and containment concept for the PPCS fusion reactor.

K-12

SAFETY GOALS FOR A FUSION REACTOR

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This paper discusses safety goals for a fusion reactor, such as ITER as an example, by applying a new approach developed for fission power plants by the author to share an equivalent safety goal for fission power plants. In developing fusion reactors, such as the International Thermonuclear Experimental Reactor (ITER), it is necessary to specify safety objectives and goals to demonstrate from the viewpoint of safety the attractiveness of fusion and thereby provide a good precedent for the safety of future fusion power reactors. The safety objectives are such as; license-able in any site after making a site-specific adaptation in design, no evacuation of the general public even in an event of major accident, and minimization of wastes. Although the no-evacuation-objective is a loosely specified safety goal, it is not specific enough to address whether the plant is ”safe enough” in comparison with other competitive energy sources, such as fission power plants, since the next generation fission plants are being developed with a similar objective.

The author proposes a new concept of fusion safety goals, which extend a framework of the International Nuclear Event Scales (INES), while incorporating medium risk characteristics of fusion power plants. The INES is a means for promptly communicating to the public in consistent terms the safety significance of events reported at nuclear power plants and other nuclear facilities, and is now widely accepted as an international standard for event scales. The criteria of severity classification of INES specified in releases are translated into doses by performing dispersion calculations for a typical site to estimate radiological consequences to the public. Qualitative health objectives after release of radioactive substances, focusing on tritium for inhalation and activated tokamak dust for land contamination, are deployed into master risk curves. To account for the health effect, an analysis of recent information of radiation health effects was made to include a surprisingly high risk, especially to the children of less than 10 years old at the time of exposure, now available through 50 years of mortality follow-up of the Life Span Study cohort of survivors of the atomic bombings, as well as significance of land contamination around Chernobyl.

The quantitative safety goals, expressed with master risk curves, can be used to identify an acceptable reliability in a safety space delineated by the probabilistic safety goals, by analyzing more frequent incidents with/without release. By doing these studies, it is shown that one of the most favorable safety advantage of the fusion reactor is in its low consequence characteristics, where an extended land contamination, equivalent to the case observed in the Chernobyl accident, is not the case. The public do not need to be evacuated, when an equivalent safety assessment is made as done in the safety goal study for fission reactors by the author.

K-13

CHEMICAL REACTIVITY OF BERYLLIUM DUST IN CAVITIES WITH OVERHEATED STEAM

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The amount of dust in the ITER vacuum vessel needs to be limited because of two major concerns: radiological hazard of activated and tritiated dust; and potential production of explosive hydrogen at chemical reactions with water or steam under accidents. The chemical reactivity of ~99% pure Be-powder obtained by impact grinding (≈15 micron size) in grooves simulating the gaps between the first wall tiles with overheated steam at 400–700°C was studied in the work that is described in the paper. The grooves (40 mm long, 0.5–1.0 mm wide; 10 mm deep) were machined in a SS-plate. The dust in grooves was compacted by vibration. The chemical interaction of a Be powder layer (200–350 micron thick) of the same amount uniformly distributed on a flat plate with overheated steam was studied too.

Schematic illustration of the experimental facility for the tests of steam interaction with Be-powder at a pressure above 100 kPa is shown. Kinetic and temperature dependences of the hydrogen generation rate under interaction of Be powder with particle size -15 micron and steam in temperature range 400–700°C are presented. The comparison of recent VNIINM's results and INEEL data on study of VNIINM's Be powders with particle size -15 micron are presented too. The comparison of the data shows a good agreement.

K-14

MEASUREMENT OF RADIATION SKYSHINE WITH D-T NEUTRON SOURCE

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The investigation of the radiation skyshine effect caused by D-T neutrons is important for the evaluation of the safety issue of fusion reactors. However there are few experimental data on the D-T neutron skyshine. Therefore, skyshine experiments have been carried out with D-T neutrons using Fusion Neutronics Source (FNS) at JAERI to evaluate the agreement of the skyshine dose rate.

The D-T neutron skyshine experiments have been carried out with the FNS 80 degree line D-T neutron source whose intensity was 1.7×10^{11} n/sec in the experiment for 56 hours. The concrete thickness of the ceiling and the wall of the FNS target room are 1 and 2 m respectively. FNS skyshine port with a size of 1 x 1 m square was opened during the experiment period. The radiation dose rate outside the target room, which was caused by the D-T neutrons through the skyshine port, was measured as far as about 550 m away from the D-T target point. Spherical remcounts dosimeters, ^{3}He and BF_3 counter were used to obtain the neutron dose rate. Secondary γ-rays were measured with Ge-semiconductor detectors and NaI scintillation counters. Also Nb and In foils and bubble detectors were used to evaluate neutron distribution across the skyshine port.

The highest total dose rate measured was about 1 μSv/h at a distance of 30 m from the D-T target point and the dose rate was attenuated to about 0.1 μSv/h at a distance of 150 m and 0.01μSv/h at a distance of 400 m. The measured neutron dose-distance dependency as far as about 150 m from the source position was in good agreement with dose rate calculation using Monte Carlo code MCNP-4B and JENDL-3.2 data and a line source model calculation represented by $(1/R)\exp(-R/L_m)$, where R and L_m are the distance from the source and the mean free path of the 14 MeV neutron, respectively. However the calculation with the line source model is in significantly better agreement with the measured values than the MCNP calculation at source distances greater than 200 m. The γ-spectrum above 6 MeV which is assumed to originate in (n,γ) reactions in silicon and iron caused by leakage neutrons from the skyshine port, was measured by Ge-semiconductor detector and NaI scintillation counter. Also it was shown that the dose rate of the secondary-γ ray is attenuated the proportionally to inverse distance which is in agreement with the line source model.

K-15

SHUTDOWN DOSE EVALUATION EXPERIMENT FOR ITER

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In the shielding design of the ITER machine, it is important to have a reliable estimation of the dose rate levels after the reactor shutdown (~10^6 seconds) for realising hands-on maintenance around the torus. In order to assure the hands-on maintenance possibility inside of the cryostat without excessive shielding, a high accuracy of the shutdown dose estimation is required. The objective of the shutdown dose evaluation experiment was to conduct experiments with complex geometry clarifying the required safety factors concerning the methodology for shutdown dose estimates developed for shielding design of ITER [1].

The experimental assembly consisted of the source reflector and the test region, which had a cylindrical shape. The test region simulated a maintenance area and was composed of type 316 stainless steel (SS-316). The source region and the test region were connected with cylindrical duct surrounded by SS-316 and water layers, which is a standard shielding structure of the ITER. Irradiation was conducted through 6 days. Total and average neutron yields were 2.1×10^{16} neutrons and 9.6×10^{10} neutrons per second respectively. The irradiation schedule was determined by the pre-analysis so that the shutdown dose could be measured at 10^6 seconds after the irradiation. The shutdown dose was measured from ~2 days to ~2 weeks after shutdown and the measured data.

The analysis of this experiment was performed by using the new direct one-step method [1], which was used for shielding design of ITER/FEAT. In the shielding design of the ITER/FDR, the conversion factor method was used to evaluate shutdown dose. Though the conversion factor method is convenient for rough estimation of shutdown dose, the design margin should be large because the uncertainty accompanied with the results may not be small. On the other hand, the direct one-step method employs 3-dimensional Monte Carlo transport code MCNP to calculate decay gamma ray source and also decay gamma ray transport. Therefore the method can eliminate uncertainty accompanied with conversion factors from neutron flux and activation rate to dose rate after shutdown. The nuclear data libraries for the evaluation of the neutron field and the decay gamma field were FENDL/2 and FENDL/2A. Shutdown dose rates evaluated by the direct one step method with FNEDL/2 library agreed with the experimental results with the tissue equivalent dose meter within the experimental error (~10%). The discrepancy between calculated data and measured data is small enough compared with the design margin and it was confirmed that the proposed method is feasible for the design calculation of ITER.

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K-16

ANALYSIS OF DOSERATE EXPERIMENT: COMPARISON BETWEEN FENDL, EFF/EAF AND JENDL NUCLEAR DATA LIBRARIES

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Recently, a neutronics experiment was conducted at the 14 MeV neutron generator FNG of ENEA Frascati to validate calculation of shutdown doserate outside a shield assembly made of stainless stee (AISI-316) irradiated with 14 MeV neutrons [1]. In particular, the purpose of the experiment was to validate neutronics analyses for ITER which are based on code calculations using more or less sophisticated models, and nuclear data with inherent uncertainties. The experiment was analysed using two different numerical approaches, both used in the ITER nuclear design: (1) a rigorous two-step method [2] which makes use of the MCNP transport code and the FISPACT inventory code, and (2) a direct one-step method using an *ad hoc* modified version of MCNP where neutron and decay gamma transport are handled in one single Monte Carlo calculation run [3]. In either approach, cross-section data from the FENDL/MC-2.0 (Monte Carlo transport) and FENDL/A-2.0 (activation) were used.

In the present work, the experiment was analysed with the two-step, rigorous method using three different nuclear data library packages developed for fusion for the purpose of comparison: the European libraries EFF (transport) and EAF (activation), the Japanese libraries JENDL–FF (transport) and JENDL-3.2 (activation), and the international libraries FENDL-2/MC (transport) and FENDL-2/A (activation). The use of several nuclear data library packages was possible thanks to the coupling scheme available in the two-step method, which enables an automated routing of the MCNP neutron flux spectrum distributions to FISPACT and the FISPACT decay gamma source distributions to MCNP. This makes it possible to perform the complex transport-activation-transport calculations in a rather simple and direct way, even when the experimental configuration is simulated with a large number of cells in the MCNP geometry.

All library packages were able to predict the measured dose rate within ±30% for decay times up to almost four months after irradiation. Differences in the range 10–20% where found in the predicted shut down dose rate calculations. Part of the difference was found already in the calculation of the neutron flux in the relevant regions of the shielding block. Comparison between the predictions from different libraries on neutron fluxes, decay gamma spectra and dose rate are analysed and presented in detail.

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K-17

NEUTRONIC DESIGN ISSUES OF THE WCLL AND HCPB POWER PLANT MODELS

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A Power Plant Conceptual Study (PPCS) is currently being conducted in the framework of the EU fusion programme with the main objective to demonstrate the safety and environmental advantages and the economic viability of fusion as a future energy source. This includes the conceptual design of models of a commercial fusion power plant, their safety, environmental and economic assessment and the demonstration of their credibility and viability. To this end, power plant designs with limited physics and technology extrapolations from present knowledge have been considered in the first place. With regard to technology, the water cooled lithium lead (WCLL) and the helium cooled pebble bed (HCPB) blanket concepts have been selected as feasible near term solutions satisfying the requirement for a credible and robust plant design. The WCLL and HCPB blanket concepts have been developed over more than one decade in the frame of the EU long-term programme accompanied by substantial R&D programmes in the participating Euratom associations. The WCLL blanket employs a quasi stagnant pool of the liquid metal breeder Pb-17Li cooled by pressurised water at conditions similar to those of a fission pressurized water reactor. The HCPB blanket is based on the use of lithium ceramics as breeder material, beryllium as neutron multiplier and high- pressure helium as coolant gas. Breeder material and multiplier are arranged as pebble beds between flat cooling plates. Both the WCLL and the HCPB blanket employ the low activation ferritic steel Eurofer as structural material.

This work addresses neutronic design issues of the two near term PPCS reactor models employing the WCLL and the HCPB blankets for the tritium and power production. Based on reactor parameters and neutron source distributions provided by UKAEA Culham, detailed three-dimensional torus sector models have been developed for the two PPCS reactor variants to enable proper design calculations with the MCNP Monte Carlo code. The models include the plasma chamber, poloidally arranged blanket modules, a bottom divertor port with integrated divertor, the vacuum vessel and the toroidal field coil. In accordance with the assumed toroidal segmentation of the blanket modules, torus sectors of 40° and 20° were considered for the HCPB and the WCLL plant models, respectively.

Neutronic design issues addressed in this work include the assessment of the tritium breeding performance, the nuclear power generation and its spatial distribution as well as the assessment of the shielding performance both with regard to the re-weldability criterion of life-time components such as the vacuum vessel and a sufficient radiation protection of the super-conducting toroidal field coils. While the tritium breeding performance of the two blanket concepts is similar, the HCPB blanket shows a higher energy multiplication which results together with a higher thermal efficiency in the power conversion system in a considerably smaller fusion power required for a given power output of the plant. The WCLL blanket, on the other hand, shows a better shielding efficiency resulting in a reduced overall thickness of the blanket/shield system. Both the WCLL and the HCPB concept can meet the requirements for a fusion power plant blanket with regard to their nuclear performance.

K-18

OXYGEN REACTIVITY OF A CARBON FIBER COMPOSITE

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Carbon Fiber Composites (CFCs) are often suggested as armor material for the first wall of a tokamak due to carbon's low atomic number, high thermal conductivity, and high melting point. However, these favorable properties of CFCs are accompanied by carbon's high chemical reactivity in an oxidizing media like oxygen and its high adsorption of tritium. The latter two properties of carbon suggest that during a Loss of Vacuum Accident (LOVA), the CFC armor will chemically react with the air ingress into the vacuum vessel and release the tritium that the CFC previously absorbed. The mobilization of this tritium and the carbon monoxide produced by the CFC-air chemical reaction are both safety concerns.

Accident analyses of a LOVA for a CFC-armored tokamak requires kinetic data on the chemical reactions of carbon in an oxygen flow stream. These reactions are temperature dependent and involve the production of carbon monoxide and carbon dioxide. For completeness, the kinetic data should include the chemical reaction's dependence on the CFC temperature, flow stream velocity, and CFC burn-off.

This paper discussed the chemical reactivity experiments that were performed with a 3-dimensional CFC in a 79%Ar–21%O_2 flow stream. Reaction rates were generated for CFC temperatures of 525, 600, 700, 800, 900, and 1000^{o}C and an Ar-O_2 flow rate of 100 standard cubic centimeters (sccm). Experiments were also performed at CFC temperatures of 700 and 1000^{o}C and an Ar-O_2 flow rate of 1000 sccm. The resulting reaction rates and activation energy compare well with the literature on carbon oxidation and burn-off. Clearly illustrated through plots of the experimental data are the three kinetic regimes of chemical kinetic control, in-pore diffusion of oxygen, and boundary layer diffusion. Lastly, it is concluded that carbon monoxide is the primary reaction at the surface of the CFC and carbon dioxide is readily produced in the flow stream beyond the CFC when the Ar-O_2 mixture has a low flow rate.

K-19

POWER PLANT CONCEPTUAL STUDY FOR THE DUAL COOLANT BLANKET CONCEPT

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The objective of this task of power plant conceptual study (PPCS) is to perform the conceptual design of the dual coolant (DC) blanket concept based on the use of helium cooled steel structures and self-cooled Pb-17Li breeding zone. The details for the DC concept study in regard to technology and plasma physic are to be selected in accordance with the overall strategy, which allows an extrapolation of the present knowledge between the near-term solutions for the helium-cooled pebble bed (HCPB) and the water-cooled lead-lithium (WCLL) blanket concepts, and the very advanced self-cooled Pb-17Li SiC/SiC (TAURO / ARIES) blanket concept. The PPCS is drawn extensively on the preparatory study on plant availability (PPA) carried out in 1999. This work is under the coordination of FZK in co-operation with CEA, UKAEA, VR and EFET.

The main features of the DC concept – following the ARIES-ST study – are the use of helium-cooled ferritic/martensitic steel (EUROFER) structure, self-cooled Pb-17Li breeding zone, and SiC/SiC flow channel inserts. The latter has no structural function but serves as electrical and thermal insulators and therefore minimize the pressure losses and enable a relatively high Pb-17Li exit temperature leading to a high thermal efficiency. Moreover, the use of oxide dispersion strengthened (ODS) ferritic steel as a 2–3 mm thin layer plated onto the first wall (FW) allows a relatively high working FW temperature, which is generally limited by creep rupture strength, of up to about 650°C which is about 100 K higher than that for ferritic/martensitic material. The partial's application of the ODS instead of full's application to the whole structure helps to avoid the fabrication difficulties encountered in welding of ODS structural parts.

In the preceding PPA study the potential of the DC blanket concept was investigated. Taking into account the temperature constraints for the FW (creep rupture strength) and the Pb-17Li breeding zone (corrosion), the latter was found to be decisive for the power limitations leading to the maximum values of neutron wall load and surface heat load of 5 MW/m^2 and 0.9 MW/m^2, respectively. Assuming a closed three-compression-stage Brayton gas turbine cycle for the power conversion system, a net efficiency for the blanket cycle of 44% was obtained. In the following PPCS study the DC blanket is normalized and adapted to a typical size of commercial reactors of 1.5 GWe which requires iterative calculations between the system code analysis and the blanket layout. The first system code calculation (UKAEA) for DC yields e.g. a major radius of 7.5 m, and a fusion power of 3.41 GW based on a net efficiency of 44% and a blanket energy gain of 1.17. For the power conversion system, the closed three-stage Brayton gas turbine cycle is further considered as reference solution. As an alternative to this a steam turbine system (VR) was proposed which leads to the same level of thermal efficiency of about 45% taken into account an integration of a He-cooled divertor into the power conversion system in order to exploit its waste heat and to increase the net efficiency.

K-20

FEASIBILITY OF D^3He/ST FUSION POWER REACTOR

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The objective of this study is to examine the possibility of a D^3He fueled spherical tokamak (ST) fusion power reactor with steady state operation. A D^3He fusion reactor that requires operation at high beta has inherent safety characteristics such as fewer neutron production, reduced radioactivity in structure and lower tritium inventory compared with DT and catalyzed DD fuel cycles. The ST concept is a very low aspect ratio ($A < 1.7$) that permits high beta operation. It also offers stability at large elongation that allows operation at large bootstrap current fraction. Therefore, it is one of the important issues to explore the engineering design feasibility of the center column of a D^3He/ST fusion reactor.

The D^3He/ST fusion reactor needs no fuel-generating blanket. The reactor configuration is therefore simplified by eliminating a center solenoid (CS) coil system. This is based on the assumption that the non-inductive current drive system, which consists of Electron Cyclotron Heating (ECH) and Neutral Beam Current Drive (NBCD), permits plasma initiation and current ramp-up process as well as plasma current-sustaining [1]. High temperature super-conductor is selected for the TF coils in order to cope with high magnetic field at the winding of TF coils and to reduce the circulation power. The conductor is based on Bi2212/Ag/Ag and its operation temperature is chosen at 20K [2]. Furthermore, the Japanese Cryogenic Steels (JJ1 and/or JN1) with the allowable stress σ_{all} of 700MPa are employed for the structures of TF coils. The inboard magnetic shield consists of low activation ferritic steel (F82H) and boric water in order to minimize the thickness of the neutron shield. The shield is capable of protecting the high Tc super-conducting magnet against radiation damage throughout the lifetime (35y) of the D^3He/ST fusion reactor. The thickness of the vacuum vessel and the thermal shield is assumed to be 0.20 m. The plasma beta limit scaling, which was parameterized with aspect ratio (A) and plasma elongation (κ) by C.P.C. Wong et al., is used to perform parametric investigations of D^3He/ST reactor system. As a representative energy confinement scaling, the ELMy H-mode energy confinement scaling denoted by IPB98 (y, 2) is adopted.

We developed a systems code that incorporates the plasma and engineering calculations, and performed parametric system studies to explore possible approaches for D^3He/ST fusion power reactors. As a result, characteristics of the D^3He/ST reactor system are clarified and the design window is specified under the constraints of the confinement-enhancement factor HHy2 < 2 and the Greenwald density limit ratio $n/n_G < 1.3$, etc. For an example of the design points, the following reactor parameters are obtained; Pe (net electric power) = 1400Mwe; Pw (first wall heat load) = 1.0 MW/m^2; A = 1.6; R = 7.1m; $\beta(\beta_N)$ = 39%(7.6); Ti(Te) = 50.0(46.1) keV; ne = 1.78E20/m^3; B(B_{max}) = 4.1(18.3)T; Ip = 94.6MA; Pw-n (DT-neutron first wall flux) = 0.03 MW/m^2; TF-coil center post radius = 1.56m.

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K-21

OPTIMAL NEAR-TERM AND ADVANCED FUSION POWER STATION DESIGN PARAMETERS FOR THE EUROPEAN POWER PLANT CONCEPTUAL STUDY

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The European Fusion Power Plant Conceptual Study (PPCS) is a collaborative study of conceptual designs for future commercial fusion power stations. Studies in the PPCS are focused on four Power Plant Models, with some variants. The intention is that all the Models will have very good safety and environmental characteristics, and will span the range from relatively near-term, based on limited technology and plasma physics extrapolations, to advanced models. The subject matter of this paper is the systems code analyses that have been performed within the PPCS, in order to specify the parameters of the four models that are receiving detailed study.

Two of the Models (Models A and B) are based on limited extrapolations in plasma physics compared to the design basis for ITER. The reasonableness of various projected advances were assessed by an expert panel. The systems analyses combine the physics assumptions in a self-consistent way with representations of engineering constraints. For Models A and B, the engineering constraints correspond respectively to use of blankets based on the European Water-cooled lithium-lead (WCLL) and Helium-cooled pebble-bed (HCPB) concepts, with associated water-cooled and helium-cooled divertors. There are important interactions and tradeoffs between the various assumed physics and technology constraints. The analyses integrate all these factors, and other considerations such as unit size and availability, to produce parameter sets with optimal economic characteristics.

Models C and D are based on more advanced concepts in physics and technology. The technology constraints correspond to more advanced blanket concepts, such as the dual-coolant (with steel and silicon carbide structures) concept and the self-cooled lithium-lead (silicon carbide structure) concept.

These Models are being developed and assessed in detail within the PPCS: but the results of the systems analyses themselves give rise to some conclusions. For Models A and B, with their relatively conservative physics, the most challenging areas are the divertor heat load capability and the need to drive a large fraction of the plasma current with the heating systems. The constraints tend to drive the designs to a large size, with relatively high recirculating power. Nevertheless, viable near-term power stations seem feasible. For Models C and D, projected improvements in physics and technology make feasible a plant size close to ITER-FEAT, with reduced recirculating power. In these Models, the plasmas are more highly shaped: this is a more challenging configuration for the technology integration.

The results overall suggest that acceptable first generation power stations can be accessed by a “fast track” route, through ITER without major materials advances, and that there is potential for a more advanced second generation of power stations.

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K-22

ASSESSMENT OF LITHIUM-LEAD / WATER INTERACTION IN WCLL BREEDER BLANKET – VALIDATION OF THE SIMMER COMPUTER CODE ON SPECIFIC EXPERIMENTS

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In case of a tube rupture within a water cooled lithium-lead (WCLL) blanket module the thermal interaction between injected water and lithium-lead can lead to a high pressure peak which could challenge the integrity of the module. Thus, it is necessary from the safety point of view to assess as accurately as possible the consequences of such an interaction. A series of tasks have been defined in the frame of the WCLL work programme which involve experiments (BLAST in the past and now LIFUS 5) and the modelling of the main physical phenomena using the SIMMER III computer code (multi-phase and multi-component thermalhydraulics in a 2D/Rz geometry). The experimental devices mainly provide for a reaction vessel containing or not tube bundles and filled with liquid lithium-lead (LiPb), a pressurised water injector into the reaction tank at the bottom of the vessel and an expansion tube joining the reaction vessel to an expansion tank partially filled with liquid LiPb. The main experimental parameters concern the water injection conditions (temperature and pressure), the LiPb temperature and the presence of obstacles inside the reaction vessel, the expansion tube diameter. The pressure evolution in the reaction tank is recorded during the test. The general objective is to validate the code models in order to be able to perform safety analyses for licensing.

The validation work programme comprises two main steps:

1. The interpretation of BLAST (BLanket Safety Tests) allows us to identify the main physical parameters which govern the interaction between lithium-lead and water and to verify that they are correctly taken into account in the SIMMER III code. The value of the pressure peak induced by the interaction is satisfactorily calculated even using a coarse mesh. Using a more accurate representation of the geometry of the mock up with a fine mesh, it is possible to correctly simulate the kinetics of the pressure evolution; in fact the refined mesh allows us to better characterise the behaviour of the water jet. A reference representation of BLAST for SIMMER has been issued which allow to simulate a series of tests in a satisfactory way.

2. The capability of SIMMER to predict the pressure evolution following the interaction between lithium-lead and water in reactor relevant conditions are assessed through pre calculations of LIFUS 5 tests; the operative conditions of LIFUS 5 are more representative of the reactor than BLAST. Using a correct representation of LIFUS 5 geometrical specificities (internal configuration of the reaction vessel, shape of the expansion tubes, …) it is possible to simulate the pressure evolution with the SIMMER code in a satisfactorily way.

K-23

INTEGRAL APPROACH TO ASSESS THE AVAILABILITY / RELIABILITY OF THE FUSION POWER REACTOR CONCEPTUAL DESIGNS

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key words: fusion power reactor conceptual design availability reliability integral approach

Fusion reactors are more complex than conventional and fission power plants. They operate under more severe operational conditions and require higher capital investments. For these reasons, plant availability is one of the basic criteria against which the attractiveness of a fusion reactor will be assessed. This assessment should begin during the early stages of the plant design in order to identify the major causes of plant unavailability and to define basic requirements. To achieve these goals, an overall approach, taking into account the functional interdependence between major systems/components in the plant, is needed.

The paper gives an overview of the functional dependence between different items involved in the determination of the plant availability. It proposes an integral-analytical approach where:

- Detection & Monitoring tasks (D&M),
- Failure Modes & Criticality Analysis (FMCA), and
- Maintenance (operational/preventive)

are considered and their functional interdependence is modeled.
The plant outage is described as a function of the unplanned and planned outages. The planned outage, which is a fixed parameter, results from a compromise between economic and maintenance procedure constraints and is optimized through a deterministic approach so as to minimize the plant downtime.

Unplanned outages depend on the occurrence rate and repair rate of failure modes. In some situations, they also depend on the detection rate of the component failure. As they result from random failures, the unplanned outage rate is a random variable, as well. The functional dependency between D&M, FMCA and maintenance will be described in terms of occurrence rate of given failure modes, detection success of the failure and the repair time of the failure.

This approach provides a qualitative and quantitative insight into the availability issue from a probabilistic point of view. It is intended as a tool to assist in the selection of design options and in the definition of technical requirements.

The quantitative analysis is strongly affected by the "quality" of the failure data. The paper will however focus on the definition of a method rather than on its practical application.

K-24

EXTERNAL COSTS OF SILICON CARBIDE FUSION POWER PLANTS COMPARED TO OTHER ADVANCED GENERATION TECHNOLOGIES

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This study was performed in the framework of the Socio-Economic Research on Fusion (SERF3), which is jointly conducted by Euratom and the fusion associations. Assessments of monetarized external impacts of the fusion fuel-cycle were previously performed (SERF1 and SERF2). Three different power plant designs were studied, with the main difference being the structural materials and cooling system used. The key variables and factors that significantly contributed to the value of externalities in the fusion fuel chain were identified and a set of design criteria and recommendations on how to reduce the external costs of fusion power was produced. In this third phase of the SERF project the external costs of three additional fusion power plant models using silicon carbide as structural material have been analyzed. External costs are those costs imposed on society by the fuel cycle but not reflected into the price of the electricity generated. This study has evaluated the external costs of three different power plant designs using silicon carbide as structural material. A comparison with other advanced generation technologies expected to be in use around 2050, when the first fusion power plant would be operative, has also been performed. These technologies include advanced fossil technologies, such as pressurized Fluidized Bed Combustion and Integrated Gasification Combined Cycle with carbon sequestration technologies; renewable technologies including fuel cells, photovoltaic systems and geothermal energy with energy storage devices and advanced fission reactors.

The study uses a methodology for evaluating, in a standardised way, the external costs of electricity generation by different fuel cycles previously developed by the Commission of the European Union in the frame of the “ExternE” project. The ExternE methodology is a bottom-up methodology, with a marginal and site specific approach. Quantification of impacts is achieved through the damage function or impact pathway approach that follows the sequence of events linking a burden to an impact and subsequent monetary valuation. This means that it involves siting a power plant, and calculating its contribution to the environmental and health situation locally, regionally and globally.

Fusion power plants using silicon carbide as structural material have higher efficiencies than plants using steel and this fact has a very positive effect on the external costs per kWh. External costs of these plants are in the lowest range of the external costs of advanced generation technologies indicating the outstanding environmental performance of fusion power.

K-25

OPERATIONAL BERYLLIUM HANDLING EXPERIENCE AT JET

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With its unique thermal properties and low atomic number, beryllium has been chosen as one of the materials favoured for use in next step fusion reactors such as ITER. However, as a toxic material the safety aspects of beryllium handling need careful consideration. The JET project in the UK has been using beryllium on a large scale since 1988, installed as first wall protection and in RF components, with up to 3000 kg on-site at one time. Considerable operational experience has been gained over the last 14 years in safe handling of beryllium components and beryllium contaminated materials. JET has developed a stringent regime for worker protection, and maintained extremely low exposures whilst conducting various maintenance operations involving vessel interventions, routine decontamination work and waste processing activities.

Although fewer solid tiles are used for current first wall configurations, considerable use is made of beryllium evaporators. The degradation of tile surfaces during plasma operations and the evaporated deposit produces dust and particles which mobilise readily to produce potentially harmful exposures. In-vessel interventions have required careful controls to avoid worker exposure, and over 15 separate manual vessel interventions have been conducted since beryllium was introduced. Procedures for handling beryllium require dedicated facilities, ventilation controls, confinement and respiratory protective equipment. Examples are given of installation and decontamination tasks undertaken, including the complete decontamination of the vessel to a virtually beryllium free surface. The methods employed for containing beryllium dust in these operations are described.

Experience has shown that the range of beryllium-in-air concentrations varies from ~ 800 $\mu g/m^3$ in the vessel during grinding work, to << 0.1 $\mu g/m^3$ for routine component handling and decontamination work. In the period 1988 to 2001, more than 77,000 personal exposure measurements were carried out. These show that when account is taken of the respiratory protection worn, 95% of exposures are < 0.03 μg (detection limit), 5% with detectable levels, and only 0.02% above exposure limit of 2 $\mu g/m^3$. More than 1200 beryllium workers have been engaged at JET in this period. Descriptions are given of the process of medical assessment and tests for beryllium workers. To date no significant health effect has been observed as a result of beryllium exposure to any of the JET workforce.

It is likely that the regulatory exposure limit for beryllium will be reduced from 2 $\mu g/m^3$ to 0.1 $\mu g/m^3$ in coming years. Future fusion devices will encounter even more challenging conditions involving the control of beryllium. JET continues to provide facilities well suited to test beryllium components. Site procedures are well established for its safe handling, and operational results demonstrate good control. The lessons learnt at JET will be applicable to the safety of larger scale fusion reactors.

K-26

FINAL DISPOSAL POSSIBILITIES OF RADIOACTIVE WASTE COMPONENTS FROM ITER

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The disposal possibilities of radioactive waste components from the International Thermonuclear Experimental Reactor, ITER, as described in volume V of the Generic Site Safety Report, GSSR, were studied. Components or zones above clearance levels in IAEA TECDOC 855 after 100 years of decay were considered as waste requiring final disposal in a suitable repository.

The possibilities for final disposal will depend on the regulations in the host country. In order to give a comprehension of the ITER waste characteristics for final disposal the acceptability of ITER waste in some European repositories in operation or in planning were studied.

All waste from ITER should be easily accepted in deep geologic option facilities in Germany, Sweden, France and Italy. However, the ITER waste should also to a large extent be accepted in less sophisticated repositories in these countries. Konrad in Germany would accept almost all ITER waste, leaving only the first wall components to Gorleben disposal.

SFR in Sweden would accept the vacuum vessel and waste components outside the vacuum vessel. From the in-vessel components only the inboard and outboard blanket backplate and its nearest blanket zones are within the limits for SFR. The more central part of the blanket and the divertors have too high concentrations of Ni-59 and C-14.

CSA in France and a repository for low level waste in Italy would accept the rear parts of the vacuum vessel and components outside the vacuum vessel. The first wall blanket and the divertor zones are above the acceptance limits. Mo-93 is the most critical nuclide for CSA, Ni-63 and Ni-59 for the Italian low-level waste.

The policy for handling very low-level waste differs between countries. In France studies are under way for establishing special surface disposal facilities for this type of waste. In Sweden landfill sites are available at nuclear power plants and at the Studsvik research station for the disposal of very low-level waste. Such facilities, if available should be possible to use also for very low-level fusion waste in order to reduce the volume needed in repositories designed for more barrier demanding radioactive waste.

K-27

EXTERNAL COSTS OF MATERIAL RECYCLING STRATEGIES FOR FUSION POWER PLANTS

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This study was performed within the framework of the Socio-Economic Research on Fusion project (SERF3), which is jointly conducted by Euratom and the fusion associations. Assessments of monetarized external impacts of the fusion fuel cycle were performed previously (SERF1 and SERF2).

In this study, several fusion power plant designs (SEAFP Models 1–6) were compared focusing on a part of the plant's life cycle: environmental impact of recycling the materials. The models differ mainly by the type materials used for core reactor components and type of cooling medium. Recycling is considered for materials replaced during normal operation, as well as materials from decommissioning of the plant. Several recycling schemes for activated parts have been suggested, one of which proposes using recycled fusion components when building new fusion reactors.

Environmental impact was assessed and expressed as external costs normalised with the total electrical energy output during plants operation. This facilitates comparison with other options for electricity generation. The methodology used for this study was developed by the Commission of the European Union within the frame of the "ExternE" project. It is a bottom-up methodology, with a marginal and site specific approach. Quantification of impacts is achieved through the damage function or impact pathway approach that follows the sequence of events linking a burden to an impact and subsequent monetary valuation. This means that it involves siting a power plant, and calculating its contribution to the environmental and health situation locally, regionally and globally.

Different material streams were used for activated recyclable parts and "common" recyclable parts. The regulations regarding transports and handling of radioactive materials will govern how they are treated. Some components will be possible to recycle only after up to 100 years of cooling, during which it must be stored. Even after that, some parts may have to be taken to final repositories. This amount varies between designs.

The external costs of different scenarios for managing used fusion plant materials can be used as a contribution to the basis for decisions regarding waste from fusion plants.

K-28

MATERIALS OPTIMISATION FOR FUSION POWER PLANTS WASTE MANAGEMENT FROM NEUTRONICS AND ACTIVATION ASSESSMENT

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This paper describes a study, carried out for the EFDA (European Fusion Development Agreement) Technology Workprogramme 2000, Safety and Environment – Waste Management, dedicated to the decommissioning and waste management issues of a commercial fusion power station. It is focused on the optimisation of structural materials for ex-vessel components. The choice of such materials may have a variety of effects. A material less active in the long-term may have higher short-term decay heat, or have a poor shielding effect, and then increase the total waste volume. Thus the choice has to be a balance between all possible consequences. In particular, two SEAFP (Safety and Environmental Assessment of Fusion Power) Plant Models (PM2 water cooled, PM3 helium cooled) have been investigated. The optimisation of the reactor ex-vessel components composition was aimed to make possible their clearance, considering both disposal as non-active waste (NAW), and recycling outside the nuclear industry (NARM = Non-Active Recyclable Material). The two clearance options and related clearance limits and indexes have been defined.

The following steps have been performed to carry out this study. One-dimensional Sn radiation transport calculations have been performed; neutron and gamma flux distributions have been determined by means of the Bonami-Xsdrn Sn coupled n-g one-dimensional discrete ordinates transport calculation sequence from Scale 4.4a computer code system. The Vitamin-ENEA Master Library (174n-38g groups), based on ENDF/B-VI data, is used for transport calculation. The neutron flux spectra have been provided in Fispact format to perform the activation calculation with the Fispact-99 activation code. Activation results for the different material of the ex-vessel component have been used to calculate the clearance indices for both pathways NAW and NARM. The clearance levels for disposal or recycling have been taken from an IAEA proposal, applying safety factors, and from an E.U. Radiation Protection Recommendation (RP 89).

The results of this study applied to PM2 and PM3 are:

- all the PM2 ex-vessel components, including the VV itself, up to the concrete cryostat can be cleared with the exception of inboard winding pack, insulator, vessel wall and pipes;
- the same for PM3. However, the inboard B_4C neutron shield and all the VV wall and related pipes also cannot be cleared.

Possible optimisations proposed are related to the outboard vessel both for PM2 and PM3. They deal with impurity level control and/or use OPTSTAB, a reduced-activation austenitic high-Mn steel. The poor shielding capability offered by PM3 has been confirmed in this study and in a parallel dose rate calculation for both PM2 and PM3.

K-29

COMPREHENSIVE ACTIVATION CALCULATIONS OF REFERENCE MATERIALS

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Part of the European fusion technology programme on materials has involved the specification and manufacture of reference materials, such as Eurofer-97. It is important to have activation calculations of such materials in a variety of positions in conceptual fusion devices as an aid to materials experts and designers and in formulating R&D programmes. This paper describes systematic and comprehensive neutronics and activation calculations for materials in two near-term devices based on the European blanket concepts, water-cooled lithium lead and helium-cooled pebble bed.

MCNP calculations of the devices have been carried out and neutron spectra for the first wall, blanket and shield produced. The spectra are used by FISPACT-2001 for activation calculations of a variety of typical materials for each location using EAF-2001 data libraries and considering 5 full power years of operation (25 years for the shield). For each material three variants are considered: the pure material with no impurities, the best estimate of impurities and the best specification with aggressive reduction of impurities.

Activity, □-dose rate, heat production, ingestion and inhalation dose and clearance index are calculated. The effect of statistical variations in Monte-Carlo calculations and the uncertainty data held in EAF-2001 are assessed. As a way of summarising each material variant a set of 'importance diagrams' are constructed. These show the activation properties for irradiation by mono-energetic neutrons in the range $1\times10^{-5} – 2\times10^{7}$ eV and for decay times of 1s – 1×10^{6} y. Another innovative way of presenting results is the use of a sensitivity profile. If the amount of a particular impurity is doubled then the percentage change in the response (e.g. dose rate) gives the sensitivity coefficient. The resulting curve shows clearly the importance of that impurity in determining the response of the material. Particular examples of these results are discussed in detail to bring out important points in the reference material specification.

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K-30

ANITA-IEAF: A CODE PACKAGE FOR PERFORMING FUSION MATERIAL TRANSMUTATION AND ACTIVATION ANALYSIS INDUCED BY INTERMEDIATE ENERGY NEUTRONS

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To study the irradiation effects on fusion materials some facilities have been proposed to produce accelerator-based neutron sources at sufficient intensity to test samples of candidate materials to be used in future fusion plants. In these facilities there is a considerable amount of neutrons produced with energy above 20 MeV. One of the most important effects of the neutron irradiation is the induced activation of the materials. Nuclear activation data above 20 MeV are still scarce and the existing available activation codes cannot treat the numerous and “exotic” reaction channels that open at these energies.

This paper presents the Anita-IEAF code package for the activation characterisation of materials exposed to neutrons with energies above 20 MeV. Its origins trace back to the Anita-2000 code (NEA-1638, RSICC CCC-606). Anita-IEAF is able to manage the many reaction channels for neutron energies up to 150 MeV. It computes the radioactive inventories of materials exposed to neutron irradiation, continuous or stepwise. It provides activity, isotopic nuclide density, decay heat, biological hazard, clearance index and gamma ray source spectra at shutdown and at different cooling times. The code package is provided with a complete database that includes neutron activation data library, decay, hazard and clearance data library, and gamma library. The Anita-IEAF neutron activation library was produced by processing the IEAF-2001 data activation files that have been recently released by FZK. It contains the neutron activation cross-sections for 679 nuclides in the 256 neutron energy group structure up to 150 MeV, in EAF format. That group structure includes the standard Vitamin-J 175 groups for energies below 20 MeV and 81 groups for the highest energies.

The paper presents also an application of the Anita-IEAF code package to the neutron exposure characterisation for the SS-316 liner and heat shield of the Test Cell area of the International Fusion Materials Irradiation Facility (IFMIF). The delayed gamma source evaluation for SS-316, needed for dose rate calculations at beam-off IFMIF phase for shielding analysis, is discussed too.

K-31

WATER COOLED AND GAS COOLED SEAFP FUSION PLANT MODELS COMPARISON BASED ON THE OUTSIDE-CRYOSTAT DOSE RATES

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Dose rate in operating accessible zones of a fusion power plant is a crucial point in calculating the Occupational Radiation Exposure ORE for maintenance staff. It also plays a significant role in radiation shielding design. In a SOFT-21 paper, the results of biological dose rate calculations for different locations in the vicinity of the concrete shield wall of the cryostat pit of the SEAFP-2 fusion plant were given. In the present paper, the assessment is focused on the Plant Model 3 design (low-activation martensitic steel for first wall and blanket, Li_4SiO_4 as breeder material, helium cooled). A one-dimensional cylindrical model representing the radial build at the plant mid-plane from the central solenoid up to the cryostat concrete wall was set-up. A total of 44 radial zones have been considered for the radiation transport assessment. Dose rates are evaluated both during plant normal running operation and after plasma shutdown before long term maintenance activity for blanket change out. Contribute due to neutrons and gamma-photons are considered through a 174n-38γ coupled Sn transport calculation. Radioactive decay of materials is included. Effects of streaming due to penetrations are also evaluated. The results are compared with the corresponding ones referred to the Plant Model 2 design (low-activation martensitic steel for first wall and blanket, $Li_{17}Pb_{83}$ as breeder/multiplier material, water cooled).

The neutron and gamma transport analysis has been performed with the Bonami-Nitawl-Xsdnr calculation sequence of the Scale 4.4a code system, using the Vitenea_E library, based on ENDF/B-VI cross-section data. The Anita-2000 code (with the FENDL/A-2 activation data library and the FENDL/D-2 decay data library) has been used for activation calculation to evaluate the gamma sources due to material radioactive decay. The neutron and gamma dose rate at different locations outside the cryostat concrete wall has been obtained with the Xsdose code using ANSI standard neutron and gamma flux-to-dose-rate factors

The dose rate values into pit cells result to be higher for Plant Model 3. In any case that area can be classified as Green zone (i.e. unlimited access for radiation workers) according to the ITER zone classification, both during plant normal running and 24 hours after plant shut down, for all the toroidal directions considered on the equatorial plane. One of the most relevant results of the study is that the effectiveness of the cryostat wall concrete is similar for the two plant models. This make possible, from the dose rate point of view, to set up the cryostat wall layout independently from the plant model that will be chosen for SEAFP design.

K-32

EXTENSION OF ITER WASTE ASSESSMENT

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The results, in terms of tons of radioactive materials and waste, reported in Volume V of the ITER Generic Site Safety Report (GSSR) were based on data available from the beginning of 2000 until spring 2001. However, some data of importance have been revised by ITER CTA since GSSR was issued as well as aspects of importance have been addressed and investigated further.

The ITER waste assessment has been (until March 2002) extended by the following issues:
- Revised mass of the Vacuum Vessel – Impact on radioactive materials and waste masses.
- A lower Cobalt content in ITER reference structural steel SS 316L(N)-IG - Impact on clearance index and remaining radioactive waste masses.
- Clearance index of the bioshield concrete – Impact of shielding by the TF coils.
- Rare earth and primordial radioactive elements as impurities in steel - Impact on clearance index. The topic is addressed in GSSR V and a supplementary study is underway

Revised mass of the Vacuum Vessel – Impact on radioactive materials and waste masses
The impact of reduced Vacuum Vessel mass is a decrease of waste mass between 680 and 2500 tons, depending on how well the waste can be separated from the material to which clearance can be applied.

A lower Cobalt content in the ITER reference structural steel SS 316L(N)-IG – Impact on clearance index and remaining radioactive waste masses.
Nickel is an alloying element whereas Cobalt is an impurity (Ref. specification: 0.05wt.%).The impact of lower Cobalt content, 0.01 and 0.00 wt.% in SS 316L(N)-IG, has been investigated. As expected, the result shows that the radioisotope Cobalt-60 in ITER steels has no significance whatsoever for the radiological parameters ‘specific radioactivity’, ‘clearance index’ and ‘contact dose rate’ beyond about 100 years after final shutdown.
The reduction of the Cobalt and Nickel concentrations in ITER steels cannot exclusively be discussed in terms of clearance. Rather, also ALARA, in particular with regards to dose rates as well as technical feasibility and cost have to be considered.

Clearance of the bioshield concrete – Impact of shielding by the TF coils.
The one-dimensional model used for activation calculation, which assumes toroidally continuous outer TF coil legs in front of the bioshield, does not take into account gaps between the TF-coils. Therefore, the analyses have been supplemented by an activation calculation where the TF-coils are completely disregarded.
The result shows that there exists neutron activation of the bioshield concrete. It occurs in the plasma-side front region of the concrete. In the interior of the concrete (about 1 m deep), there is virtually no activation. There the associated clearance index is about 0.1 (from natural Potassium 40) as reported in GSSR/V. The activation at the concrete front is not relevant from the clearance point of view because the potential starting time of the ITER waste time scale is expected to range from 10 to 100 years after shut-down due to the half life of the isotopes generated.

K-33

THE POSSIBLE ROLE OF FUSION IN THE INDIAN ENERGY SYSTEM OF THE FUTURE

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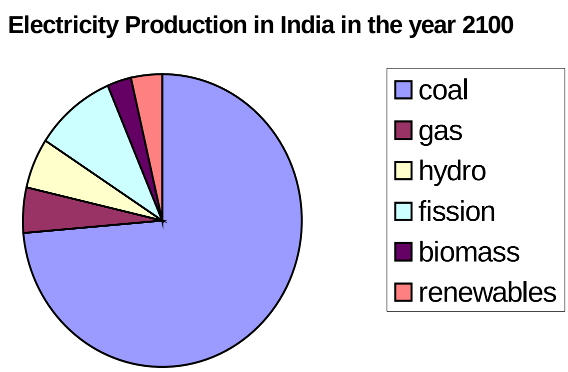
India and China together will cover roughly one third of the world population in the year 2100. With 1.6 billion people India will be the most populated country in the world. Sustained economic growth in the last decades makes it rather likely that both countries will become major economies in the course of the 21st century.

Economic growth and increase of energy demand is still coupled especially in developing economies. Therefor a strong increase in the primary energy demand is expected in India in the next hundred years. Demand for electricity is likely to increase by a factor of ten to more than 4000 TWh/a. The question arises: which energy sources are capable of supplying the energy and what is the possible impact on the greenhouse gas emissions. To develop a sound answer to these questions a model of the Indian energy economy was developed based on various established software tools like AIM/Enduse and MARKAL.



The results indicate that the answer to most of the energy problems in India is coal, if no special politic interventions are taken to steer the energy system into other directions. Coal is cheap in India and large resources are available.

The paper discusses the model developed and presents various results assuming different politics interventions, especially measures to mitigate greenhouse gas emissions. The possible role of nuclear fusion to supply electricity for India will get special attention. The paper is based on a study commonly conducted by the Indian Institute of Management in Ahmedabad, India, the Dutch energy research centre ECN and the Max-Planck-Institute for Plasmaphysics in Germany.

K-34

FEASIBILITY OF L / ILW FISSION WASTE REPOSITORY CONCEPTS FOR FUSION WASTE

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In this paper the feasibility of a L/ILW (low and intermediate level waste) fission waste repository, similar to those operating in Finland, for final disposal of fusion waste (both operational and decommissioning) is discussed. Based on the experience of the performance assessments for fission waste repositories, a preliminary assessment of the performance of natural barriers with respect to fusion waste is made. In this assessment those materials of fusion waste - relevant to MINERVA-W and H concepts - which dominate the long-term ingestion radiotoxicity are considered. The real waste package and layout configuration in a future repository is not addressed in this assessment and therefore the near-field model is a simple sorption model with an anticipated water-backfilling ratio, which is likely to overestimate the consequences. In the far-field model the dilution of groundwater, as well as sorption and matrix diffusion in the bedrock fractures are considered. Due to the experience of fission waste repository performance assessments, radiotoxicity concentrations in a postulated drinking water well are calculated, since it is in most cases the limiting scenario.

According to the calculations, during the first thousand years after package degradation, which was assumed to happen 1000 years after closing the repository, major contributors to radiotoxicity in the well (or lake or sea, if receiving the same water) are Ar-39 and Mo-93 (also Cl-36 and U-233 in some materials). Major contributors to the long-term (10 000–100 000a) radiotoxicity concentration in the well are plutonium isotopes Pu-239 and Pu-240 and Re-186m. Plutonium isotopes are generated from uranium existing in small impurity fractions in some fusion materials. Re-186m is an activation product of tungsten. Behaviour of plutonium and rhenium in the geo- and biosphere is a crucial issue in the long-term safety of final disposal.

Other crucial issues to be studied further, are beryllium's behaviour and its chemical toxicity. Also C-14 in fusion waste is of more concern than it has been in L/ILW or spent fuel from fission plants. Uncertainties related to the chemical form and thus release of C-14 from fusion waste should be studied further

Due to a preliminary nature of this work no definite conclusions about the feasibility of the considered repository can be drawn. The study however implies that the repository concept might be feasible for most of the fusion waste, and the open questions probably can be solved by waste conditioning and further studies on behaviour of radionuclides and management of related uncertainties. The work identifies some special issues to be considered e.g. in material choices and impurity level definitions and on the other hand in waste conditioning and packaging.

K-35

ELECTROMAGNETIC STRAY FIELDS OF THE POWER SUPPLY SYSTEMS OF TEXTOR IN REGARD OF THE EUROPEAN COMMISSION COUNCIL RECOMMENDATION (1999/519/EC)

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Different types of high current power supplies (up to 82kA) feeding the TEXTOR magnetic coils for toroidal and poloidal field, and the auxiliary heating systems (up to 70kV) produce electromagnetic stray field. Furthermore the tokamak TEXTOR operates medium frequency power suppliers for the Dynamic Ergodic Divertor with currents up to 15kA at frequencies up to 10 kHz which are even more critical. The TEXTOR power supply systems are installed at distances up to 200 m from the tokamak itself. Electromagnetic stray field from the supplies and their feeder lines are unavoidable in places with sensitive electronic equipment and with public access during operation. Electronic equipment has to work reliably if electromagnetic stray field occurs. Furthermore, the European Commission has adopted a Council Recommendation to limit the exposure of the general public to electromagnetic fields (0 Hz – 300 GHz) in July 1999. The new protection limits for electromagnetic fields in public areas are much lower than the values which had been valid in the past. Therefore fields at the TEXTOR site have to be known to be sure that protection limits are not exceeded. Otherwise, protective measures have to be taken against exposure to the fields.

The electromagnetic stray field was measured in all relevant areas during different kinds of operation. The DC field was registered by a 3-axis Hall probe measuring system. The AC measurements were done with commercial measuring instruments for magnetic fields with a frequency range from 3 Hz to 30 kH<.

In this paper the different sources of magnetic stray fields are described. The results of the measurements are given and discussed. For areas, where the recommended limits are exceeded, suggestions for the modification of the equipment and other measures are given.

K-36

DESIGN OF RADIATION SHIELD AND SAFETY SYSTEM FOR HT-7U TOKAMAK DEVICE

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HT-7U is a fusion experimental device being designed and built in the Institute of Plasma Physics, Chinese Academy of Sciences, which will generate neutron rate of the order 10^{15} per second. The design of radiation shield and safety system based on CAN for the device has been presented in this contribution.

An inner shield, which is made of neutron absorbing materials such as borated water, and an outer shield, which is the concrete walls and ceiling of the shield building of the device, are considered in the shield design. Calculations and analyses have been done to optimize the shield design by using one-dimensional and two-dimensional transport codes ANISN and DOT3.5. ANISN code is mainly to estimate the self-shielding effect of the device components, while the code DOT3.5 is used to optimize the shield thickness. In the meantime, detailed 3D MCNP/4B (Monte Carlo Neutron/Photon transport calculation code) model for the neutron spectrum and activation calculation by the inventory code FISPACT are done to estimate the dose rate to workers after one pulse operation of the device. The latest version of data library i.e. the Fusion Evaluated Nuclear Data Library FENDL-2.0 released by IAEA in 1999 is used both for transport calculations and activation calculations.

On the other hand, a radiation protection and control computer system based on CAN Field Bus has been designed in order to protect staff and publics during the operation of HT-7U device. It includes real time radiation monitoring, messages display, operation control of HT-7U device, disposing of emergency events and gateway management. The last one is achieved by monitoring and controlling RF-IC cards of the doorway, turnstile, emergency exit, shielding door, safety lock, emergent shutdown buttons and so on.

It is concluded that the plasma discharge time can reach 10^4 seconds per year if 1.5 meter thick side concrete wall and 1.0 meter thick top concrete ceiling are used as the outer shield, the dose rates to site workers and publics are much lower than the levels given in the law for activity protection. The maintenance and replacement can only be done when the shutdown time is greater than 7 hours for the discharge time of 10^3 seconds. On the other hand, radiation monitoring system, gateway management system and other radiation protection facilities are integrated into one uniform system, this makes it easier to exchange messages among subsystems and more propitious to bring into play integrative function of the system. The radiation safety system based on CAN BUS has the advantages of multi-master protocol, real-time capability, error correction, long communication distance and high noise immunity.

K-37

NUMERICAL EVALUATION OF EXPERIMENTAL MODELS TO INVESTIGATE THE DYNAMIC BEHAVIOR OF THE ITER TOKAMAK ASSEMBLY

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The tokamak assembly of the International Thermonuclear Experimental Reactor (ITER) consists mainly of a building, a cryostat, a coil system, and a vacuum vessel (VV). Those components with different operating temperatures are connected to each other. It is anticipated that the natural frequency of the tokamak assembly is about 2–3 Hz level, which is very low and unusual for large-sized structures. Such low natural frequencies are mainly due to the flexible support structures between the coil system and the cryostat and between the VV and the coil system. The low natural frequencies will affect the whole tokamak design, particularly seismic design. Therefore, it is essential to establish the assessment methods for the dynamic behavior of the tokamak, including the comparison between the analytical and experimental investigations.

The dynamic behavior of the tokamak assembly has been investigated [1]. Three experimental models have been considered to validate the numerical analysis methods for the dynamic events, mainly seismic events. A 1/8-scaled tokamak model, which is based on the 1998 ITER design, is under construction for vibration tests. The scaling ratio was chosen in a way that the stresses induced on the model are equivalent to those of ITER. Due to subsequent ITER design changes, this model no longer precisely represents the current ITER design. Therefore, the model cannot induce the stresses simulating the current ITER. However, it will be used to examine the vibration and response characteristics for the validation of the analysis methods.

Non-linear vibration characteristics, such as damping, can only be identified by a full-scale model. Therefore, a full-scale gravity support structure for the coil system has been designed and will be tested. In addition, for the sub-scaled tokamak model, the VV is assumed to be a rigid structure. This assumption is to be verified using a 1/20-scaled model.

The above experimental models and their testing conditions have analytically and numerically evaluated. For example, both the static and dynamic spring constants obtained by static analysis and eigen-value analysis, respectively, were evaluated to be in good agreement.

This report presents the investigation results.

[1] N. Takeda et al. Presented at 18th IAEA Fusion Energy Conference, October 2000, Sorrento, Italy.

K-38

EXPERIMENTAL STUDY OF STEAM CHEMICAL REACTIVITY WITH BERYLLIUM POWDER ON HOT SURFACE INSIDE THE GROOVES

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The ingress of coolant event (ICE) in the ITER vacuum vessel (VV) might cause a substantial amount of hydrogen to generate in the VV, since water vapor oxidizes heated in-vessel components, and, hence, might bring about an explosion. The most dangerous are the erosion products of the first wall in the form of beryllium dust on hot surfaces, in particular, on the divertor. The goal of the experiments was to compare the reactivity of the beryllium dust in the water vapor atmosphere depending on the dust location geometry (on the surface, in slots between armoring tiles) and temperature.

On our experimental facility the temperature regime of the ITER vacuum vessel was simulated followed by simulation of emergency leak. The concentrations of gaseous reagents (water, hydrogen) were measured by the spectroscopic method with a high time resolution, i.e. up to 10 measurements per second. Beryllium powder CBP-56 with a specific surface of 0.38 m^2/kg was provided for investigation by the Bochvar Institute (Moscow). The initial temperature of the beryllium dust varied in the range of 500–900°C. The investigations showed that at a beryllium dust temperature below 700°C the reactivity depends but slightly on the dust location geometry; at a temperature of 800°C the initial reactivity differed by several times; at 900°C the initial reactivity differed by a factor of 30. The results are presented and discussed.

The results of the experiments indicate that at temperatures below 700°C the main diffusion barrier keeping beryllium from vapor is the beryllium oxide layer growing as a result of oxidation on the surface of each grain of the initial powder. As the dust temperature increases, the dust location geometry becomes more important, i.e. the determining factor is the time of vapor diffusion from the sample surface to the grain boundary. The obtained results are necessary when calculating different emergency scenarios. In particular, they will help to define how the hydrogen pressure in the ITER VV will grow in case of an accident.

K-39

IN-VESSEL COMPONENT DESIGNS FOR A SELF-COOLED LITHIUM-LEAD FUSION REACTOR

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Within the framework of the European Power Plant Conceptual Studies (PPCS), launched in January 2000, the most advanced reactor model is based on a Self-Cooled Lithium-Lead (SCLL) blanket concept. This reactor is associated with the largest attractiveness and at the same time with the largest development risk. It is based on the use of SiC/SiC structures which allow high coolant temperature, and show very low short term activation and afterheat levels. Associated with the use of Pb-17Li (Li enriched at 90% in ^{6}Li) as breeder, coolant, neutron multiplier and tritium carrier, this system allow to achieve high plant efficiency and has the potential for reactor passive safety.

High efficiency and advanced plasma physics applied to a 1500 MWe reactor have lead to a new set of attractive reactor parameters such as a 6 m major radius, low additional heating and an heat flux on the divertor of 5 MW/m^2. This paper presents these new parameters and describes the major design features of the associated in-vessel components and the related performed thermo-mechanical, neutronics, thermo-hydraulic and MHD analyses and results. The new SCLL blanket design is fully described. It is based on the most attractive features of previous blanket designs such as TAURO and ARIES-AT. It is essentially formed by two 8 m high concentric boxes where the Pb-17Li flow in at 600°C and high velocity (above 4 m/s) in the outer box and flow out at low velocity in the inner box with an outlet temperature of 1100°C. The maximum SiC/SiC temperature is maintained below 1000°C. Despite the high Pb-17Li velocity required, MHD pressure drops are acceptable because of the relatively low electrical conductivity of the SiC/SiC structures. The shielding is a combination of CaH and TiH enclosed in C/SiC structures and cooled either with Pb-17Li or He. The vacuum vessel is an Helium-cooled borated steel structure.

The newly developed divertor concept is cooled by Pb-17Li with an inlet temperature of 600°C and an outer temperature of about 860°C. It uses SiC/SiC structures and W tiles. The reactor has 16 TF coils which forms 16 VV sectors. In the poloidal direction, the blanket is divided in three parts: outboard, inboard and topboard blankets. Toroidally, the outboard blanket is formed by 48 segments while top and inboard blankets by 32 segments. The maintenance scheme foresees empty blanket segments replacement from the top, vertical ports while the divertor is replaced from the bottom, horizontal ports. Pb-17Li feeding pipes and drain system enter the vessel from the bottom. The remote connection and disconnection of the blanket and divertor pipes before removal represent a major design challenge.

A new proposal of installing thin breeding elements behind the divertor target plates for increasing the tritium production is discussed and evaluated. The technological extrapolations assumed in this study provide an indication of the R&D to be launched for addressing the most critical issues. From the structural material point of view, the most severe issue for the SiC/SiC is the required improvement of the thermal conductivity at end-of-life conditions which has to be increased, with respect to today's value, by at least a factor five.

K-40

INVESTIGATION OF MITIGATION MEASURES FOR THE REACTION OF BERYLLIUM PEBBLES IN STEAM

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The reference breeding blanket design for ITER foresees the use of beryllium as a neutron multiplier in the form of a pebble bed. The blanket will be water-cooled. Therefore, an in-blanket loss of coolant accident (LOCA) entails the highly reaction between water/steam and beryllium in the temperature range up to 500°C. This reaction produces heat and hydrogen gas and hence presents a safety hazard if it becomes self-sustaining or if the hydrogen concentration surpasses the lower explosion limit. Therefore, and in order to contribute to the intrinsic safety of the breeding blanket design, we investigated measures to mitigate the chemical reaction between beryllium pebbles and water/steam.

In the temperature range of interest for the accidental scenarios in the breeding blanket (300–500°C), the rate-determining step of the beryllium/steam reaction is the diffusion of Be^{++} through the oxide layer, followed by reaction with oxygen at the oxide/steam interface. The corresponding kinetics are parabolic. In view of this oxidation mechanism, mitigation strategies should in the first place hamper the diffusion of Be^{++}. We investigated two possible measures to achieve this: oxide doping, and pre-oxidising. To investigate the possibilities of doping the beryllium metal to influence the chemical reactivity, ITN Lissabon implanted calcium and aluminium ions directly into the oxide layer of beryllium samples. The purpose is to decrease the diffusion coefficient of Be^{++} in the modified oxide layer. The chemical reactivity of the samples at 500°C was measured at SCK-CEN with coupled thermo-gravimetry/mass spectrometry and the results were validated by direct measurement of the oxide thickness by ITN before and after the chemical reactivity tests. The same experiments were performed with reference non-doped beryllium. The second candidate mitigation method was pre-oxidising the beryllium pebbles in air prior to the experiments in steam, in order to increase the initial thickness of the oxide film and hence lower the concentration gradient and diffusion rate of Be^{++}. In our tests, we pre-oxidised beryllium samples at 300°C in air during twelve hours before exposure to steam at 500°C during six hours. From the TG and mass spectrometry results, we obtained the parabolic rate constants by curve-fitting. The current paper first discusses the mechanistic aspects of the mitigation strategies and presents our first experimental results.

K-41

PUBLIC INFORMATION ACTIVITIES ON FUSION IN THE EUROPEAN UNION

Presented by F.Casci[1], with the support of the PI Staff in the Associations

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One of the EFDA tasks is to take the lead in all Public Information activities at a European level, to provide decision makers, the media and the general public with the necessary information about all fusion activities carried out in Europe.

The main objective is to present the results of the work carried out by the European fusion community with the aim to get fusion recognised as a key player in the debate over long term energy provision and climate change. This objective can only be achieved by spreading the information outside the boundary of the Fusion community.

The EFDA Public Information activities are monitored by the Public Information Committee, which is itself appointed by the EFDA Steering Committee to provide recommendations to the EFDA Leader on policy issues. Public Information activities in the EU are carried out both at a national level, in each Association, and in EFDA. EFDA also takes care of the co-ordination of these activities at a European level. The co-ordination between EFDA, the Associations and the Commission is of paramount importance for a successful result in this area.

This paper will present an overview of activities, at both an Association and EFDA level. Among the topics to be presented is the general philosophy of the approach and the tools in order to achieve the given objective. Topics such as the participation to conferences, brochures, films, web activities, audio-visual support and the itinerant exhibition “Fusion Expo” will be included in the paper.

The EFDA outreach activities in the field of scientific education will also be presented. As a partner of CERN, ESA, ESO, EMBL, ESRF and ILL in EIROFORUM, an intergovernmental organisation of European leading scientific institutions, EFDA is also active in promoting fundamental science through special programmes for teachers and students.

K-42

POWER PLANT CONCEPTUAL STUDY – WCLL CONCEPT

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A Power Plant Conceptual Study has been launched in the framework of the EU fusion programme the objective of which is to demonstrate the credibility of fusion power plant design and the claims for safety and environmental advantages and for economic viability of fusion power. A generic set of requirements, addressing in particular safety, operation and economic aspects, has been set out with inputs from industry and from utilities. Four reactor models have been identified for a complete evaluation. The model which is presented in this paper is based on little extrapolation on both physics and technology, using a water-cooled divertor based on ITER technology and associated to the Water Cooled Lithium Lead (WCLL) blanket. The WCLL blanket uses low activation ferritic martensitic steel EUROFER as structural material, pressurized water as coolant and lithium-lead as breeder and neutron multiplier. For maintenance, a segmentation of the blanket in large modules and a segmentation of the divertor in cassettes are considered.

The design of the blanket and divertor modules is presented. The low level of extrapolation on physics has led to consider a maximum heat load of 15 MW/m2 for the divertor in order to limit the major radius. Copper alloy (CuCrZr) has been chosen as structural material due to its high thermal conductivity and its better fracture toughness, and tungsten alloy as armour material. It is shown that the proposed divertor structure withstands the thermo-mechanical loads. The blanket modules are designed to meet the requirements of both the maintenance scheme and neutronics (tritium breeding ratio). Other constraints are the maximum temperature of the coolant (margin to saturation), the maximum temperature of the steel (avoid significant creep conditions), the minimum temperature of the steel (prevent irradiation embrittlement) and the maximum temperature of the interface between steel and lithium-lead (prevent corrosion). It is shown that thermal and mechanical constraints are satisfied with important margins.

The heat transport system is described. The blanket and the divertor have separate cooling loops. The number of cooling loops is determined so as to limit the coolant inventory per loop for safety reasons but also to limit the size of the balance of plant. Each blanket loop has a steam generator. The divertor power contributes to the electricity generation. The heat is exchanged between the divertor loops and the secondary circuits of the blanket heat transport system through preheaters. The power conversion cycle of he WCLL concept is based on current PWR technology. As it works at PWR conditions its overall thermal efficiency is similar to the PWR one.

The WCLL concept has limited extrapolations. However, some R&D needs are defined, mainly concerning the properties of the materials.

K-43

PRELIMINARY ANALYSIS OF WASTE RECYCLING SCENARIOS FOR FUTURE FUSION POWER PLANTS

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This paper summarises the work carried out on the task "Waste Generic Issues" of the Power Plant Conceptual Study (PPCS) and addresses the policy issues associated with fusion operational waste – namely the in-vessel components that need to be replaced on a regular basis. The issue is similar to that of the fission power industry, where the used fuel is the key operational waste of concern. Therefore, there is a valid analogy between in-vessel components and used fission reactor fuel, even if there are significant differences in the type of waste and radiotoxicity. The fission power industry has followed two basic strategies with respect to used fuel disposal. The Americans have chosen the once-through fuel cycle, while Europe, in general, and Japan have chosen a closed-fuel cycle, which includes reprocessing of the used fuel. These two strategies are also available to the future fusion power industry. The key question is to determine the technical and economic feasibility of used, in-vessel component refurbishment and reprocessing. The objective of this study was to develop and assess various scenarios for dealing with the interim storage, refurbishment/reprocessing, and final disposal of waste arising from used in-vessel components. The key criterion, in evaluating the various scenarios, was the environmental acceptability of the fusion power plant. More specifically, the aim was to identify scenarios that would eliminate the concerns that became evident in some European countries, in connection with the shipment of radioactive materials to and from centralised fuel reprocessing facilities. Such problems were brought to bear on the fission industry, even though fuel reprocessing significantly reduced the radioactive waste burden to future generations, relative to the open fuel cycle strategy.

In considering a future fusion power industry, it would be advisable to avoid such problems, by utilising reactor materials with low neutron-induced activation. Such materials would have a significantly lower radio-toxicity, relative to fission fuel, but would not eliminate the need for long-term disposal. In addition to the activation and contamination (tritium), the physical size and mass of in-vessel components would make shipments from the power plant problematic. Therefore, it is necessary to consider strategies and approaches that reduce, or minimise, the need for off-site transport of radioactive materials. Seven scenarios have been developed and studied to identify the factors that are important in the development of a suitable fusion-power waste management strategy. The scenarios studied range from doing very little refurbishment/reprocessing to doing the maximum refurbishing/reprocessing practical, both on-site and off-site. The following conclusions have been drawn:

1. Future fusion power plants should be constructed as multi-unit plants with adjacent in-vessel component refurbishing facilities.
2. Tritium recovery from blanket modules is expected to reduce the cost of in-vessel component refurbishment;
3. Reprocessing the breeder and neutron multiplier material may not be economical unless the reprocessing unit cost will be significantly lower than a few hundred Euro/kg;
4. Without IVC refurbishing, the operational IVC waste volume will far exceed the decommissioning waste volume.

K-44

STUDY OF ENERGY FLUX ON THE TARGET SURFACE AT PULSED PLASMA TREATMENT OF MATERIALS

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The treatment of the structural materials by the high heat pulsed plasma streams is worth-while direction of plasma technology [1]. One of the most important effect of such treatment – the amorphisation of the metal surface layer – can be achieved at the high degree of both melting and cooling ($\sim 10^6$ K/s). The well-founded choice of the modes of such plasma irradiation have to be based on the knowledge of energy flux on the target surface with consideration of the effects of the plasma shielding layer (SL).

The significant experience in the study of high heat plasma-material interaction in disruption simulation experiments has allowed to solve this problem. The modes of operation with pulse duration t_p = 0.09 ms, irradiation power P_{irr} = 20–100 GW/m^2 and magnetic field B = 0–3 T were used on the VIKA facility [2]. The visible radiation (400–700 nm) energy flux on the target surface and absorbed energy were studied.

The measured values of the specific energy of visible radiation from SL on the target surface can be characterized by the level Q_R ○ 50 kJ/m^2 for P_{irr} ~20 GW/m^2 and the absence of the magnetic field and increases up to Q_R ~300 kJ/m^2 at max. irradiation power and B = 3 T.

The total energy flux on the targets surface was identified with the measured values of the specific energy absorbed by the samples during irradiation. The linear growth of the absorbed energy with the rise of irradiation power was observed. The level of absorbed energy Q_{abs}~100 kJ/m^2 sufficient for the melting of the surface layer of most structural metals can be achieved at irradiation power flux P_{irr} ~30 GW/m^2. The values of absorbed energy depends feebly on the material and magnetic field values.

The modes of operation for pulse plasma treatment of metals are proposed and discussed.

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M-1

SPIN-OFF AND COOPERATIVE EFFORT INVOLVED IN FUSION MAGNETIC RESEARCH IN EURATOM-CEA ASSOCIATION

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The strong and extensive industrial spin-offs developed by the Fusion Magnetic Research for the last forty years in Euratom-CEA association have been induced by a continuous process of scientific, technology and managerial exchange between the fusion scientist and manufacture engineer knowledges. These growing knowledges and the associated innovative applications can be particularly pointed out in three major contributions.

The first is the large development produced in the frame of the generic technological programme of the Fusion in term of R&D in the nuclear safety, the remote handling operation, the low activation steel or hot isostatic pressure assembling which gave industrial spin-offs supported by a 18 M€ average expense a year.

The second is supplied by the construction of Tore Supra and especially in three specific domains: i) the cryomagnetism technology and the fabrication of the superconducting coils of Tore Supra* which has given rise to direct applications to other large scientific devices using superconductors (CERN, DESY...) and also to medical instrumentation (MRI), ii) the high heat flux components initiated by the plasma facing components developed for the Toroidal Pump Limiter of CIEL/Tore Supra**, these CfC (Carbon fiber Carbon) components linked to copper by a special bonding handles up to 10 MW/m² in steady state operation: aerospace and military applications are mainly concerned by this process, iii) the large high power high frequency generators*** (klystrons at 3.7 GHz and gyrotrons at 118 GHz in the 0.5–1 MW range) which can be used in other large scientific devices as CERN or other Fusion machines.
The third is given by the recent assessment of the Cadarache site for ITER and the impacts of the construction of Tore Supra in terms of improvement of technological knowledges, spin-offs and complex system management appreciated by companies implied by the construction of Tore Supra: The impacts are more than 3 times higher for these companies than for the average of the French SME’s (Small and Medium Entreprise) and this effect is especially important for the specific technologies as cryogeny, superconducters, and high heat flux components.

At least, an important contribution of the cooperative effort is also given by the different test-beds developed in collaboration with the Association, which allows acceptance tests (destructive or non destructive control of materials as the electron beam facility: FE-200 or thermography test-bed) and high frequency qualification of HF components or windows and offer large test facilities for specific industrial applications.

M-2

NONDESTRUCTIVE TESTING DURING TFCI MANUFACTURING AND FOR MULTI-LAYERED ITER COMPONENTS

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The toroidal field coil insert (TFCI) was developed and manufactured by the NIIEFA jointly with Russian institutions. One of the objectives of the TFCI manufacturing was to develop and to evaluate the post-operation quality control system.

The paper presents the particular results on detection and estimation of real defects in materials and welds of the titanium and incalloy jackets for superconductors. On the basis of the results of nondestructive testing of incalloy tubes intended for superconductor jacket it was decided to use titanium tubes for TFCI jacket. The results of nondestructive testing of joints of dissimilar materials of the TFCI terminations are presented.

The nondestructive testing of multi-layered structures joined by different joining technologies is of considerable applied interest. It is shown that the capabilities of ultrasound methods by amplitude signs are limited, as applied to detection of small-opening defects in materials and on the boundaries of the W-Cu-CuCrZr-SS and Be-CuCrZr-SS components.

The paper presents the theoretical calculations of the models of acoustical paths of multi-layered elements performed for optimization of sizes of defects which are assumed to be introduced into test models. Based on the results of the theoretical and experimental researches of small samples of multi-layered elements with artificial defects and «introduced» real defects the recommendations were devised for selection of the defect sizes corresponding to the NDT capabilities and the methods of «introduction» of defects into large-scale mock-ups intended for tests under design loads.

The analysis of the obtained results allows the conclusion to be made on the urgency of developments of effective NDT systems for the ITER Project.

M-3

FRIDAY, 13th September

Session 12: Safety, Environment, Power Plant Studies

Chairman: I. Cook

Session 13: ITER Siting

Chairman: S. Païdassi

DEMO AND MEDIUM-TERM FUTURE

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The aim of the European fusion program is to make available, as early as possible, fusion power as a source of electric energy. Both the strategic considerations of an EU Fusion Fast Track Experts Group, as well as the bottom-up studies within the European Power Plant Conceptual Design Studies concluded, that this could be best achieved by making DEMO as closely identical to a first generation commercial power plant. Already at a significantly earlier date, the fusion program should provide conclusive information on the practicability of fusion power production, and on its safety, environmental and economic aspects. Both targets can be achieved by an R&D program centered around the integrated fusion physics and technology studies in the ITER device, and on a material development program, including a neutron irradiation facility with a fusion relevant spectrum. Due to the associated long lead times, also the study of certain technologies which can possibly come to fruition only in a second generation of power plants should be initiated now. The presentation summarizes the physics and technology developments needed for DEMO/first generation power plant, and the most profitable advances that could be implemented in later power plants to further improve efficiency/economics and waste management. The latter include, in particular, the development of low activation materials that can operate at higher temperature with respect to EUROFER, and also the attainment of more ambitious physics performance targets.

The development path to be outlined is based on the conventional tokamak reactor line, although all of the technological developments would be equally applicable to a stellarator, or a spherical tokamak power plant. Assuming substantial, but plausible, advances in our theoretical understanding of magnetic confinement, results of the now nascent generation of stellarator and spherical tokamak devices could be combined with those of ITER to possibly proceed even to a DEMO without any additional, intermediary DT experiment.

IN-14

SOCIO-ECONOMIC STUDY OF FUSION ENERGY AT THE JAPAN ATOMIC ENERGY RESEARCH INSTITUTE

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Recently, various aspects of fusion energy are of worldwide interests in the context of future global energy mix. One of the major activities is organized under the framework of International Energy Agency multi-lateral collaboration. In Japan, a participant of this collaboration, a number of assessments of broad aspects have been performed to reveal the characteristics of fusion energy as a potential energy source in the future. The present paper intends to summarize the socio-economic study of fusion performed by JAERI and its collaborators in Japan, where steady-state tokamaks have been designed as a promising candidate for commercial plants.

In the recent fusion plant design in Japan, some novel concepts were introduced without assuming exotic technologies. For better acceptance in the future market, economy is pursued by using supercritical water heat transfer medium for the first generation, and high temperature gas for future. Not only efficient power generation, application to fuel production process such as hydrogen by utilizing heat not limited by the Carnot efficiency is investigated. Concept to initiate operation without initial tritium loading liberates fusion from fuel supply constraints, particularly in developing countries where major energy market growth is anticipated. Improved shielding design drastically decreased rad-waste.

Energy scenarios study in Japan estimates the contribution of fusion in the latter half of the century under the global environment constraints. In some scenarios fusion is anticipated to acquire 20–30% world market share while releasing no greenhouse effect gases. Criteria for fusion to be successful in the market was analysed, and the economic value of fusion development was estimated based on this economy model.

Particular emphasis on the socio-economic aspects of fusion is now focused on the Externality studies in Japan. Based on the successful activities of Extern-E project in the Europe, impacts of fusion energy on the environment and society is now investigated. Life cycle analysis of fusion revealed its superior feature on CO_2 emission over renewables. Normal tritium release and its effects in the environment is quantitatively studied. Investigation on social impact such as energy security or proliferation resistivity has started.

The findings in this recent socio-economic consideration on fusion energy is being reflected in the ongoing plant design studies. Based on the anticipated achievement of ITER, both in plasma physics and technology, possible power generating fusion plant is expected to be designed and constructed in late 2020s to early 30s, incorporating modest extrapolation and realistic improvement of technology, to maximize attractiveness of fusion identified in this study.

KO-17

WASTE MANAGEMENT WITHIN THE FRAMEWORK OF ITER IN CADARACHE

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ITER will produce wastes both from operational activities (component replacements and technological wastes) as well as the decommissioning phase (dismantling of the machine and associated technological wastes).

The waste management described in the ITER documentation is based on the clearance possibility for materials, using the IAEA recommendations. Since Cadarache (France) is a potential European site for ITER, and since clearance is not accepted in France, it has been necessary to review all the wastes and to determine which disposal is possible as a function of the French regulation criteria.

The classification has been performed as a function of the four categories of wastes currently used in France (depending on mean activity and nuclides half life) : TFA wastes (Très Faiblement Actifs), type A wastes for which the repository-type currently used is the final surface disposal of the Centre de Stockage de l'Aube, type B wastes and type C wastes (no type C waste in ITER case).

Different repositories can be considered for the repartition of waste packages :

- Industrial dump and specific industrial waste disposal,
- On site buffer interim storage waiting for a definitive repository,
- TFA dedicated repository,
- Shallow land final disposal for type A wastes,
- Long term interim storage : Its duration is not determined for the moment but should be less than 100 years. This is a pertinent solution when radioactive decay allows waste declassification from type B to type A or TFA repository, or for wastes needing deep storage (if not industrially available),
- Deep final disposal : only for wastes that can not be stored in shallow land disposal for type A wastes and can not be declassified.

After a first analysis, all the by-products of the ITER machine appear to be manageable.
Mixed wastes (chemical and radiological wastes such as beryllium irradiated wastes) are a first key issue. For these wastes, the maximum allowance performances, related to safety constraints, will be put on the packaging that has to be carefully designed and qualified.
The technical processes used during the operation and the de-activation periods, are a second key issue since the classification of waste packages will depend on guarantees (mainly demonstrations on contamination, beryllium rate) provided by the process.

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CADARACHE, A EUROPEAN SITE FOR ITER STATUS OF SITE STUDIES

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At CCE-FU meeting in July 2000, the French delegation proposed to study Cadarache as a possible site to host ITER. A European ITER Site Studies Group (EISS Group) has thus been created to study all technical, safety, licensing and socioeconomic aspects of this site.

This group, which works in the EFDA framework (European Fusion Development Agreement), has:

- thoroughly examined all technical aspects, taking as a reference the ITER Site Requirements and Design Assumptions,
- launched the French licensing procedure, with the writing and delivery to the Safety Authorities of the so-called “Dossier d’Options de Sûreté”, first step of this procedure,
- investigated the socioeconomic impact and socio-cultural environment around Cadarache,
- estimated the resulting cost both of Cadarache adaptation and ITER adaptation to the site.

The conclusion of this first phase of studies is that Cadarache fulfils all ITER requirements and is thus perfectly suited to host ITER, by providing in addition the support of a major nuclear site also hosting a fusion team working on Tore Supra, JET and ITER. The French minister of research thus offered to the Commission the possibility to propose Cadarache on behalf of Europe, which was done on June 4th, 2002 during the fourth negotiation meeting.

To optimise the overall planning, site studies are now construction-oriented, with:

- the preparation, in strong relationship with the ITER International Team, of the Preliminary Safety Report, the second safety document in the French regulatory process,
- detailed investigations of the site layout and site preparation,
- off-site studies (water and electricity supply, road adaptation),
- socioeconomic calculations, socio-cultural preparation, communication in the region.

This work involves about 50 people in Euratom associations and European industry.

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* European ITER Site Studies, under European Fusion Development Agreement.

ITER SITE SELECTION STUDIES IN SPAIN

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The purpose of the paper is to present the studies that have been carried out for the evaluation and selection of a possible site in Spain for ITER construction.

The ITER Engineering Design Activities (EDA), finished in July 2001, have considered a “generic design” for ITER with the technical requirements to be fulfilled by the selected site for ITER construction. Some assumptions have been made concerning ITER site until the actual one is selected in order to carry on design aspects and costs.

Based on these requirements, we have applied these criteria over the whole country so that two possible areas with good generic conditions for hosting ITER have been identified. Both are in the Mediterranean coast, within Catalonia and Valencia respectively.

On the technical aspects, the following items have been studied for several concrete proposals in the areas identified: land geotechnical characterization, seismic studies, power supply, cooling water supply, transport considerations for the largest ITER components.

In addition to this selection process, other important aspects relevant for a possible Spanish candidature for hosting ITER have been preliminary assessed. Particularly, licensing issues and socio-economical aspects will be presented.

As no specific legislation on fusion installations exists in the Spanish law, the licensing process that is being proposed to the Spanish regulatory body is particular for the ITER project. According to the law ITER could be classified as a second category radioactive facility. But, considering the complex installation that ITER represents, the inventory of activity, the handling of radioactive wastes and so on, the ITER licensing process could be an adaptation of the one applied to a nuclear installation and radioactive installation of first category. This means a multi-stage process to get authorizations and permits, the tentative schedule of which will be included.

Socio-economic studies have been performed up to now as preliminary ones, and show that the requirements imposed by ITER project are completely fulfilled: communications and infrastructure, society, education, health, housing, leisure and culture. Also studies on public perception will be referred for the selected site.

As a summary, the selection process followed for choosing the best possible site in Spain for hosting ITER will be described, as well as the preliminary studies carried out for assessing the licensing issues and the socio-economic features of the area.

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ITER SITE SELECTION STUDIES IN CANADA

Honourable Bob Rae

Member, Iter Canada Host Inc. Board of Directors

The Government of Canada submitted the Canadian Offer to Host Iter in Moscow in June of 2001. Mr. Rae's audio-visual presentation will discuss the attributes of the Canadian Offer, including a description of the technical and socio-cultural characteristics of the Clarington site, the licensing and environmental assessment process underway, and the unique Canadian private-public sector partnership approach to Iter in Canada.

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JAPANESE SITE FOR ITER: ROKKASHO

Japanese ITER Site Forum* (presented by S. Konishi[a])

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ITER is a critical path in the fusion energy development and the Atomic Energy Commission of Japan authorized ITER as the principal device in the third phase basic program of the fusion development in Japan. Aiming to construct a fusion power plant, a set of detailed technical objectives were defined by the participant Parties in the Engineering Design Activity (EDA). During the Activities, the design of the device and R&D of key elements were developed. The ITER design was, however, based on some assumptions, i.e., the Site Requirements and the Site Design Assumptions, because the ITER site was not identified during the EDA.

Japanese Government has proposed an ITER site, Rokkasho in Aomori Prefecture, a northern part of the main island. This site satisfies the Site Requirements and The Site Design Assumptions. In addition, the ITER design has a great potential flexibility over the detailed technical objectives, such as the capability of long pulse and high beta operations which are key elements in a future fusion power plant. Japan has selected the candidate site based on the investigation of a comprehensive scenario for the construction, operation and de-commissioning of the ITER device, by taking account of the flexibility.

For the construction, industrial infrastructure is an important item. The Rokkasho area has sufficient experiences in the construction of large nuclear and energy facilities. The transportation capability of large and heavy components has a great impact on the machine assembly and schedule. An analysis suggests that the Rokkasho site has a sufficient capability for transportation of the 1000 ton class components which are more than the design assumptions. Consequently it has more freedom in the transportation. The seismic design and the development of seismic isolators have been conducted. The site specific design adaptation is underway by taking account of possible extensions in systems in the operation phase, e.g. the heat rejection system and the electric power supply with a local energy storage. For decommissioning, this area has sufficient experiences in the fuel cycle facilities. Details in the technical assessments will be presented in those technical items.

The socio-cultural environments are the key items for the scientist and engineers who will participate in the Project. Such non-technical aspects will be also included in our presentation.

* composed of Japanese Government (Ministry of Education, Culture, Sport, Science and Technology), Aomori Prefecture, and Japan Atomic Energy Research Institute

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Author's Index

Finn Fusion

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